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FEBRUARY 1969

**SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR
DEVELOPMENT PROGRAM
TWENTY- NINTH QUARTERLY REPORT
NOVEMBER 1968 – JANUARY 1969**

U.S. ATOMIC ENERGY COMMISSION
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BREEDER REACTOR DEVELOPMENT OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

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SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

Twenty-Ninth Quarterly Report
November 1968 - January 1969

W. E. Baily
W. E. Baily, Project Engineer
Fast Ceramic Reactor Development Program

E. L. Zebroski
E. L. Zebroski, Manager
Sodium Reactor Technology

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TABLE OF CONTENTS

		<u>Page</u>
SECTION I	INTRODUCTION	1
SECTION II	SUMMARY	1
	2.1 Task B—Defected Fuel Behavior	1
	2.2 Task C—Safety Experimentation	3
	2.3 Task E—Fuel Performance Evaluation	3
	2.4 Task F—Integral Fast Flux Irradiation	4
	2.5 Task G—Core Neutronics, Thermal-Hydraulics, and Safety Analysis	4
	2.6 Task L—Cladding Development	4
SECTION III	TASK B—DEFECTED FUEL BEHAVIOR	11
	3.1 General	11
	3.2 Defected Fuel Testing (Series B8 and B9)	12
	3.3 "Soda Pop" Tests	20
	3.4 Capsules B8C and B8D (Fuel, Sodium, Cladding Compatibility Studies)	21
SECTION IV	TASK C—SAFETY EXPERIMENTATION	22
	4.1 General	22
	4.2 Axially-Restrained Fuel Experiments	23
	4.3 Axial Fuel Relocation Experiments	29
	4.4 Defected or Sodium-Bonded Fuel Experiments	30
	4.5 Flow Blockage Experiments	30
SECTION V	TASK E—FUEL PERFORMANCE EVALUATION	41
	5.1 General	41
	5.2 Center Temperature Limits	41
	5.3 High Burnup Irradiations	42
	5.4 Physical Property Studies	42
	5.5 Integral Thermal Flux Irradiations	45
SECTION VI	TASK F—INTEGRAL FAST FLUX IRRADIATIONS	51
	6.1 General	51
	6.2 Medium Power, Long Term Irradiations	51
	6.3 Low Power Fuel Swelling Irradiations	59
	6.4 High Power Intermediate Burnup Irradiations	59
	6.5 Elevated Cladding Temperature Tests	59
	6.6 Unencapsulated Fuel Pin Irradiations	60
SECTION VII	TASK G—CORE NEUTRONICS, THERMAL HYDRAULICS AND SAFETY ANALYSIS	61
	7.1 General	61
	7.2 Nuclear Data and Group Constants	61
	7.3 Core Analysis	64
	7.4 Transient and Safety Analysis	67
	7.5 Thermal Hydraulics and Fuel Analysis	68

TABLE OF CONTENTS (Continued)

	<u>Page</u>
SECTION VIII	
TASK L—CLADDING DEVELOPMENT	81
8.1 General	81
8.2 Material Irradiation Performance	82
8.3 Cladding Swelling Program	93
8.4 Materials Management and Specification Program	99
REFERENCES	99
APPENDIX	101
CONTRIBUTIONS	105
DISTRIBUTION	106

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
3-1	Schematic of 3-Pin, Forced-Convection Capsule, Gas De-entrainer	13
3-2	Mockup of 3-Pin, Forced-Convection Capsule, Gas De-entrainer	14
3-3	Capsule E1J-B1 Fuel Pin	15
3-4	Post-irradiation Appearance of Notched 316 Stainless Steel Clad From Fuel Pin E1J-B1	18
3-5	Sodium Fill Loop	19
3-6	Notched Cladding Torque Defecting Test Rig	20
4-1	Conceptual Sketch, Series IV, Transient Capsule	24
4-2	Conceptual Design of TREAT Capsule for Series IV Pre-irradiated Specimens	25
4-3	Conceptual Design of a TREAT Capsule for Task C Series IV Pre-irradiated Fuel Specimens	27
4-4	Sodium Fill System	31
4-5	C5B Transverse Macro Specimens with Neutron Radiograph Showing Locations	33
4-6	C5B Transverse Macro Specimens with Neutron Radiograph Showing Locations	35
4-7	Beta-Gamma Autoradiograph (Top) and Macrograph (Bottom) of Fuel/Blanket Interface Specimen	37
4-8	Micrographs of Longitudinal Fuel/Blanket Interface Specimen	39
5-1	Capsule E2J Fuel (125,000 MWd/Te at 22 kW/ft)	43
5-2	Capsule E5B Fuel (135,000 MWd/Te at 22 kW/ft)	46
5-3	Micrograph From Capsule E5B Showing Fuel-Cladding Interaction Zone	47
5-4	Microprobe Specimen Current and X-Ray Pulse Images of Fuel-Cladding Interaction Region of Figure 5-3	48
5-5	Microprobe X-Ray Images of Alloying Elements Within Fuel-Cladding Interaction Region of Figure 5-4	49
6-1	Transverse Cross-Section From Fuel Pin F2C	52
6-2	Transverse Cross-Section From Fuel Pin F2R	52
6-3	Transverse Cross-Sections From Fuel Pin F2H	53
6-4	Fission Products Typically Observed in Cracks Near Fuel Periphery (Fuel Pin F2T)	55
6-5	Post-Irradiation Examination of Fuel Pin F2O	57
7-1	Comparison of Measured and Computed Pu-239 σ_f Values	65
7-2	Fully Developed Circumferential Cladding Temperature Difference (Linear Power = 15 kW/ft)	69
7-3	Edge Rod and Coolant Geometry	70

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>	<u>Title</u>	<u>Page</u>
7-4	Radial Geometry and Node Structure	71
7-5	Cladding Temperature at Various Distances From Inlet	73
7-6	Cladding Temperature for Unsupported Pin	74
7-7	Cladding Temperature at 11.50 inches From Inlet	75
7-8	Predicted Fuel Swelling Due to Intragranular Bubbles at a Linear Power of 16 kW/ft as a Function of Bubble Size and Temperature. [Radius of the Fuel (r_f) = 0.11 inch, Diffusion Constant (D) = 6.6×10^{-6} Exp. $(-42200/RT)$ cm ² /sec.]	78
7-9	Predicted Size of Bubbles Reaching Grain Boundaries as a Function of Fuel Temperature and ϕ , where ϕ = Distance Travelled Times Probability of Coalescence Times 6.29	79
8-1	Effects of Deformation Rate on Ring Ductility at 1300° F For Irradiated Carbide Agglomerated Type 304 and 316	85
8-2	Comparison of Post-Irradiation Ductility as Measured in Ring Ductility and Tensile Tests for Various Fluence Levels at 1300° F	86
8-3	Tensile Specimen Cut From XG05 Hexagonal Tube	87
8-4	Tensile Properties of the XG05 Hexagonal Tube	89
8-5	Transmission Electron Micrograph From Gage Section of Failed Specimen No. 218 (Mill-Annealed Type-304)	91
8-6	Surface Microstructure of Failed Specimen No. 218 (Mill-Annealed Type-304)	91
8-7	Surface Microstructure of Failed Specimen No. 211 (Mill-Annealed Type-304)	92
8-8	Surface Microstructure of Failed Specimen No. S7 (Carbide Agglomerated Type-304)	94
8-9	Radiation History and Sectioning Plan for a Representative XG05 Fuel Pin Capsule Tube (F2T)	96
8-10	Volume Change in F2 Fuel Pin Capsule Tubes as a Function of Diametral Change	98

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
2-1	Task B—Summary of Current Mixed Oxide Capsule Irradiations	2
2-2	Task C—Current Experimental Test Series	5
2-3	Task E—Summary of Current Mixed Oxide Capsule Irradiations	7
2-4	Summary of Task F Fast Flux Fuel Testing Program in EBR-II	9
2-5	Summary of Task L Materials Irradiated in EBR-II	10
3-1	Fuel Pin Design Parameters for Capsule E1J-B1	17
3-2	Candidate Alloys for Meltable Defect	20
3-3	"Soda Pop" Tests	21
5-1	Fe, Cr, and Ni Concentrations at Selected Points in the Cladding-Fuel Reaction Zone; Capsule E5B	50
6-1	Percent Gas Release from Group 2 Fuel Pins	56
7-1	Comparison of Integral Quantities for ZPR-III Assembly 48, as Measured and Computed with the MC ² and the ENDRUN Codes	62
7-2	Computed Fluxes and Adjoint Fluxes at the Center of Assembly 48, Using ENDRUN Approach	63
7-3	Results of Neutronics Calculation for the 1000 MWe LMFBR Reference Design Using Alternate Cross Section Sets	66
7-4	Temperature Differentials for an Edge Pin	76
8-1	1300° F Ring Ductility Results for Carbide Agglomerated Types 304 and 316	84
8-2	Tensile Properties of the XG05 Hcx Tube	88
8-3	Certified Chemistry of Type 304 Tubing, Heat No. 136272	95
8-4	Density Measurements of F2 Fuel Pin Capsule Tubes	97

SECTION I

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 are:

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

ceramic fuels operating at high specific power.

Progress during the period November 1, 1968 to January 31, 1969 on the currently active tasks of this program is described in subsequent sections.

This is the twenty-ninth 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.

SECTION II

SUMMARY

2.1 TASK B—DEFECTED FUEL BEHAVIOR

The current status of the program of irradiation of mixed oxide fuel capsules in this task is summarized in Table 2-1.

All parts and major components for capsule B9A (3-pin forced convection capsule) have been completed and the capsule is presently being assembled for sodium fill.

Installation of the sodium loop and accessory equipment for the sodium fill system is complete. The loop secondary containment is presently being installed.

An in-pile test of the reference notched cladding defect mechanism for capsule B9A fuel pins proved negative. The defecting mechanism

was, therefore, changed to the back-up defect mechanism consisting of a soluble-melttable plug placed in the cladding.

Three "Soda Pop" tests were completed using 1500 gm of NaK at 1100°F to 1200°F. Conditions tested simulated both reactor operation and shutdown. No serious damage appeared to be incurred by the simulated core components.

Work was initiated on developing an on-demand defecting mechanism for subsequent forced convection capsules.

Capsule B8C completed its scheduled irradiation. Neutron radiographs of the capsule fuel pins (sodium bonded) indicate continued expansion of fuel into the plenum regions of the pins. Fuel pin B8C-1b shows approximately

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

Designation		Purpose and Major Parameters	Design Parameters						Operating Conditions				Status
			Fuel			Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Rating (kW/ft)	Burnup (MWd/Te)			
			Diameter (inch)	Length (inch)	Density (Smear) (% TD)						Fuel-Cladding Gap (Diameter inch)	PuO ₂ %	
B3C	Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	88	0.003	25	2.00	Type 347 stainless steel	V-Raft GETR	1250	21	45,000	Examination completed.
B3D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	85	0.006	25	1.97	Type 316 stainless steel	V-Raft GETR	1100	18	6,700	Stored
B3E	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.213	23	83	0.008	25	1.98	Type 316 stainless steel	V-Raft GETR	1100	18	8,700	Stored
B3F	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.212	23	84	0.008	25	1.98	Type 316 stainless steel	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	84	0.007	25	1.98	Type 316 stainless steel	V-Raft GETR	1100	15	47,500	Examination completed.
B8B	Fuel-sodium compatibility. Sodium-bonded fuel central temperature measured versus exposure time. Two pins.	0.216	2	80	0.010	25	2.00	Type 347 stainless steel	V-Raft GETR	800	14	14,500	Examination completed.
B8C	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel-sodium-clad compatibility. Two pins.	0.144	2.2	86	0.006	25	1.97	Type 316 stainless steel	V-Raft GETR	1100	21	113,500	Irradiation completed
		0.204	2.25	84	0.017*	25	1.97					57,000	
B8D	Fuel-sodium compatibility versus density and stoichiometry. Three pins.	0.214	1.75	80-88	0.0065	40	1.97	Type 316 stainless steel	V-Raft GETR	950	17.5	38,000	Irradiation in progress.
		0.214	1.75	80-88	0.0065	25	1.97					40,600	
			0.214	1.75	80-88	0.0065	25	1.94					

*Large gap required to accommodate sampling device

10% $\Delta V/V$ expansion (113,500 MWd/Te burnup) while fuel pin B8C-2 shows approximately 7% $\Delta V/V$ expansion (57,000 MWd/Te burnup).

Interim neutron radiographs of capsule B8D fuel pins (sodium bonded) indicate fuel expansion into the plenum regions of the pins similar to that observed in B8C fuel pins. Fuel expansion values indicate increasing sodium to fuel reactivity with increasing Pu content of the fuel and increasing O/M ratio in the fuel.

2.2 TASK C—SAFETY EXPERIMENTATION

The current status of the program of irradiation of fuel specimens to establish their safety during transient high rate irradiation is summarized in Table 2-2.

Two fuel specimens, destined for transient irradiation in TREAT, have reached target burnup in EBR-II (~50,000 MWd/Te) and will be removed for non-destructive examination prior to re-encapsulation. Conceptual design of the TREAT capsule for these specimens has been completed. The design includes pressure, flow, and temperature monitors and is based on the need to sodium fill and weld remotely in the hot cells.

Destructive examination of the unfailed molten fuel experiment has verified the extent of melting (~35%) during the transient and the extrusion of molten fuel into the annular upper blanket, a distance of some 10 inches. This allowance for fuel expansion apparently prevented the cladding failure which occurred in a parallel test with a solid blanket.

Fabrication was continued on two 3-pin capsules designed to compare defected or sodium-bonded pellet and powder fuel under transient overpower conditions.

Heaters and instrumentation for the out-of-pile flow coastdown experiment were received and the completed portions of the test rig stored for later assembly.

2.3 TASK E—FUEL PERFORMANCE EVALUATION

The current status of irradiation tests in this task is shown in Table 2-3.

Instrumented capsule E1AA, measuring the temperatures of mixed oxide fuel pellets during GETR operation at 10-15 kW/ft, yielded data to about 18,000 MWd/Te. At that burnup, the tungsten-rhenium thermocouple at the fuel center failed, and the capsule was removed.

The initial E1J capsule, intended to measure the effective thermal conductivity of restructured fuel pellets, operated successfully in its planned 88-hour irradiation cycle. The capsule is being sectioned to establish the amount of melting, after which additional remaining capsules will be irradiated.

Capsule E2J containing pins with both solid and annular mixed oxide pellets of 89% smeared theoretical density was examined after operating at 22 kW/ft to 125,000 MWd/Te. No significant behavioral differences in dimensions, microstructures, or gas releases were noted between the two pellet configurations.

Capsule E2P with 15 mil thick cladding was rebuilt after examination indicated that fuel melting was responsible for its having a greater diametral increase than E2Q with 10 mil thick cladding. The re-encapsulated unit, intended to study the effect of cladding restraint on fuel swelling, is operating successfully.

Microprobe examination of the extensive fuel-cladding reaction phase of E5B showed it to be primarily Fe, Ni, and Cr, with small concentrations of Cs, Ba, Pd, and Mo. No fission products were found within the cladding. Metallic-appearing fission products were identified as containing Fe, Ni, and Cr along with Mo and Ru.

Capsules E5C, E5D, and E5E comparing solid and annular pellets and powder at high

burnup are continuing irradiation. Burnup is estimated at 120,000 MWd/Te.

Capsule E5F measuring mixed oxide fuel linear expansion with a transducer operated to 44,000 MWd/Te prior to removal from GETR because of an open circuit. Continuous monitoring of fuel movement was achieved during reactor exposure, indicating differential fuel-cladding interaction as well as fuel swelling.

2.4 TASK F—INTEGRAL FAST FLUX IRRADIATION

The current status of the irradiation of fuel pins in EBR-II under the fast flux fuel testing program is summarized in Table 2-4.

Approval to irradiate the unencapsulated Group 9A fuel pins was granted by ANL. This is the first thirty-seven unencapsulated pin subassembly of 0.250-inch diameter accepted under the PA 10 program.

The sixteen 0.290-inch diameter unencapsulated Group 3A fuel pins have reached a burnup of 20,900 MWd/Te.

A subassembly design for high temperature cladding tests in EBR-II was agreed upon by GE and ANL-RE. The first pins to be irradiated in the subassembly will be the Group 10B.

The thirty-seven Group 9D fuel pins were successfully xenon tagged as requested by ANL.

2.5 TASK G—CORE NEUTRONICS, THERMAL-HYDRAULICS, AND SAFETY ANALYSIS

A comparison of calculations for ZPR-III, Assembly 48 using the rapid cross section adjusting routine (through the ENDRUN code) with results from MC² calculations performed at ANL, show generally good agreement. Both computations followed the specifications of the Data Test Subcommittee of the Cross Section Evaluation Working Group. The ENDRUN calculation of k_{eff} was within 0.3% of the MC²

results, and all fission and capture ratios were within 2%. As expected, central worths of light materials showed the largest disagreement. This comparison indicates that the rapid adjusting technique will be a useful tool for testing of ENDF/B data.

Testing of the code, DOT2DB, which incorporates 2-dimensional diffusion and transport theory flux solutions into one code, has indicated that convergence of the transport calculation is significantly enhanced by preceding it with a diffusion theory computation. This increase in efficiency was the major reason for combining the two techniques into one code.

Comparisons of the fuel cycle codes BICYCL and RAPFU have shown excellent agreement, indicating the BICYCL equations are correct. Experience with BICYCL has revealed it to be very flexible, allowing the user many choices on fuel management schemes while retaining fairly simple input preparation.

A computerized model has been set up on THT to study the transient temperature distribution in a fuel pin during accident conditions.

The temperature distribution around the edge fuel pin of a bundle was studied with a 3-dimensional model.

Using the available experimental data, a method was formulated to predict the pressure drop in a spiral wire wrapped rod bundle.

A statistical method has been developed and tested to analyze the Task G irradiation data.

2.6 TASK L—CLADDING DEVELOPMENT

The current status of materials irradiation capsules in EBR-II is shown in Table 2-5.

Irradiation of L-6, L-14, and L-15 materials capsules is continuing in the EBR-II. Ten L-14 material capsules and L-2I have been discharged from EBR-II after reaching their target

TASK C 29th QUARTERLY

TABLE 2-2
TASK C—CURRENT EXPERIMENTAL TEST SERIES

Fuel Specimen	Description*	Fabrication	Pre-Irradiated**		TREAT-Transient Irradiated		Nondestructive Examination†	Destructive Examination	Test Objective
			Yes	No	Calibration	Test			
C4A	UO ₂ -20% PuO ₂ fuel 90% smeared density pelleted fuel	C		X	C	C	C	C	Axial restraint at fuel melting (~ 5000°F).
C4B	Identical to C4A	C		X	C	C	C	C	Axial restraint similar to C4A. Test to identify vol % melting necessary to cause cladding failure.
C4C	UO ₂ -20% PuO ₂ fuel 80% smeared density pelleted fuel	C		X	C	C	C	C	Axial restraint similar to C4A. Test to observe effect of fuel density on incipient failure point identified in C4A.
C4D ⁺⁺	Identical to C4C	C		X	C	C	C	C	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).
C4E (57A)	UO ₂ -28% PuO ₂ fuel 80% smeared density pelleted fuel	C	P		P	P	<div><div>P⁺⁺⁺</div><div>P</div></div>	P	Axial restraint designed to simulate the restraint of C4A after steady-state irradiation is complete. Specific transient test goals will be determined after nondestructive examination of the fuel pin, following its removal from EBR-II.
C4F (F7B)	Identical to C4E	C	P		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4G (F7C)	UO ₂ -28% PuO ₂ fuel 90% smeared density pelleted fuel	C	NC		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4H (F7D)	Identical to C4G	C	NC		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4I ⁺⁺	UO ₂ -20% PuO ₂ fuel 85% dense powder fuel	C		X	C	C	C	C	Axial restraint similar to C4A. Test transient is intended to be similar to that of C4D.
C4J	Identical to C4I	C		X	C	C	C	C	Axial restraint similar to C4A. Test transient is intended to be similar to that of C4A.
C4K (F7E)	UO ₂ -28% PuO ₂ fuel 85% dense powder fuel	C	P		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4L (F7F)	Identical to C4K	C	P		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C5A	10% enriched UO ₂ 90% smeared density pelleted fuel	C	C		C	C	<div><div>C</div><div>C</div></div>	I	Test to study mobility of transient melted fuel: investigate potential for fuel relocation under prototypical conditions of length, restraint, and fission gas inventory.
C5B	Identical to C5A	C	C		C	C	<div><div>C</div><div>C</div></div>	NC	Same as C5A.
C6A	10% enriched UO ₂ - 3 Pins Na-bonded pellet and powder, gas bonded pellet	I		X	P	P	<div><div>P</div><div>P</div></div>	P	Compare transient performance of Na-bonded/logged fuel to gas-bonded fuel.
C6B	Identical to C6A	I		X	P	P	<div><div>P</div><div>P</div></div>	P	Identical to C6A.

P—Planned
I—Initiated
NC—Nearly Completed
C—Completed

*—O/M Ratio 1.97 to 1.985 for all cases
**—~ 50,000 MWd/Te—EBR-II fast flux for Series IV specimens; ~ 30,000 MWd/Te—GETR thermal flux for Series V specimens
†—Consists of x rays, neutron radiographs, deencapsulation, cladding diametral measurements, and γ-transmission scan
++—Contains a small, lower fuel assembly during transient irradiation. This lower assembly contains a W-5% Re, W-26% Re in-fuel thermocouple

+++

Before Transient

After Transient

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

Capsule	Description	Design Parameters						Operating Conditions						Status
		Diameter (inch)	Fuel Length (inch)	Density, % TD		Composition		Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)		
				Pellet	Smearcd	PuO ₂ %	O/M							
E1AA	Continuous in-pile thermal conductivity.	0.219	6.0	92.0	91.0	20	1.97	Type 316 stainless steel.	GETR	1300	15	0	Examination in progress.	
E1B-1	Gap conductivity. Four pins with diametral gaps ranging from 2 to 10 mils.	0.218	1.25	95.9	92.5	20	1.98	Type 347 stainless steel	GETR	—	20	500	Report in publication.	
2		0.218	1.25	94.2	94.2	20	1.98		GETR	974	20	500		
3		0.218	1.25	96.1	85.0	20	1.98		GETR	956	20	500		
4		0.218	1.25	96.2	92.5	20	1.98		GETR	969	20	500		
E1C-1	Relative smeared conductivity. Two compartments in each capsule. Effect of stoichiometry.	0.218	0.8	95.7	93.8	0	2.00	Type 347 stainless steel	MTR	1170	39	< 100	Report issued.	
2		0.218	0.8	97.4	96.1	20	2.00		MTR	1170	39	< 100		
E1D-1	Relative smeared conductivity. Two compartments in each capsule. Effect of stoichiometry.	0.218	0.8	96.3	94.5	20	2.00	Type 304 stainless steel	MTR	1355	46	< 100	Report issued.	
2		0.218	0.6	96.9	94.8	20	1.98		MTR	1355	46	< 100		
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	Type 347 stainless steel	MTR	875	20	85,000	Data analysis in progress.	
E1F		0.218	1.9	94.6	94.8	20	2.00		MTR	654/875	20/28	77,000		
E1G-1	Relative smeared conductivity. Two pins different density.	0.217	0.8	93.4	90.9	20	1.97	Type 304 stainless steel	MTR	1031	34	< 100	Report issued.	
2		0.218	0.8	87.2	85.4	20	1.97		MTR	1031	34	< 100		
E2C-1	High burnup. Reference fuel irradiated at near reference thermal conductivity.	0.218	5	84.5	82.2	25	2.00	Type 347 stainless steel	GETR	1060	20	100,000	Report in progress.	
2		0.218	5	94.3	91.4	25	2.00		GETR	1060	20	100,000		
E2D-1	High burnup. Reference fuel irradiated at near reference thermal conditions. Two pins in each capsule	0.150	5	90.1	87.5	25	2.00	Type 316A stainless steel	GETR	1040	17	120,000	Report in review.	
2		0.150	5	91.6	88.3	25	1.96		GETR	1040	17	120,000		
E2E	Fuel swelling. Four separate pins at three power levels.	0.100	3	98.5	95.4	25	2.00	Type 304 stainless steel	MTR	785	11.6	199,000	Report in progress.	
F		0.100	3	96.9	96.5	25	2.00		MTR	655	9.4	172,000		
G		0.100	3	97	95.6	25	2.00		MTR	560	7.9	255,000		
H		0.100	3	96.6	94	25	2.00		MTR	785	11.6			
E2J-1	High burnup. Annular versus solid pellet.	0.218	5	90.4	89.5	25	1.98	Type 316 stainless steel	GETR	1100	22	120,000	Report in progress.	
2		0.218	5	96.0	88.5	25	1.98			1100	22			
E5A-1	Powder and pellet fuel comparison at low burnup.	0.218	4.9	86.8	84.3	25	2.00	Type 347 stainless steel	GETR	1100	27	2,000	Report in review.	
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 high burnup.	0.218	4.9	85.8	84.1	25	2.00	Type 347 stainless steel	GETR	1100	20	100,000	Report in review.	
2		0.218	4.9	84.1	84.1	25	2.00		GETR	1100	20	100,000		
E5C-1	Powder, pellet, and annular fuel comparison at high burnup.(Pellet)	0.215	16.0	87.5	82.9	25	1.96	Type 316 stainless steel	GETR	1050	19.5	115,000	Irradiation in progress.	
E5D-1	Powder, pellet, and annular fuel comparison at high burnup.(Powder)	0.218	16.0	81.8	81.8	25	1.97	Type 316 stainless steel	GETR	1050	19.5	115,000	Irradiation in progress.	
E5E-1	Powder, pellet, and annular fuel comparison at high burnup.	0.215 (Annular)	16.0	90.5	83.3	25	1.96	Type 316 stainless steel	GETR	1050	19.5	115,000	Irradiation in progress.	
E5F	Linear fuel movement versus burnup.	0.214	10.0	92.0	87.1	25	1.97	Type 316 stainless steel	GETR	850	15	44,000	Data Analysis in progress.	
E2P	Effect of radial restraint on fuel swelling.	0.219	2.0	92.0	91.0	25	1.98	Type 316 stainless steel	GETR	1160	15	500	Irradiation in progress	
E2Q	Effect of radial restraint on fuel swelling.	0.229	2.0	92.0	91.0	25	1.98	Type 316 stainless steel	GETR	1160	10	500		
E1J-1	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	1.93	Type 316 stainless steel	GETR	1100	24	~ 2000	Irradiation in progress.	
E1J-2	Effect of restructuring on smeared thermal conductivity	0.217	5.0	92.0	89.6	25	1.93	Type 316 stainless steel	GETR	1100	24	~ 2000	Fabrication in progress.	
E1J-3	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	1.93	Type 316 stainless steel	GETR	1100	24	~ 2000	Fabrication in progress.	
E1J-4	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	1.97	Type 316 stainless steel	GETR	1100	24	~ 2000	Fabrication in progress.	
E1J-5	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	1.97	Type 316 stainless steel	GETR	1100	24	~ 2000	Fabrication in progress.	
E1J-6	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	2.00	Type 316 stainless steel	GETR	1100	24	~ 2000	Fabrication in progress.	
E1J-7	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	2.00	Type 316 stainless steel	GETR	1100	24	~ 2000	Examination in progress.	
E1J-8	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	2.00	Type 316 stainless steel	GETR	1100	24	~ 2000	Fabrication in progress.	
E1J-9	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	89.6	25	2.00	Type 316 stainless steel	GETR	1100	24	~ 2000	Fabrication in progress.	

TABLE 2-4

SUMMARY OF TASK F FAST FLUX FUEL TESTING PROGRAM IN EBR-II

January 31, 1969

Group	Purpose and Major Parameters	Row	Number		Power kW/ft	Target MWd/Te	Current Burnup MWd/Te	Irradiation	
			S/A	Pins				Finish*	Status
F-0	Low power swelling rate of high density fuel at various burnup levels.	7	XG02	1	5-6	40k	40k	8/68	Discharged from EBR-II core.
			XG03	2		60k	49k	8/70	In reactor.
			XG04	2		100k	49k	5/73	In reactor.
F-1	Short time performance and proof tests.	4	1	6	14-16	2k	1.8k	5/65	Reported in GEAP 5570.
F-2	Medium power scoping ir- radiations. Major parameters include smear density, stoichiome- try, cladding coolant inter- actions and burnup.	4	XG06	12	11-16	50k	51.1k	4/67	Post examination completed.
			XG05	5	11-16	75k	74k	6/68	Post examination in progress. At EBR-II for incorporation in new S/A.
				4		100k	70k		
F-3A	Bare-pin irradiations to evaluate fuel cladding and cladding coolant interactions.	5	XO40	16	15-18	50k	20.9k	10/69	In reactor.
F-3E (E1H)	Density and stoichiometry effects for encapsulated fuel pins.	4	XO27	18	13-16	40.80k 120k	40.2k	4/69, 4/70 5/71	In reactor.
F-4	Low density fuel behavior at high heat ratings.	2	XO11	9	15-18	50k	35k	6/67	Post examination in progress. Discharged from EBR-II core 12/68 for interim examination.
		3	XO39	7	15-18	50k	36.9k	2/69	
F-5	Fuel performance at peak FCR cladding temperature.	4	1	16	15-16	100k	--	3/71	Fuel pin fabrication in progress.
F-6	Fuel/cladding interactions under accelerated swelling.	7	XO10	4	7-9	50k	50.5k	6/69	In reactor.
F-7	Pre-irradiated axially re- strained fuel prior to tran- sient test. Main variable is fuel density.	4	XO15	2	12-15	50k	56.6k	3/69	In reactor. Fuel pins at EBR-II site.
			1	4	12-15	50k	--	12/69	
F-8A	Effect of density on fuel swelling rate.	6	XO19	7	6-9	50k	33.3k	5/70	In reactor.
			XO20	9		100k	34.0k	12/72	
F-8B	Effect of void deployment on fuel cladding interaction.	7	XO36	19	6-9	50k	11.8k	1/71	In reactor.
F-9A	High power, high burnup, un- encapsulated fuel pin irra- diations.	4	XO43	37	11-16	100k	--	3/71	Fuel pins at EBR-II site.
F-9B	Medium power, high burnup, un- encapsulated fuel pin ir- radiations.	4	1	37	10-13	100k	--	3/71	Fuel pin fabrication in progress.
F-9C	High power, high burnup, un- encapsulated fuel pin irra- diations.	4	1	37	13-16	100k	--	3/71	Fuel pin fabrication in progress.
F-9D	Medium power, high burnup, un- encapsulated fuel pin ir- radiations.	4	1	37	8-10	100k	--	3/71	Fuel pin fabrication in progress.
F-10B	Cladding performance at peak FCR cladding temperatures	4	1	19	14-17	100k	--	--	Fuel pin fabrication in progress.

* Based on EBR-II operation at 45 MW as of 6/1/68 and at 50% load factor.

TABLE 2-5

SUMMARY OF TASK L MATERIALS IRRADIATED IN EBR-II

Series	Row-S/A	Capsules	Capsule Materials	Number of Specimen		Total Fluence			Projected Completion Date (2)
				Biaxial	Tensile	Target;	December 1968		
						10 ²³ nvt	10 ²³ nvt (3)	10 ²³ nvt (4)	
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	0.25	February 1967
L-4	4—XO09	2	Type 316 stainless steel.	10	20	0.20	0.19	0.15	December 1966
L-4	2—XO14	5	One each of Types 304, 321, and 347 stainless steels; and two of Incoloy-800.	25	50	0.20	0.17	0.10	April 1967
L-10	4—XO16	10	One each of Types 321 and 347 stainless steels; two of Type 304 stainless steel; three each of Type 316 stainless steel; and Incoloy-800.	70	60 ⁽¹⁾	0.20	0.29	0.22	October 1968
L-14	4—XO25	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 ⁽¹⁾	0.20	0.31	0.23	December 1968
L-21	4—XG05 and XO39	1	Type 321	3	16	0.76	0.61	0.45	December 1968
Irradiation in Progress									
L-2	4—XG05	4	One each of Types 304, 316, and 347 stainless steels; and Incoloy-800.	12	64	0.76	0.47	0.34	July 1969
L-6	2—XO18	9	One each of Types 321 and 347 stainless steels; two each of Type 304 stainless steel and Incoloy-800; three of Type 316 stainless steel.	63	126 ⁽¹⁾	1.00	0.50	0.29	September 1970
L-15		4	All five alloys:	None	288	0.20	0.14	0.10	May 1969
MT-3 ⁽⁵⁾ and 4	4—XA08	2	32 Incoloy-800; 32 Inconel-625 and 32 Hastelloy-X Tensile Samples.	None	96	1.00	0.59	0.43	December 1970

- 1: Includes specimens of experimental alloy of base compositions Fe, (9-14) Ni, (9-14) Cr; (0.0004 - 0.05) C.
- 2: Assuming EBR-II continues to operate at 45 MW with 50% load factor.
- 3: Fluence based on EBR-II core physics conversion factors.
- 4: Fluence based on flux wire analysis conversion factors.
- 5: Capsules from previous AEC program under contract AT(04-3)-189, P.A. 29.

fluence. The total neutron fluences, as of December 31, 1968, are 5.9, 4.9, 3.0, and 1.4×10^{22} n/cm², for the L-2I, L-6, L-14, and L-15 respectively.

Ring ductility testing of carbide agglomerated Type 304 and Type 316 rings irradiated to 3.5×10^{21} n/cm² ($E_n > 1$ MeV) was completed. The results show a general loss in ductility with decreasing strain rates and increasing fast-neutron fluence for both alloys. It was observed that the ring ductility was greater than comparable uniaxial tensile elongation.

A substantial increase in yield strength (from 27,500 psi to between 78,000 and 101,000 psi) and reduction in total elongation (from 27% to between 2.6% to 3.5%) were observed for the XG05 hex tube as compared to typical unirradiated values. The reduction in area, however, was not as greatly affected and ranged between 14% and 27% for the hex tube.

Transmission and replication electron microscopy examinations have been conducted on failed tensile specimens of Type 304. Two structural conditions were examined: mill-annealed and carbide-agglomerated. Large discrete intergranular carbides were observed in the carbide-agglomerated material while the mill-annealed sample exhibited a microstructure of continuous carbides in the grain boundary.

A comprehensive density survey of two XG05 fuel pin capsule tubes indicated that clad swelling is isotropic in an unstressed tube.

All seamless tubing ordered for the purpose of assessing vendor capability for providing high quality material has been received and is undergoing complete quality characterization including dimensional surveying, chemical and microstructural analyses, and mechanical property evaluation.

SECTION III

TASK B—DEFECTED FUEL BEHAVIOR

3.1 GENERAL

The purpose of Task B is to determine analytically and experimentally the manner and consequences of propagation of fuel defects, the behavior and performance of the fuel cladding interface in intimate contact with sodium, the distribution and rate of release of fission products from high performance mixed oxide fuels, 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. It appears attainable to limit long-term

activity 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. Even though vented pins are a prospective reference design for 300 MW and 1000 MW plants, funding limitations have made it necessary to terminate this aspect in future work on Task B.

3.2 DEFECTED FUEL TESTING (SERIES B8 and B9)

3.2.1 Forced Convection Capsule (Series B9)

Fabrication of the 3-pin forced convection capsule (B9A) continued. All parts and major components for the capsule have been completed, and the capsule is presently in the process of being assembled (approximately 80% complete) and prepared for sodium fill.

3.2.2 Gas De-entrainment

Design and fabrication was completed on the gas de-entrainment device to be used with the 3-pin forced convection capsule to remove any backfill gas and fission gas released from the fuel pins upon cladding failure. Figure 3-1 shows a schematic view of the capsule gas de-entrainer. The gas de-entrainer is mounted at the top of the capsule above the EM pump. The gas is de-entrained in the following manner. The sodium stream issuing from the capsule inner annulus is deflected to the side and downward into the capsule outer annulus by means of the gas de-entrainment device. The velocity of the sodium is reduced as it issues from the inner annulus and is turned by the flow deflector in the de-entrainer. This permits bubbles entrapped in the sodium to rise toward the top edge of the sodium stream at this point. The sodium stream is partitioned so a small portion at the top of the stream is directed upward into the plenum area of the capsule while the remaining and major portion of the stream passes downward into the outer annulus which leads to the capsule EM pump. The bubbles entrapped in the upper portion of the stream, therefore, tend to be carried into the plenum area where they rise to the sodium surface and are de-entrained. With this configuration, the greater the size of the bubble, the greater is its chance of being de-entrained. The vanes on the top of the gas de-entrainer prevent any large counter currents being generated in the plenum area which could carry bubbles from the plenum

back into the sodium stream. Water tests with a plastic model of the gas de-entrainer shown in Figure 3-2 indicate 10% to 30% gas de-entrainment per pass through the device.

3.2.3 Defect Mechanism for B9A

An in-pile test (E1J-B1) of the reference notched cladding defect mechanism for the 3-pin forced convection capsule fuel pin was conducted in the GETR trail cable facility. The defect mechanism, based upon thermal expansion calculations that show fuel inside of notched, tight-fitting cladding (diametral fuel/cladding gap of 0.005 in.), should expand sufficiently at 15 to 20 kW/ft to rupture the cladding at the notch.

The test capsule fuel pin was built to identical specifications to those planned for the B9A fuel pin. Figure 3-3 shows the fuel pin design used for the test. Table 3-1 summarizes the fuel pin design parameters. As shown in Figure 3-3, the defecting notch consisted of a 60° groove, 1-inch long and 0.012-inch deep (cladding wall thickness 0.015 inch) machined in the cladding wall. The fuel column was axially restrained to provide maximum radial expansion.

The capsule was irradiated at steady state for approximately 2 hours at 16 kW/ft and then removed from the reactor to obtain neutron radiographs. No indication of cladding failure could be determined from the neutron radiographs. Therefore, the capsule was reinserted in the trail cable facility and re-irradiated for three more hours at powers up to 17.5 kW/ft. The capsule was thermal cycled twice during this period. After irradiation, the capsule was transferred to the α -cell at the Radioactive Materials Laboratory for destructive examination.

Vacuum tests of the fuel pin cladding, as well as photomicrographic examination of the notched area, indicated no cladding failure resulted from irradiation of the test pin. Figure 3-4 shows the appearance of the notched cladding area after irradiation.

FIGURE 3-1. SCHEMATIC OF 3-PIN, FORCED-CONVECTION CAPSULE, GAS DE-ENTRAINER

FIGURE 3-2. MOCKUP OF 3-PIN, FORCED-CONVECTION CAPSULE, GAS DE-ENTRAINER

As a result of the non-failure of the cladding in this in-pile test, the defect mechanism incorporated in B9A was changed to a soluble-melttable plug placed in the fuel pin cladding. Such plugs have been used successfully in previous experiments (B2C, B3D, B3E, B3F, B4A, B4D). The cladding in each fuel pin was notched with a 0.012-inch deep, 60° groove that is 1-inch long. Three 0.032-inch diameter holes were drilled through the cladding in the notched area. These holes were filled with solder consisting of 75% Pb, 15% Zn, and 10% Sn which has a melting point of 514°F. Less than 20 mg of solder was used to manufacture both defects. The dissolution of this solder in the capsule sodium will result in an added impurity level of 15 ppm Pb, 3 ppm Zn, and 2 ppm Sn.

3.2.4 Sodium Fill System

Fabrication and installation of the loop piping and components for the sodium fill system is complete. Figure 3-5a shows the loop piping and components prior to the addition of the

strip heaters and insulation. Figure 3-5b shows the loop after the addition of the strip heaters and insulation. For reference, the general layout and component parts of the loop are shown in Figure 3-4 of the 27th Quarterly Report.⁽¹⁾ The loop secondary containment (for protection in case of a premature failure of the defected fuel and subsequent contamination of the loop sodium) is mounted on the square frame work surrounding the loop and is presently being installed.

The loop is designed for peak operation at 1200°F. Pumping capacity is sufficient to circulate 10 to 12 gal/min through the loop or forced convection capsule. Sodium purity control is sufficient to obtain a Na plugging temperature of 250°F (< 5 ppm O₂). All loop control and monitoring of temperatures and flows are performed remotely because of the complete enclosure of the loop. A negative atmospheric pressure is maintained inside the enclosure and an automatic fire suppression system has been installed.

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING:—				REV. NO. /	TITLE	
APPLIED PRACTICES.	SURFACES	TOLERANCES ON MACHINED DIMENSIONS		106D3906 CONT ON SHEET	GENERAL ELECTRIC	106D3906 SH. NO.
		FRACTIONS	DECIMALS			
745A5481	3. ✓	+	—	106D3906 CONT ON SHEET	FUEL PIN (EN-BI)	
		+	—		FIRST MADE FOR FCC	
					NEXT ASSEM. — 161F495GI	
					GI P. No. NAME DESCRIPTION	
					1 1 END PLUG 1/4 DIA. x 2 1/2 LG. — 316 STN. STL.	
					1 2 CLAD 1/4 O.D. x .015 WALL x 5 3/8 LG. — 316 STN. STL.	
					1 3 SPACER 1/4 DIA. x 1/4 LG. — 316 STN. STL.	
					2 4 INSULATOR PELLET 163 B 8263 P3	
					20 5 FUEL PELLET 163 B 8263 P1	
					1 6 END PLUG 1/4 DIA x 1/2 LG 316 STN. STL.	

* SEAMLESS TUBING - COLD WORKED
SUPPLIED BY CUSTOMER

VIBRA-ETCH A $\frac{1}{32}$ WIDE X -
2" LONG LINE ON THIS
SURFACE. LINE IS TO BE
IN LINE WITH NOTCH
IN GLAD.

2. NEUTROGRAPH FUEL PIN AFTER FINAL WELD TO VERIFY TOTAL ACCUMULATED AXIAL GAP LESS THAN .015.
8. MEASURE FUEL COLUMN & CLAD TO INSURE TOTAL ACCUMULATED AXIAL GAP LESS THAN .015
7. BACKFILL WITH HELIUM AT 1 ATMOSPHERE.
6. CLEAN ALL STN. STL. PARTS PER APED SPEC. P4C-YP3 AND PASSIVATE PER SPEC. P4A-YP1.
5. NO NICKS OR SCRATCHES GREATER THAN 10% OF THE WALL THICKNESS WILL BE ACCEPTABLE.
4. ALL RADIOGRAPHS SHALL BE REVIEWED BY RESPONSIBLE ENGINEER.
3. ALL WELDS IDENTIFIED BY $\diamond$ ARE TO BE LEAK CHECKED WITH A HELIUM MASS SPECTROMETER. A LEAK RATE GREATER THAN 2×10^{-8} STD. CC/SEC. SHALL NOT BE ACCEPTABLE.
2. ALL WELDS IDENTIFIED BY $\diamond$ ARE TO BE RADIOGRAPHED TO DETERMINE DEPTH OF PENETRATION. MINIMUM DEPTH OF PENETRATION TO BE 90%. RADIOGRAPH IN FOUR ORIENTATIONS 45 DEGREES APART. WELD DIA. NOT TO EXCEED .256
1. $\diamond$ WELD IDENTIFICATION ONLY.

NOTES:

FIGURE 3-3. CAPSULE E1J-B1 FUEL PIN

TABLE 3-1

FUEL PIN DESIGN PARAMETERS FOR CAPSULE E1J-B1FUEL

Material	Co-precipitated (Pu, U)O ₂
Ratio Pu/Pu + U	0.25 ± 0.015
Ratio U-235/U-235 + U-238	91.5 - 94
O/M Ratio	1.98 ± 0.01
Pellet Density (theoretical)	94 ± 2%
Pellet Dimensions:	
Outside Diameter	0.220 inch Nominal
Length	0.25 inch Nominal
Fuel-Cladding Bond	Helium
Fuel-Cladding Diametral Gap (Cold)	0.0005 ± 0.0005 inch
Fuel Column Length	5 ± 1/8 inch
Total Cold Axial Gap	0.000 - 0.015 inch

BLANKET AND INSULATOR

Material	Natural
Pellet Density (theoretical)	94 ± 2%
Pellet Dimensions:	
Outside Diameter	0.215 inch Nominal
Blanket Length (above fuel)	0.25 ± 0.03 inch
Insulator Length (below fuel)	0.25 ± 0.03 inch
Diametral Blanket-Insulator/Cladding Gap (Cold)	0.005 ± 0.0025 inch

CLADDING

Material	316 stainless steel
Yield Strength (100% cold worked)	131,600 to 135,200 psi
Outside Diameter	0.250 inch Nominal
Wall Thickness	0.015 inch Nominal

FIGURE 3-4. POST-IRRADIATION APPEARANCE OF NOTCHED 316 STAINLESS STEEL CLAD FROM FUEL PIN E1J-B1

Because of the necessity of changing the reference cladding defect mechanism for B9A from the notched cladding defect mechanism to a soluble-meltable plug, the capsule fill operation was modified to prevent possible contamination of the sodium loop during filling. An intermediate sodium fill tank was installed between the sodium fill loop and the capsule. Sodium is circulated through this tank and purified by means of the sodium fill loop. The loop is then isolated from the tank, and the capsule is filled from the tank using gas pressure as the driving force. Should the intermediate tank become contaminated during the filling operation, it could be easily replaced. It is planned to fill a dummy capsule prior to filling B9A to check out the sodium fill system and capsule filling procedure.

3.2.5 Defect Mechanism Development

Work was initiated to develop an on-demand defecting mechanism to be used with subsequent forced convection capsules. Three

possible mechanisms were considered: internal gas pressurization of notched cladding, brazing of a meltable plug, and applying a torsional load to notched cladding.

A number of tests were performed to determine that 2500 to 3500 psi gas pressure are required to rupture notched 316 stainless steel cladding (0.012 in. notch in 0.015 in. wall) at 1150°F. This is approximately one-half the pressure required to rupture unnotched cladding at the same temperature.

Additional work is required to reduce or accommodate the gas released to FCC coolant after rupture of the cladding.

Table 3-2 shows a number of brazing alloys selected for further evaluation and possible testing as a meltable plug defect.

Tests were performed on the torque defect mechanism using the apparatus shown schematically in Figure 3-6. Optimum

A.

B.

FIGURE 3-5. SODIUM FILL LOOP

TABLE 3-2

CANDIDATE ALLOYS FOR MELTABLE DEFECT

<u>Braze Material</u>	<u>Melting Point (STP)</u>	<u>Remarks</u>
Al - 11.7% Si	1070°F	Binary Eutectic
Al - 13% Mg ₂ Si	1105°F	Ternary Eutectic
Al - 5.7% Ni	1184°F	Binary Eutectic
Al	1220°F	Ti Undercoat Necessary

defecting was observed with cold worked cladding containing a 0.5-inch long, 0.012-inch deep notch (cladding wall thickness 0.015 in.) oriented at 45° to the cladding axis. Under these conditions, the cladding ruptured after applying 25 inch-pounds of torque. A twist of 6° was required to rupture a 5-inch long sample containing the above defect. More development is required to establish the effect of temperature and the interdependency

of cladding length with twist angle and post-defect yielding of the cladding.

3.3 "SODA POP" TESTS

Three additional capsule rupture tests were completed. Each test contained 1500 gm of high temperature NaK with water in a simulated test reactor environment. Table 3-3 summarizes the test conditions for the three

FIGURE 3-6. NOTCHED CLADDING TORQUE DEFECTING TEST RIG

TABLE 3-3

"SODA POP" TESTS

<u>Test</u>	<u>NaK Inventory</u>	<u>NaK Temperature</u>	<u>Reaction Orifice</u>	<u>Capsule Pressure</u>	<u>Reactor Pressure</u>	<u>Core</u>
1	1500 gm	1100° F	1 in. *	50 psig	0 psig	Not Simulated
2	1500 gm	1200° F	1-1/4 in.	50 psig	0 psig	Simulated
3	1500 gm	1100° F	1-1/4 in.	60 psig	135 psig	Simulated

* 1/8 inch orifice installed but loosened during operation which resulted in NaK flowing through an equivalent 1-inch diameter area.

tests. For reference, Figures 3-5 and 3-6 in the 27th Quarterly⁽¹⁾ show the configuration of the "Soda Pop" test facilities.

In the first two tests, the NaK in the simulated forced convection capsule was pressurized to a higher pressure than the surrounding water as would be in the case while the reactor is down. In the third test, the surrounding water was pressurized to a higher pressure than the NaK in the capsule simulating capsule failure during reactor operation.

Comparison of the present 1500 gm tests with previous 100 gm tests indicate increasing duration of reaction with increasing NaK inventory. A relatively large initial pressure pulse is observed immediately after the capsule is opened by the explosive severance device. The NaK-water reaction in these tests produced a "chugging" type noise, suggesting a pulsating contact of NaK and water through the capsule orifice. In the first test, it is estimated that approximately 90% of the NaK inventory reacted in the first 2 seconds.

The magnitude of the pressure pulses generated by the reactions, however, do not appear to increase significantly with increasing

NaK inventory. In the 100 gm tests, a maximum pressure of 200 psig was observed while in the 1500 gm tests, a maximum pressure of 400 psig was observed. The 200 psi burst diaphragm on the simulated test reactor pressure vessel was ruptured in each of the 1500 gm NaK tests and water was ejected.

In the first two 1500 gm NaK tests no capsule wall deformation was observed. However, in the third test where the water pressure was higher than the capsule pressure, the capsule walls were deformed but not breached. Some pressure vessel motion was observed in the third test, probably caused by a water hammer effect. In all tests, no serious damage appeared to be incurred by the simulated core components, and control rods were operable after the tests.

3.4 CAPSULES B8C AND B8D (FUEL, SODIUM, CLADDING COMPATIBILITY STUDIES)

Capsule B8C completed its scheduled irradiation. The capsule was removed from the GETR and neutron radiographs made of the two sodium bonded pins. Both pins continued to show fuel expansion into the plenum regions. The top pin (B8C-1b), with a calculated burnup of 113,000 MWd/Te, showed approximately 10%

$\Delta V/V$ fuel expansion ($\Delta V/V$ values corrected for center void formation but not for fuel densification).

The bottom pin (B8C-2) with a calculated burnup of 57,000 MWd/Te showed approximately 7% $\Delta V/V$ fuel expansion. Because of the difficulty in measuring the fuel radiographs and until more accurate values can be obtained during destructive examination of the fuel pins, it is considered that the fuel in both pins is showing approximately the same fuel expansion rate for the same burnup.

The fuel in pin B8C-2 shows a center void 0.10-inch diameter at the top of the fuel column and approximately 0.075-inch towards the bottom of the fuel column (initial pellet diameter 0.204 in.). The center void size is greater than can be accounted for by the initial porosity present in the fuel pellets and it therefore appears that part of the void space present in the fuel cladding gap has been transferred to the center void.

Capsule B8D was also removed from the GETR for interim neutron radiography. Similar

to the fuel pins in capsule B8C, B8D fuel pins (three sodium bonded) are showing expansion of the low density portion of the fuel column into the plenum regions of the fuel pins (low density fuel axially unrestrained).

The greatest expansion is observed for the fuel containing 40% PuO_2 by weight with 1.97 stoichiometry. The next greatest expansion rate is observed for fuel containing 25% PuO_2 by weight and 1.97 O/M stoichiometry. The least expansion is observed for fuel containing 25% PuO_2 by weight and 1.935 O/M stoichiometry. The observed % ΔL 's are 26.2%, 17.1%, and 14.5%, respectively. The data appear to indicate increasing chemical reactivity between the fuel and sodium with increasing Pu content of the fuel and, in agreement with out-of-pile tests, with increasing stoichiometry.

Capsule B8D was returned to the GETR for further irradiation. The accumulated burnup on fuel pins B8D 1, 2, and 3 are 38,000, 40,500, and 38,500 MWd/Te, respectively.

SECTION IV

TASK C—SAFETY EXPERIMENTATION

4.1 GENERAL

The purpose of Task C is to study fuel and cladding behavior under various casualty and accident situations, using both analytical techniques and an experimental program of testing the response of fuel to severe transients. These investigations include a study of the performance of fuel and cladding under transient conditions as a function of axial restraint, density, performance of fuel in the

presence of a defect, and comparison of the behavior of power and pelleted fuel. Additional tests are conducted to determine the potential for molten fuel movement and to observe the effect of the prototypical range of fuel and blanket lengths on movement. Testing includes experiments which involve multiple-pin assemblies to study flow blockage conditions and experiments to evaluate the work-energy relationships and energy partition data needed in the analysis of large, terminal accidents.

During the current contract period, it is expected that work will progress through issuance of a final report on transient irradiation of zero-burnup, axially-restrained fuel as a function of density and fuel form (pellet and powder). Experiments will be conducted on pre-irradiated long pins to study relocation of molten fuel during transients and tests of defected or sodium-bonded fuel pins will be initiated. Out-of-pile scoping studies on flow blockage experiments will be performed, and conceptual design of work-energy experiments will continue.

4.2 AXIALLY-RESTRAINED FUEL EXPERIMENTS

4.2.1 Zero-Burnup Fuel Specimens

Preparation of a report has been initiated, covering the completed transient irradiation of the six zero-burnup fuel specimens (C4A to D, I, and J). This initial report will be primarily a quantitative comparison of expected and actual results from these irradiation experiments.

4.2.2 EBR-II Irradiations

EBR-II irradiation has continued for fuel specimens C4G and C4H (EBR-II capsules F7C and F7D). The accumulated burnup for these two specimens stands at approximately 56,600 MWd/Te (slightly exceeding the expected target burnup of 50,000 MWd/Te) as of January 16, 1969. These two specimens will be removed from EBR-II S/A XO15 and returned to Vallecitos Nuclear Laboratory to start non-destructive examination during the next quarter. Four additional specimens (C4E, F, K and L) are awaiting insertion.

4.2.3 TREAT Capsule Design for Preirradiated Fuel Specimens

Since it is planned that some of the pre-irradiated fuel specimens (C4E to H, K, and L) will fail during their transient irradiations, it

is planned to significantly alter the TREAT capsule design previously used to irradiate the zero-burnup fuel specimens (C4A - D, I, and J) in an attempt to study the following failure-related phenomena:

- The ability of molten fuel, expelled from the fuel specimen as a result of failure, to cause sodium voiding of a simulated fuel pin array.
- The ability of this same expelled molten fuel to directly cause failure propagation by melting through the cladding of adjacent (in this case, dummy) fuel pins.

A schematic of the inner capsule assembly of this proposed capsule is shown in Figure 4-1. This figure presents the general arrangement of the inner capsule with the preirradiated fuel specimen as the central pin in a seven-pin array. The six outer pins are dummy fuel pins containing depleted UO_2 pellets. They are located on a 1.20 P/D array with respect to the actual test specimen.

Figure 4-2 presents the planned radial dimensions of the complete TREAT assembly at a typical elevation in the fueled region of the test specimen. The design has been developed to fit within the dimensions of the Series 6 pressure vessel (see 28th quarterly), represented as the outer pressure vessel wall in Figure 4-2.

The overall longitudinal arrangement of the inner capsule assembly, as it would appear in the Series 6 pressure vessel, is shown in Figure 4-3.

4.2.3.1 Design Features of the Inner Capsule Assembly

a. Geometry:

The preirradiated fuel specimen is centered in a dummy fuel pin array which simulates typical reactor spacing (1.20 P/D ratio).

FIGURE 4-1. CONCEPTUAL SKETCH, SERIES IV, TRANSIENT CAPSULE

FIGURE 4-2. CONCEPTUAL DESIGN OF TREAT CAPSULE FOR SERIES IV PRE-IRRADIATED SPECIMENS

The six surrounding dummy fuel rods (0.250-inch diameter by 0.015-inch wall thickness, 316 stainless steel cladding) contain depleted UO_2 pellets to passively simulate the thermal resistance of adjacent fuel pins.

When failure occurs in the test specimen during transient irradiation, it is hoped that this dummy array will yield quantitative data on the kinetics of molten fuel penetration of cladding on adjacent fuel pins: i.e., a measure of the ability of expelled molten fuel to directly cause failure propagation.

b. Pressure Measurement:

A pressure transducer is planned at the bottom end-closure of the inner capsule. The purpose of the transducer is to monitor the pressure history of the coolant during the transient irradiation, which is of particular interest in the event of test specimen failure. In such a case, the pressure sensed should indicate the severity of the molten-fuel-sodium reaction (e.g., a determination of whether such a reaction is dominated by shock phenomena, or by a slower pressure buildup, reflecting a quasi-steady sodium boiling phenomenon).

In addition, a measure of peak pressure will be provided by short, hollow, closed, stainless-steel cylinders of individually varying wall thickness, to be mounted on top of the dummy fuel pins (Figure 4-3).

c. Flow Instrumentation

An electromagnetic flowmeter is located at the upper end of the heat sink (Figures 4-1 and 4-3). Again, as with the pressure transducer, the flowmeter is primarily intended to monitor events once test specimen failure has occurred. In this case, the flowmeter will monitor the rate of

sodium coolant expulsion from the dummy array, following failure. This should provide a quantitative indication of the rate and extent of sodium voiding resulting from a molten-fuel-sodium reaction.

Since this capsule design contains stagnant sodium, the possible significance of such a measurement would be: (a) If the maximum coolant velocity is on the order of 1 ft/sec, then expulsion would probably be over-ridden by normal coolant flow in an actual LMFBR core; or (b) if the maximum velocity is on the order of 10 ft/sec, the expulsion of coolant could seriously compete with and possibly cause a reversal of the normal coolant flow in the core.

The combined output of pressure transducer and the flowmeter is anticipated to define the failure-induced hydraulics of the capsule, leading to possible experimental verification of the assumed process of voiding.

d. Heat Sink:

The heat sink is to be made of pure nickel or of a high nickel content inconel stock. This is necessary to reduce the effective TREAT power factor of these highly enriched EBR-II pre-irradiated fuel specimens to an acceptable level.

The lower portion of the heat sink, encompassing all of the fuel region of the test specimen, is split (with an overlapping tongue-in-groove type of joint which precludes any possibility of contact between the inner capsule wall and molten fuel expelled during a failure, even if the split sections separate to their maximum). These split sections will allow easier remote hot cell disassembly of the inner capsule and, as a result, a better (undisturbed) visualization of any failure-induced molten fuel movement in the dummy array.

e. Coolant Temperature Measurement:

Eleven chromel-alumel thermocouples will be used to monitor the sodium temperature history in the region of the fuel.

There will be three sets of three thermocouples each, located radially (Figure 4-2) with axial locations corresponding to the approximate quarter points of the fuel column length (Figure 4-3). Thermocouples are also located at the upper end of the heat sink, adjacent to the test specimen, to monitor the temperature of the sodium passing the flowmeter, in the event of sodium expulsion.

4.2.3.2 Capsule Assembly

Because of the necessity of remote encapsulation of the irradiated test specimen, followed by a remote sodium fill, the following features have been included in the conceptual design:

- All instrumentation lead penetrations of the inner capsule containment are completed at open assembly shop level, prior to the inner capsule being inserted into the hot cell area (this includes the pressure transducer which is attached to the bottom end closure).
- The heat sink-flowmeter assembly, containing the six dummy fuel pins (maintained in position by upper and lower spiders) is also assembled and mounted inside the inner capsule at open assembly shop level.
- The six dummy fuel pins are maintained in place by the above mentioned spiders and circumferential springs (Figure 4-3). The design of the spiders and the spring arrangement allows the bare test specimen to be remotely inserted into the already assembled dummy array from the bottom. The spring system allows the dummy array

to spread, and adjust, for passage of the test specimens end cap welds or permanent radial cladding deformation of the test specimen, or both.

- Once the specimen is inserted into the dummy array, the bottom end closure is placed in position and welded (the only remote weld required in the entire assembly sequence).
- The outer envelope of the inner capsule assembly must be compatible with the remote sodium-fill system built for these re-encapsulations (Figure 4-4). This system allows remote hot-cell operation of all the sodium coolant flushing and filling operations to be performed on the complete inner capsule assembly.

Two sodium vent-and-level lines penetrate the upper end of the inner capsule. One of the two lines ends flush with the bottom side of the upper end closure, allowing total filling of the inner capsule with sodium. This line will allow flush-cleaning of the entire inner capsule. The second line extends through the upper end closure and down nearly to the upper end of the flowmeter. By applying back pressure to the first line and venting through this second line, the proper sodium level can be established.

4.3 AXIAL FUEL RELOCATION EXPERIMENTS

The initial phase of the destructive examination of Capsule C5B was completed. Further metallographic examination has been deferred until mid-1969.

The fuel pin was de-encapsulated and about 33 cc (at STP) of fission gas collected. This corresponds to a gas release (steady state and transient) of ~38%, assuming a steady state burnup of 20,000 MWd/Te.

Profilometry results indicate cladding deformation of ~2.5 to 4 mils with sharp peaks (~3 to 5.5 mils) at the approximate fuel-blanket interface locations.

The pin was sectioned for detailed examination. The specimens selected for initial study are shown in Figures 4-5 through 4-7.

Fuel movement is very evident (Figure 4-5) from the porous structure of the once-molten central core. The molten fuel volume is estimated from these transverse specimens to be ~35%. Figure 4-6, b through e, shows the annular blanket with once-molten fuel in the center, confirming the earlier observation of fuel movement into the upper blanket to a height of ~10 inches above the fuel-blanket interface. It is significant that this pin did not fail, whereas C5A, which was identical in all respects to C5B except that the blanket was solid rather than annular pellets (0.070-in. diameter hole in C5B blanket), did experience gross failure (See 28th quarterly). Apparently, the pressure relief path provided by the annular blanket was the prime factor in preventing cladding failure. A distinct, central core is evident in Figure 4-6d, indicating the possibility of secondary flow, either during initial expansion or subsequent cooling and solidification.

Figure 4-7 and 4-8 verify that the high local cladding deformation (~5.5 mils) was a result of molten fuel forcing a separation between the blanket and the uppermost fuel pellet (see 28th quarterly). Figure 4-8a shows the cladding in the area of peak deformation. There is no indication of overheating due to contact or proximity of the molten fuel. This occurrence (molten fuel contact with no cladding damage other than local deformation) has been detected in previous Task C experiments, C3B and C4A.

The bright spots dispersed through the fuel (Figures 4-7 and 4-8b) are believed to be caused by metallic fission products which have

coalesced as a result of melting of the fuel. This has also been seen in steady-state irradiation experiments (Task E, F) but, the "spherical" shape of the fission products is apparently related only to fuel melt conditions.

4.4 DEFECTED OR SODIUM-BONDED FUEL EXPERIMENTS

The high pressure sodium fill system, after completing functional checkouts, was placed in temporary storage. Installation, checkout, and sodium fill of the C6 fuel pins have been delayed to coincide with the capsule assembly schedule (one capsule to be completed by mid-June 1969).

The fuel pins have been completed and also placed in storage to await installation of the fill system. Fabrication of component parts for the C6 capsule assemblies has continued.

4.5 FLOW BLOCKAGE EXPERIMENTS

Components fabrication for Mockup 1A of the forced convection capsule has been completed and the pressure transducer has been delivered. The flowmeter was also delivered but had to be returned for repair of a leaky weld. The leak was found and repaired and the flowmeter is being stored awaiting final assembly of the mockup.

Fabrication of eight of the "backup design" fuel simulators (electrical resistance heaters) has been completed. (Flaws in the "primary" heaters, reported in the 28th quarterly, greatly reduced the probability of their successful operation.) One of the backup heaters was rejected because of a hot spot on the cladding observed during low power tests. The seven "good" heaters should permit several tests in the 3-pin mockup.

The test performed on these heaters include resistance measurements and x-rays in addition to low power-operation. The low

FIGURE 4-5. C5B TRANSVERSE MACRO SPECIMENS WITH NEUTRON RADIOGRAPH SHOWING LOCATIONS

FIGURE 4-6. C5B TRANSVERSE MACRO SPECIMENS WITH NEUTRON RADIOGRAPH SHOWING LOCATIONS

FIGURE 4-7. BETA-GAMMA AUTORADIOGRAPH (TOP) AND MACROGRAPH (BOTTOM) OF FUEL/BLANKET INTERFACE SPECIMEN

100X, ETCHED 10% OXALIC

A191-15 THROUGH 20

(a)

15X, AS POLISHED

(b)

A191-02

APPROXIMATE
MOLTEN FUEL
BOUNDARIES

FIGURE 4-8. MICROGRAPHS OF LONGITUDINAL FUEL/BLANKET INTERFACE SPECIMEN

power tests were performed in air with temperature sensitive paint (set to change color at 554°F) sprayed on the cladding. Power was applied and adverse temperature effects or distributions noted. Five heaters underwent a uniform change in color while two gave an indication that one end was slightly cooler than the other. The final heater was rejected

because of a hot spot which was interpreted as a defect in the cladding.

Operation of Mockup 1A has been temporarily postponed because of funding limitations. This test, however, is not on the critical path for development of the multi-pin capsule.

SECTION V

TASK E—FUEL PERFORMANCE EVALUATION

5.1 GENERAL

The purpose of Task E is to develop technology required to specify the fuel materials, fuel performance, and quantitative design limits for fast reactor oxide fuels. A typical target performance is to operate at linear powers up to 18.5 kW/ft (600 W/cm) and to peak burnups greater than 130,000 MWd/Te (35×10^{20} fission/cc). The program includes determination of physical and mechanical properties of fuel using both in-pile and out-of-pile testing. Emphasis is on determining the properties of the fuel, and its in-pile behavior at power levels typical of near-term FCR cores.

5.2 CENTER TEMPERATURE LIMITS

5.2.1 Capsule E1AA (Direct Measurement of In-Pile Thermal Conductivity)

Capsule E1AA test was terminated at about 18,000 MWd/Te following the failure of the tungsten-rhenium center fuel thermocouple. During operation, mixed oxide thermal conductivity data were obtained over the temperature range 1200°C to 2100°C. The results are being analyzed to determine values of effective thermal conductivity and gap coefficient.

The capsule is undergoing destructive examination. Disassembly of the capsule confirmed that it was intact. Visual examination showed that all thermocouples were in intimate contact with their respective claddings. A profilometer trace showed no distinguishable swelling of the fuel pin. The fuel pin has been sectioned and is awaiting metallographic examination. Macrographs of the cut sections showed that the center thermocouples were in place.

5.2.2 Capsule Series E1J (Effect of Restructuring on Effective Thermal Conductivity)

The first E1J capsule with O/M = 2.00 has successfully completed irradiation in the GETR trail cable facility. Operation was for a total of 88 hours in which the following sequence was used to develop a center hole with an equilibrium microstructure prior to melting:

- Initial power at 7.6 kW/ft
- Power increased to 17.9 kW/ft in steps of 2 kW/ft in 2-hour intervals
- Steady operation at 17.9 kW/ft for 8 hours

- Capsule cycled between 18 kW/ft and 10 kW/ft three times in increments of 4 kW/ft
- Capsule operated 24 hours at 18 kW/ft
- Capsule operated 24 hours at 24 kW/ft, center of fuel in molten state.

The capsule is presently undergoing destructive examination.

During irradiation all thermocouples but one functioned properly. No problems were encountered in obtaining the desired operating powers, and thermocouple indicated powers were always within 12% to 14% of those predicted by GETR physics.

Additional E1J irradiations are scheduled to resume in mid-February 1969, assuming that examination of the first E1J capsule is completed on schedule.

5.3 HIGH BURNUP IRRADIATIONS

5.3.1 Capsule E2J (High Burnup Performance of Pellet and Annular Fuel, 89% Smeared Density)

The evaluation of capsule E2J indicates equivalent performance for solid and annular fuel of similar smeared density when irradiated to 125,000 MWd/Te at a linear power of 22 kW/ft.

Average fuel burnup values based on the Neodymium to Heavy Element ratio indicate 125,000 MWd/Te for the pellet fuel and 124,000 MWd/Te for the annular fuel. Fission gas releases were 76% for the solid pellets and 80% for the annular fuel.

Examination of a traverse metallographic section from each fuel pin revealed fuel morphology similar to previous high burnup mixed oxide fuel irradiated under Task E (Figure 5-1). Average central void diameters were 0.025 inch in the pellet fuel and 0.049 inch in the annular

fuel. Since the annular fuel contained a 0.060 inch diameter hole when fabricated, its partial closure may be an indication of fuel swelling into the central area.

No significant reaction occurred in the welded and drawn 316 cladding at temperature of 1100°F.

5.3.2 Capsules E2P-Q (Swelling Under Varied Radial Restraint)

Re-encapsulation of the E2P-Q fuel pins was successfully completed and the capsule was inserted in GETR on November 25, 1968. Power calculations from capsule thermocouples and GETR physics are in agreement within 7%. The NaK convection problem which distorted thermocouple readings during the earlier irradiation has been solved by redesigning the capsule thermal barriers to allow minimal communication between the NaK annuli.

E2P-Q will be removed for an interim examination in early February 1969.

5.4 PHYSICAL PROPERTY STUDIES

5.4.1 Mechanical Properties of Ceramic Fuels

The creep and plasticity study of UO_2 has been completed and a topical report is in preparation. Work has been initiated on a similar study of plutonium-uranium oxides. The test apparatus is currently being installed in a glove-box and test samples are being prepared. Initial tests will study the effects of stoichiometry and density on creep rates, testing samples with the following parameters:

<u>Stoichiometry</u>	<u>% Theoretical Density</u>
$(Pu_{0.2}U_{0.8})O_{1.93}$	96
$(Pu_{0.2}U_{0.8})O_{1.97}$	96
$(Pu_{0.2}U_{0.8})O_{2.00}$	96
$(Pu_{0.2}U_{0.8})O_{1.97}$	93
$(Pu_{0.2}U_{0.8})O_{1.97}$	90

E2J-1
SOLID PELLETS

(Initial Center Void Diameter, 0.000 in.)
(Final Center Void Diameter, 0.035 in.)

E2J-2
ANNULAR PELLETS

(Initial Center Void Diameter, 0.060 in.)
(Final Center Void Diameter, 0.049 in.)

FIGURE 5-1. CAPSULE E2J FUEL (125,000 MWd/Te at 22 kW/ft)

Test temperatures will vary from 1200°C to 1600°C with compressive loads from 1000 to 10,000 psi. Additional samples will also be prepared to study the effects of plutonium content and grain size on the creep properties.

5.4.2 High Temperature Phase Studies

Preliminary analysis of diffraction patterns obtained from 1.92, 1.96 and 2.00 O/M mixed oxide samples indicated the material to be multiphase.⁽²⁾ Further investigation has shown that the extra phases are surface phenomena and that the bulk of the material is single phase.

5.4.3 Electron Microprobe Examination of a Fuel Sample from Capsule E5B

An electron microprobe examination has been completed on a fuel sample from high burnup capsule E5B.⁽³⁾ The microprobe sample was from the pellet fuel pin which has a burnup of 135,000 MWd/Te at a linear power of 22 kW/ft. The cladding i.d. temperature has been calculated to be 1320°F during the 8400 hours exposure. Significant interaction occurred between the stoichiometric fuel and 347 stainless steel cladding. Cladding material 3 to 4 mils thick was separated from the base metal (Figure 5-2).

5.4.3.1 Fuel-Cladding Reactions

A qualitative determination of the distribution of cladding elements and fission products in the fuel-cladding interaction zone was made. Figure 5-3 is a 500× micrograph of the area examined and shows a reaction zone which is 4 to 5 mils thick. A specimen current image of this area is displayed in Figure 5-4a. K α x-ray pulse images for the primary cladding constituents, Fe, Cr, and Ni, are displayed in Figures 5-4b, 5-5a, and 5-5b respectively. These are mirror images of Figure 5-3. A bright area on the K α x-ray pulse image display is indicative of the presence of the specific constituent. Stratification of these elements is also evident. A quantitative determination of the relative Fe, Cr, and Ni concentrations was made at intervals

across the reaction zone. The composition at each point going from the fuel side to the cladding side is given in Table 5-1. Other elements must also be present since the total of these values at each point is low, much less than 100%. Pd, Mo, Cs, and Ba were detected, with the highest concentrations being adjacent to the unreacted cladding. None of these fission products were identified in the cladding. Fuel was found only in the grey inclusions seen in Figure 5-3, some of which are marked by arrows. These fuel inclusions appear as black areas in the specimen current photograph (Figure 5-4a) and in the K α x-ray pulse image.

5.4.3.2 Fuel Examination

Metallic inclusions were analyzed at three points along the fuel radius. Mo and Ru were present in almost equal amounts as the predominant fission products. The only other elements detected were trace amounts of Tc and Rh. Points in the fuel matrix adjacent to these inclusions were also analyzed. The only fission product detected was a trace of Zr.

A grey phase in the fuel matrix was also examined. Zr, Ba, Cs, and Sr were found. It was determined that the spatial distribution of Ba and Zr were identical. The Cs-rich areas were in close proximity with, but not identical to, the Ba-Zr rich areas. The Sr was barely detectable and the exact distribution could not be determined. It is plausible that the second phase is barium zirconate.

5.5 INTEGRAL THERMAL FLUX IRRADIATIONS

5.5.1 Capsules E5C, E5D, E5E (High Power, High Burnup Tests of Long Fuel Columns with Solid and Annular Pellets and Powder Fuel)

Irradiation of capsules E5C, E5D, and E5E continued in GETR. Burnup through January 31, 1969 is approximately 120,000 MWd/Te.

FIGURE 5-2. CAPSULE E5B FUEL (135,000 MWd/Te at 22 kW/ft)

MIXED OXIDE
FUEL

REACTION
ZONE
0.004 in.-0.005 in.

347 STAINLESS
STEEL CLADDING

500X

VOID

NOTE: FUEL PARTICLES INDICATED BY ARROWS

FIGURE 5-3. MICROGRAPH FROM CAPSULE E5B SHOWING FUEL-CLADDING INTERACTION ZONE

NOTE: FUEL PARTICLES INDICATED BY ARROWS

(a) SPECIMEN CURRENT IMAGE $\sim 425 \times$

(b) Fe K α X-RAY PULSE IMAGE (BRIGHT REGIONS INDICATE PRESENCE OF Fe)

FIGURE 5-4. MICROPROBE SPECIMEN CURRENT AND X-RAY PULSE IMAGES OF FUEL-CLADDING INTERACTION REGION OF FIGURE 5-3

(a) $\text{Cr } K\alpha$ X-RAY PULSE IMAGE (BRIGHT REGIONS INDICATE PRESENCE OF Cr)

(b) $\text{Ni } K\alpha$ X-RAY PULSE IMAGE (BRIGHT REGIONS INDICATE PRESENCE OF Ni)

FIGURE 5-5. MICROPROBE X-RAY IMAGES OF ALLOYING ELEMENTS WITHIN FUEL-CLADDING INTERACTION REGION OF FIGURE 5-4.

TABLE 5-1

Fe, Cr, AND Ni CONCENTRATIONS AT SELECTED POINTS
IN THE CLADDING-FUEL REACTION ZONE; CAPSULE E5B

<u>Point</u>	<u>% Fe</u>	<u>% Cr</u>	<u>% Ni</u>	<u>Total</u>
1 (Near Fuel)	58.5	15.5	5.3	79.3
2	30.8	11.0	2.0	43.8
3	30.5	9.0	2.4	41.9
4	37.5	12.0	2.8	52.3
5	52.3	3.0	4.6	59.9
6	37.3	19.0	3.0	59.3
7 (Near Cladding)	37.0	17.9	2.6	57.5
Cladding Matrix	67.7	17.4	8.9	94.0

5.5.2 Capsule E5F (Linear Fuel Movement)

The purpose of capsule E5F is to measure the axial change in length of mixed oxide in-pile as a function of power and burnup. Changes in the 10-inch fuel column length are monitored via a transducer [capsule and fuel pin drawings were presented in the Twenty-Fifth Quarterly Report⁽⁴⁾].

After 44,000 MWd/Te, the E5F capsule was removed from the reactor, neutrographed to check the condition of the fuel and of the linear displacement transducer and reinserted. Upon startup the transducer instrumentation was observed to behave anomalously and an investigation of the transducer electrical circuit revealed an open circuit in the transducer. Since the transducer could no longer be used to monitor fuel movements and since this was the primary purpose of the E5F capsule, the capsule was removed from the GETR.

Neutrography consisted of four photographs, two of the fuel region at 0° and 90° orientations and two of the transducer region at 0° and 90° orientations. The exposures of the fuel section revealed that the fuel and insulator pellets were in excellent condition, with no apparent reaction between them. Axial gaps which were present in the fuel column before irradiation no longer existed. A central void had developed in the fuel and could be detected readily in the neutrographs; however, precise physical measurements of the void size were impossible because of the diffuse nature of the neutrographs in this particular area.

A comparison of measurements made from the neutrographs of the pre- and post-irradiated fuel and the transducer probe position shows good agreement on the amount of fuel swelling which has occurred in the axial direction. The slight discrepancy which does exist is most likely due to the inability to make accurate measurements from neutrographs.

SECTION VI

TASK F—INTEGRAL FAST FLUX

6.1 GENERAL

This task factors the more promising results of the basic fuel technology developments into the design, fabrication, and integral testing of fuel elements in fast fluxes. Important fuel performance characteristics, such as fission gas release, fuel cladding interactions, and fuel thermal performance, are determined under conditions simulating, as closely as possible, those expected in a fast ceramic reactor. Testing of fuel pins in a fast flux, in principle, will permit testing of fuel bundles under fair approximation of prototypical conditions although this has not as yet been possible in EBR-II.

The test parameters for Task F are selected by a combination of design analysis (which relies on basic properties data) and on the results of idealized, differential experiments (such as those in Task E and Task L). In addition, the results of prior integral tests are interpreted as quantitatively as possible to guide the selection of future tests. The selection of test parameters is also influenced and guided by the work under Task C, which focuses on the preferred range for safety considerations of LMFBR designs. 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 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.

6.2 MEDIUM POWER, LONG TERM IRRADIATIONS

6.2.1 Group 2 (Medium Power Scoping Irradiations)

Metallographic examination of fuel pins F2C, F2H, F2R, and F2T was completed. These pins were irradiated to peak burnups of 74,000 MWd/Te in EBR-II subassembly XG05. Destructive examination of a failed pin (F2O) from the same subassembly was initiated. The remaining four capsules irradiated in the XG05 subassembly are being retained at the EBR-II site for additional irradiation to their target burnup of 100,000 MWd/Te.

Photomicrographs of transverse cross-sections from pins F2C, F2H, F2R and F2T are shown in Figures 6-1 through 6-4.

No degradation of the F2T and F2R cladding (type 347 stainless steel) was observed. A single region of grain boundary attack was observed on the i. d. of Incoloy-800 clad pin F2C. Several intergranular cracks were observed around the i. d. of the type 316 stainless steel clad pin F2H. The cracks were observed in a section of the pin approximately 5-1/2 inches from the bottom (see Figure 6-3). No cracks were observed in a second section 12-3/4 inches from the bottom of the pin. No abnormal operation was indicated by the structure of the fuel in the sections examined from the four pins. Fission products were observed throughout the fuel of each pin. Large concentrations of fission products (white and grey

FIGURE 6-1. TRANSVERSE CROSS-SECTION FROM FUEL PIN F2C

FIGURE 6-2. TRANSVERSE CROSS-SECTION FROM FUEL PIN F2R

HIGH MAGNIFICATION
PHOTOMICROGRAPH SHOWS
SECOND PHASE MATERIAL
OBSERVED NEAR FUEL
CLADDING INTERFACE

FIGURE 6-3. TRANSVERSE CROSS SECTIONS FROM FUEL PIN F2H

FIGURE 6-4. FISSION PRODUCTS TYPICALLY OBSERVED IN CRACKS NEAR FUEL PERIPHERY (FUEL PIN F2T)

regions in micrographs in Figures 6-3 and 6-4), similar to those observed in the Group 2 pins examined at 50,000 MWd/Te, were observed at the fuel-cladding interface and in cracks near the fuel periphery. Previous microprobe examinations of high burnup specimens showed that the grey appearing phases contain zirconium, barium, cesium, strontium, and silicon;

the white appearing phases contain molybdenum, ruthenium, technetium, rhodium, and palladium.

The calculated percent fission gas released from each pin was based on maximum burnup values reported by EBR-II. The calculated values are listed in Table 6-1.

TABLE 6-1

PERCENT GAS RELEASE FROM GROUP 2 FUEL PINS

<u>Pin</u>	<u>Average Burnup</u>	<u>Gas Generated (cc)</u>	<u>Gas Released (cc)</u>	<u>% Release</u>
F2C	63,100 MWd/Te	130	83.2	64.0
F2H	65,800 MWd/Te	136	91.8	67.5
F2R	63,100 MWd/Te	131	79.5	60.8
F2T	65,800 MWd/Te	136	80.7	59.2

The non-destructive examination of the failed pin, F2O, in the Group 2 series was completed and the destructive examination initiated. Neutron radiographs, profilometer traces, and the gamma scan of the capsule were presented previously. ⁽⁵⁾

The F2O pin was removed from the capsule nearly intact. A picture of the pin is shown in Figure 6-5. A longitudinal rupture extended over most of the fuel column. Four overheated regions were observed along the cladding approximately 1-1/2 inches apart. The first overheated region occurred about three-quarters of an inch from the bottom of the fuel column. Each region was located on the lower side of the wire wrap. It is postulated that these regions resulted from a perturbation in heat transfer caused by fission gas bubbles temporarily pinned below the wire wrap.

Micrometer readings taken along the length of the pin indicated diametral increases of up to 0.056 inch*. The diametral increase is attributed to the volume expansion resulting from the fuel-sodium reaction. An indication of the extent of the fuel-sodium reaction is provided by the measured increase in the fuel length. A 1.3% increase occurred, three times greater than that measured for the unfailed Group 2 fuel pins. It is estimated that the total fuel volume, including that ejected through ruptured regions of the cladding into the annulus between the pin and capsule, was 11% greater than that of unfailed pin F2H, excluding any changes in density.

* Measurements represent unrestrained expansion of the fuel after rupture, not cladding strain.

FIGURE 6-5a. THE TOP HALF OF THE CAPSULE WAS REMOVED BY LONGITUDINALLY SLITTING ALONG BOTH SIDES. THE F20 FUEL PIN IS SHOWN IN THE BOTTOM HALF OF CAPSULE.

FIGURE 6-5b. THIS FIGURE SHOWS PHOTOGRAPHS OF THE F20 PIN TAKEN AT SEVERAL ANGULAR POSITIONS AROUND THE PIN. THE PHOTOGRAPHS ARE SHOWN ADJACENT TO ONE ANOTHER TO ILLUSTRATE HOW THE RUPTURE FOLLOWED THE WIRE WRAP. THE FUEL COLUMN ORIGINATES $3 \frac{1}{4}$ INCHES FROM THE BOTTOM OF THE PICTURE. THE OVERHEATED REGIONS OF THE CLADDING AT 5, $6 \frac{1}{2}$, 8, AND 11 INCHES FROM THE BOTTOM OF THE PIN ARE POSTULATED TO HAVE RESULTED FROM A PERTURBATION IN HEAT TRANSFER CAUSED BY FISSION GAS BUBBLES TEMPORARILY PINNED BELOW THE WIRE WRAP.

FIGURE 6-5. POST-IRRADIATION EXAMINATION OF FUEL PIN F20

Percent fission gas release from the F2O pin was calculated to be 32%. This is about one-half the gas released from pins F2C, F2H, F2R, and F2T.

It is hypothesized that the low gas release indicated the pin failed at a low burnup (in the range from 0 to 30,000 MWd/Te); the low gas release would result from a lowering of the bulk fuel temperature by the ingress of sodium after the failure. Sectioning of the fuel pin and preparation of metallographic samples is presently underway.

6.3 LOW POWER FUEL SWELLING IRRADIATIONS

6.3.1 Group 0 (Low Power Swelling of High Density Fuel)

Capsule F0E was removed from EBR-II subassembly XG02 after achieving its target burnup of 40,000 MWd/Te and returned to Vallecitos for destructive examination. This is the first capsule from the Group 0 series to complete its irradiation. The remaining four capsules in this series are still in EBR-II and have reached burnups of 49,000 MWd/Te.

6.3.2 Group 6 (Fuel Cladding Interactions Under Accelerated Swelling)

Irradiation of the four Group 6 capsules in EBR-II subassembly XO10 continued. Peak burnup on the fuel in the capsules at the end of EBR-II run 32 was 50,500 MWd/Te.

6.3.3 Group 8A (Effect of Density on Fuel Swelling Rate)

Irradiation of the sixteen Group 8A capsules in the EBR-II was continued. Burnup on the capsules in subassembly XO19 is 33,000 MWd/Te; those in subassembly XO20 have reached 34,000 MWd/Te burnup.

6.3.4 Group 8B (Effect of Void Deployment on Fuel-Cladding Interaction)

The nineteen Group 8B capsules continued their irradiation in EBR-II subassembly XO36. Current burnup is 11,800 MWd/Te.

6.4 HIGH POWER INTERMEDIATE BURNUP IRRADIATIONS

6.4.1 Group 3B (Density and Stoichiometry Effects)

The eighteen Group 3B capsules continued to be irradiated in EBR-II subassembly XO27. Peak burnup on the capsules is 40,000 MWd/Te. A request to continue the irradiation of these capsules to 80,000 MWd/Te without an interim examination at 40,000 MWd/Te was submitted to ANL.

6.4.2 Group 4 (Low Density Fuel Behavior at High Heat Ratings)

Irradiation of the seven Group 4 capsules in EBR-II subassembly XO39 was terminated on December 24, 1968. Peak burnup on the fuel in these capsules was calculated to be 54,000 MWd/Te. An interim examination consisting of neutronradiography, measurement of diameters along each capsule and the length of each capsule, and measurement of the weight of each capsule is being carried out at EBR-II. Reinsertion of these capsules for additional irradiation was requested.

6.5 ELEVATED CLADDING TEMPERATURE TESTS

6.5.1 Group 5 (Fuel Performance at Peak FCR Cladding Temperatures)

Fabrication of the remaining three fuel pins of the Group 5 nineteen-capsule subassembly was continued. A design study was completed

on equipment required to pressure test capsule bottom end welds as specified by EBR-II. The equipment was fabricated, checked out, and is now operational.

6.5.2 Group 10B (Cladding Performance at Peak FCR Cladding Temperatures)

Fabrication specifications for the sixteen Group 10B fuel pins were issued and preparation of fuel pin and capsule drawings initiated.

A meeting between General Electric and Argonne National Laboratory Reactor Engineering personnel was held at ANL to establish a subassembly design for the Group 10 irradiations. An ANL-RE design incorporating an insulated sodium by-pass flow was agreed upon. This subassembly design will make possible capsule irradiations with a peak temperature of 1250° F on the outside surface of the cladding.

6.6. UNENCAPSULATED FUEL PIN IRRADIATIONS

6.6.1 Group F3A (Unencapsulated Fuel Pin Irradiation)

Irradiation of the sixteen 0.290-inch diameter Group 3A fuel pins continued in EBR-II subassembly XO40. The pins have reached peak burnups of 20,900 MWd/Te.

6.6.2 Group 9A (High Power, High Burnup, Unencapsulated Fuel Pin Irradiations)

The thirty-seven Group 9A fuel pins are at EBR-II awaiting insertion into the reactor. Final approval for irradiation was granted and the pins were loaded into EBR-II subassembly XO43.

6.6.3 Group 9B (Medium Power, High Burnup, Unencapsulated Fuel Pin Irradiation)

Fabrication of the thirty-seven Group 9B fuel pins was continued. Thirty fuel pins were completed and readied for shipment to the EBR-II site.

Preparation of the hazards analysis for the group 9B irradiation was initiated.

6.6.4 Group 9C (High Power, High Burnup, Unencapsulated Fuel Pin Irradiation)

Fabrication of the thirty-seven Group 9C fuel pins was completed and shipment to the EBR-II site began.

The hazards analysis for the Group 9C irradiation was completed and submitted to ANL for approval.

6.6.5 Group 9D (Medium Power, High Burnup, Unencapsulated Fuel Pin Irradiation)

The thirty-seven Group 9D fuel pins were completed and readied for shipment to the EBR-II site. Each of the Group 9D pins was tagged with xenon tracer gas provided by ANL. The xenon tagging procedure was qualified by making five trial runs using UO₂ filled pins identical to the Group 9D pins. Gas chromatography was used to insure that the tagging procedure was satisfactory. Approximately 1.0 milliliter of gas was inserted into each fuel pin.

Preparation of the hazards analysis for the group 9D irradiation was initiated and is nearly completed.

SECTION VII

TASK G—CORE NEUTRONICS, THERMAL
HYDRAULICS AND SAFETY ANALYSIS

7.1 GENERAL

The purpose of Task G is to provide guidance for effective planning and coordination of the experimental programs within P. A. 10, and to contribute to the basic analytical and design tools for the sodium-cooled fast breeder program. This requires integration of results of several disciplines, including core neutronics, thermal hydraulics, and safety and economic analysis. To accomplish this, improved analytical methods and computer programs are being developed and used in analytical studies of fast reactor neutronics, safety characteristics, and both analytical and experimental thermal hydraulics. These studies, together with a continual evaluation and surveillance of the results of experimental programs, form a basis for establishing current design and safety criteria. These criteria, in turn, define the range of reactor characteristics and test parameters upon which the experimental programs must focus.

During 1968, methods development and parametric analyses have been focused primarily on the areas of: (1) nuclear data and updating of group constants, with emphasis on processing of data for use in neutronics computations, (2) nuclear core analyses, including the evaluation of approximations used in neutronics computations, (3) transient and safety analysis, including accident studies and improvement of FORE II and sodium voiding models, and (4) thermal-hydraulic and fuel analysis, including fuel swelling model development and coolant mixing studies.

7.2 NUCLEAR DATA AND GROUP CONSTANTS

7.2.1 ENDF/B Evaluation and Testing

A computation of integral quantities for ZPR-III, Assembly 48, has been performed using the rapid cross section adjusting approach. These calculations were done in the manner specified by the Data Testing Subcommittee of CSEWG⁽⁶⁾ in order to permit a direct comparison between results using this approach and those from the more detailed computations performed at ANL using the ETOE-MC² approach.⁽⁷⁾ Whereas the ANL computations involved the generation of 26 group nuclear constants from ENDF/B data which are specifically for Assembly 48, the rapid cross section adjusting procedure consisted of (a) computing a generalized group constant set from ENDF/B which is applicable to many reactors but with less accuracy, using the ENDRUN code, and (b) generating a specific group constant set for Assembly 48 from the generalized set using the FCC-IV code.⁽⁸⁾ The final computations in both cases were performed in one-dimensional diffusion theory with identical problems.

Table 7-1 lists the calculated integral quantities along with the experimental values and the Argonne MC² results. Table 7-2 gives the computed fluxes and adjoint fluxes, normalized in the manner requested by the Data Testing Subcommittee.

As seen from Table 7-1, the agreement between ENDRUN and MC² results is very good,

TABLE 7-1

COMPARISON OF INTEGRAL QUANTITIES FOR ZPR-III
ASSEMBLY 48, AS MEASURED AND COMPUTED
WITH THE MC² AND THE ENDRUN CODES

Quantity	Experiment	ANL Analysis ETOE-MC ² MACH-1	GE Analysis ENDRUN-FCC-IV DEMON
Criticality	1.000	0.9730	0.9756
Ratio: <u>Fission</u> U-235 Fission			
U-238	0.0307	0.0304	0.0300
Pu-239	0.976	0.9193	0.9043
Pu-240	0.243	0.2256	0.2228
Pu-241	-	-	1.291
Pu-242	-	-	0.180
Ratio: <u>Capture</u> U-235 Fission			
U-235	-	0.2818	0.2805
U-238	0.138	0.1416	0.1408
Pu-239	-	0.1967	0.1913
Pu-240	-	0.2188	0.2181
Pu-241	-	-	0.2041
Pu-242	-	-	0.1665
B-10	-	1.314	1.314
Reactivity Coefficient (lh/gm-mole)			
Pu-239	106.4	134.60	132.5
Pu-240	19.5	20.71	20.45
Pu-241	-	202.00	241.8
Pu-242	-	17.43	17.9
U-235	79.5	102.21	103.4
U-238	-5.9	-7.91	-7.60
B-10	-89.3	-101.40	-97.08
Mo	-4.16	-9.80	-9.96
Fe	-0.69	-0.838	-0.876
Cr	-0.49	-1.108	-1.216
Ni	-1.07	-1.103	-1.334
Mn	-1.18	-2.263	-1.78
Al	-0.42	-0.420	-0.532
Na	-0.14	-0.067	-0.215
Mg	-	-0.569	-0.714
C	-0.05	-0.015	-0.203
O	-0.11	-0.100	-0.176

TABLE 7-2
COMPUTED FLUXES AND ADJOINT FLUXES
AT THE CENTER OF ASSEMBLY 48,
USING ENDRUN APPROACH

Group	ϕ	ϕ^+
1	0.47	1.48
2	1.62	1.24
3	3.84	1.21
4	5.39	1.12
5	7.04	1.01
6	9.97	1.01
7	11.72	1.00
8	11.11	0.99
9	9.84	0.98
10	9.09	0.96
11	7.22	0.95
12	6.06	0.94
13	5.17	0.95
14	3.74	0.97
15	2.59	1.00
16	1.50	1.05
17	0.73	1.10
18	1.29	1.13
19	0.81	1.19
20	0.42	1.35
21	0.23	1.40
22	0.106	1.54
23	0.038	1.46
24	0.012	1.64
25	0.003	1.37
26-29	0.0015	1.11

with the exception of the central worths of Pu-241 and the lighter scattering materials. The k_{eff} values agree to within 0.3% and all of the fission and capture ratios to within 2%. The disagreement in the central worth of Pu-241 has not yet been explained. The ENDRUN approach was expected to be least accurate for the calculation of central worths of light materials. However, there were a number of unexpected and unnecessary approximations which occurred in the computations which may

have adversely affected these results and which represent departures from the normal ENDRUN approach. These fall into the following categories:

- Changes in data: Essentially the same changes were made as were used for the MC² calculations. These were required because of inconsistencies between ENDF/B and ENDRUN.
- Approximations: The computational results include a number of small approximations, primarily due to present code limitations (such as truncation of data transferred from one code to another). Also, an expanded group structure was used below 37.3 eV in place of Group 26 of the specified structure.
- Hand calculations: In a few cases only partial results were obtained from earlier versions of ENDRUN and hand calculations were performed to combine them. These hand calculations represent departures from systematic processing of the data.
- Questionable results: Reviewing the group constants generated by ENDRUN revealed a few values which look questionable and still must be checked.

More accurate computations will be possible as ENDF/B data is corrected, ENDRUN checkout is completed and the TDOWN code becomes available.

7.2.2 Cross Section Files

Neutronics parameters for the 1000 MWe LMFBR reference design were computed using (a) cross sections generated from ENDF/B by the MC² computer code⁽⁷⁾ and (b) cross sections from the General Electric standard data file BASO2. The BASO2 file has a less rigorous basis in ENDF/B so the comparison provides a test of the consistency between BASO2 and ENDF/B. The results of the computations,

which were performed in 16 groups, are compared in Table 7-3.

The neutronics calculations with the alternate cross sections sets gave close agreement. The fissile enrichment and breeding ratio for the MC² calculations were about 0.6% higher and 0.01 higher, respectively, than the BASO2 calculations. Effective one-group cross sections are also given in Table 7-3 and the agreement was again found to be good.

Further analysis of the resonance parameters of Pu-239 shows good agreement between measured and computed values of alpha and the total and fission cross sections. The high resolution σ_f measurements of Schunk, et al.⁽⁹⁾ (Petrel data) have been so averaged that a direct comparison with James⁽¹⁰⁾ values can be made. Figure 7-1 shows both the Petrel and James results, along with the fission cross section computed from parameters which are consistent with the measured ORNL-RPI alpha values. The agreement between the Petrel, James, and computed data is very good, particularly below 1 keV. This agreement tends to substantiate both sets of experimental data, as well as the ORNL-RPI alpha values.

Twenty-nine-group generalized cross sections for Pu-239 have been generated with the ENDRUN code from the new resonance parameters. These cross sections are being used to compute ZPR-III, Assembly 48 parameters in order to determine the effect of the newly evaluated Pu-239 data.

7.2.3 Multigroup Cross Section Adjusting

The first results of the testing of the multigroup cross section adjusting approach show promise of good accuracy for this technique. (See Section 7.2.1, above.) Since some departures from the normal approach were necessary in this first test, the accuracy may increase as improved versions of ENDF/B and ENDRUN become available and the TDOWN code is completed.

While the major computational capability of the ENDRUN code has been confirmed, various options are now being tested. The extreme variation permitted in the form of the ENDF/B data which ENDRUN must use requires a large amount of flexibility in the code. The code has been checked out on materials containing a number of isotopes, for more than one resolved or unresolved energy range, with only partial data calculated from resonance parameters, with resonance overlap corrections, plus other options. The possibility of reading energy variable reduced neutron widths by cards gives ENDRUN a capability which overcomes a serious limitation in the present ENDF/B format.

The ENDRUN code has been modified to include an option to input a given neutron flux spectrum for use in averaging cross sections. Coding of the first version of TDOWN is about 80% completed.

7.3 CORE ANALYSIS

7.3.1 Two-dimensional Methods

The new code, DOT2DB, which incorporates the diffusion theory subroutines of the PNL code 2DB^(11, 12) into the two-dimensional transport theory code DOT,⁽¹³⁾ has been successfully run for a number of cases, including the one-group k_{eff} and fixed source problems supplied with DOT. Both the transport and diffusion theory options have been checked out on these test problems.

Tests of the transport theory option in four and 13 groups using a flat input flux guess resulted in a very slow convergence rate because of the dominance of the self-scatter source over the fission source in each group. This difficulty was not encountered when the diffusion theory option was chosen for the same problem. By using the converged flux output tape of the diffusion problems as an input guess, the transport theory calculations of the same problem converged very rapidly (only two or three more outer iterations). This approach

GEAP-5753

FIGURE 7-1. COMPARISON OF MEASURED AND COMPUTED $\text{Pu-239 } \sigma_t$ VALUES

TABLE 7-3

RESULTS OF NEUTRONICS CALCULATIONS FOR THE 1000 MWe LMFBR
REFERENCE DESIGN USING ALTERNATE CROSS SECTION SETS

	<u>Cross Section Set</u>	
	<u>From MC²</u>	<u>BASO2</u>
k_{eff}	1.00008	1.00015
Average Core Fissile Enrichment (% Pu-239 + Pu-241)	12.55	12.48
Total Breeding Ratio	1.45	1.44
One Group Cross Sections - Core Center (barns)		
U-238 σ_c	3.10	3.00
U-238 σ_f	0.045	0.045
Pu-239 σ_c	0.417	0.397
Pu-239 σ_f	1.80	1.77
Pu-240 σ_c	0.433	0.427
Pu-240 σ_f	0.360	0.357
Pu-241 σ_f	2.598	2.540

should yield a very substantial cost saving in transport problems, in addition to providing exactly comparable diffusion theory calculations of the same problems.

The convergence acceleration procedures, which were added to DOT2DB have not been checked out.

Work has begun on extending the two-dimensional flux-synthesis code BISYN to more regions and downscatter groups. These limitations of the current version have been found too restrictive. Other minor improvements are being incorporated. In the course of this work, several minor errors have already been discovered and corrected.

7.3.2 Fuel Cycle Methods

All major areas of the fuel cycle code BICYCL have been checked out. A code description is being prepared for publication.

A set of comparison problems were run using both BICYCL and the fuel cycle code RAPFU.⁽¹⁴⁾ Excellent agreement was found in most areas of comparison. In a few of the cases run a discrepancy of up to 10% in the breeding ratio was discovered. This discrepancy is believed to be due to the specification of too few types on the input. A set of problems will be run to ascertain to what extent regions may be averaged together into a single fuel type before results become unacceptable.

BICYCL has been found to be extremely flexible, allowing the user many choices on fuel management schemes and modes of calculation while retaining fairly simple input preparation.

The fuel cycle iteration routine and the reload routine for the fuel management and burnup code, FUMBLE, have been coded. The fuel cycle iteration routine permits the user to stipulate guide lines concerning the rated power, average fuel burnup, burnup reactivity, operating interval time and peak power density, by performing the adjustments which are necessary to satisfy these design criteria. It includes the capability of renormalizing the regional fluxes, adjusting the variable fissile enrichment or adjusting the operating interval time from cycle to cycle.

The reload routine of the FUMBLE code is very flexible. Fuel sources for the reloading procedure include several different makeup sources, discharged and reprocessed fuel from earlier core loadings and temporarily stored fuel, including fuel shuffling with no holdup time. A preferred source, a backup source (in case the preferred source runs dry) and a makeup source for designated core zones are permitted. The reload scheme is required to satisfy both a specified reactivity and a specified power distribution.

Coding of the FUMBLE output routine, including fuel cycle costs for specified economic assumptions, is nearly completed.

7.4 TRANSIENT AND SAFETY ANALYSIS

7.4.1 Accident Analysis

Routines which calculate radial heat transfer and point kinetics in the FREADM code have been coded and checked out.

A single axial node calculation for radial heat transfer has been coupled to the point

kinetics model. Various ways of approximating the average nodal temperature between successive heat transfer calculations are being tried using this system. Cases were run with the following limitations on the kinetics time step:

- Maximum fractional change on neutron number density = 2.0.
- Maximum estimated change in fuel temperature = 50.0°F.
- Maximum estimated reactivity change = 0.1\$.

A new heat transfer calculation was performed whenever the estimated change in average fuel temperature exceeded 50.0°F, or whenever the number of kinetics steps between heat transfer calculations reached an input number, NMK. Preliminary results with NMK equal to 1, 5, and 10 show that the number of kinetics time steps allowed for detailed calculation may be well greater than one for cases in which the temperature is slowly varying, such as transients starting from low power or scram conditions.

The sodium voiding and coolant flow model for the FREADM code has been coded and is being debugged. This model calculates the dynamic response of the primary coolant system to accident conditions such as loss of pumping power, local coolant blockages or power excursions. Sodium voiding in any of the 12 fuel bundle types can be calculated.

7.4.2 Fuel Failure and Propagation Study

In order to study the temperature distribution which results in the fuel and coolant due to accident conditions, a heat transfer model was developed. The model, which consists of a 30° symmetrical segment of a fuel pin and the adjacent coolant, was set up on the Transient Heat Transfer (THT) computer code (see Figure 7-2).

The fuel pin diameter (0.25 inch) and the cladding thickness (15 mils) must be kept constant; however, the pitch to diameter ratio may vary from 1.1 to 1.3.

In order to check out the model, a simplified sample case was run and the results were found to be identical to the analytical solution.

A series of slug flow cases were run in which the linear power was assumed to be 15 kW/ft. The circumferential cladding surface temperature difference is plotted in Figure 7-2 for different pitch to diameter ratios.

7.5 THERMAL HYDRAULICS AND FUEL ANALYSIS

7.5.1 Temperature Distribution Around an Edge Fuel Pin

A nodalized 3-dimensional model of an edge fuel pin was set up to study the effect of the pin to channel wall spacing and the wire spacer, on the circumferential cladding temperature distribution.

The model, which was set up on the THTD heat transfer code, is shown in Figures 7-3 and 7-4. Incorporated in the model are the following assumptions: pitch-to-diameter ratio of 1.2; slug coolant flow at 20 ft/sec, linear power of 15 kW/ft; wire wrap was mocked up only at the point where it contacts the channel wall.

Of primary interest was the circumferential cladding temperature distribution which was examined for various pin-to-wall spacings with and without wire wrap. The circumferential profiles calculated are for slug flow, neglecting any mixing, and therefore give an over-estimate of the gradients. The results, however, are still useful as semi-quantitative indicators of the effect of unsymmetric cooling from an edge pin.

Figure 7-3 illustrates the geometric description of the model. The entire axial length of the model is taken to be 12 inches, which is assumed to be the length for one turn of a wire wrap; i. e., the spacer is set up with one wrap per foot.

The model is divided into three axial sections for nodal analysis. There are two axial nodes each 5-1/2 inches long and then a 1-inch node on the top, giving a total of 12 inches. Note that the presence of the wire wrap is neglected except for the third, or top, axial section. The wire wrap is thus assumed to contact the cladding and channel wall for an axial distance of 1-inch in the position shown in Figure 7-3. A contact surface 5 mils wide is also assumed, although this is a rather arbitrary value. The location of the nodes, for the thermal analysis of the model, are taken at the axial midplane of each section, as shown in Figure 7-3. This gives temperature profiles at 2.75 inches, 8.25 inches, and 11.50 inches from the inlet.

The radial nodal structure is shown in Figure 7-4. Each axial section is connected by convection of the coolant. That is, the analysis is constrained to require that the coolant outlet temperature from one axial section become the coolant inlet temperature for the next downstream axial section. Axial conduction in all of the materials is neglected.

The model was set up in three dimensions so that it was possible to determine how much the axial temperature profile developed in the distance of one turn of the wire wrap, assuming a uniform inlet temperature. In this respect, the results will be pessimistic, as slug flow has been assumed with no turbulent mixing and no swirling of the flow due to the wire wraps. For comparison, the problem was also run with a completely developed axial temperature profile to see how far from this condition the actual model departs in the distance of 12 inches.

FIGURE 7-2. FULLY DEVELOPED CIRCUMFERENTIAL CLADDING TEMPERATURE DIFFERENCE (linear power = 15 kW/ft)

FIGURE 7-3. EDGE ROD AND COOLANT GEOMETRY

FIGURE 7-4. RADIAL GEOMETRY AND NODE STRUCTURE

Figure 7-5 is a plot of the circumferential cladding temperature distributions at various distances from inlet for a pin-to-wall spacing of 25 mils. It is plotted with the average cladding temperature (t_w) equal to zero. This shows the axial development of the thermal profile as the flow proceeds up the channel. Figure 7-6 compares the circumferential profile at 11.50 inches from inlet to complete axial thermal development. Figure 7-7 shows the effect of inserting a wire wrap between the channel wall and the pin for a length of 1-inch, as illustrated in Figure 7-3.

The important temperature variables for all cases examined are shown in Table 7-4.

Keeping in mind the fact that these are slug flow results, the numbers generated by these solutions should be taken as semi-quantitative indicators of the probable effects of the different situations examined with the model.

The following general conclusions about the thermal response of the system can be drawn:

- a. Using the temperature profile around an unsupported pin, a local temperature "spike" or perturbation can be superimposed on this to indicate the presence of a wire wrap.
- b. It seems apparent that a reduced wire wrap thickness at the wall is an effective means of reducing the temperature variation caused by local coolant channel asymmetry. This analysis indicates that the reduced wrap thickness should be in the area of 10-20 mils.
- c. The axial development of the temperature profile requires a long distance because of the asymmetry of the coolant channel, in excess of the 12-inch model examined here. This justified the necessity of using a three-dimensional model in the analysis, as opposed to assuming complete axial

thermal development which can be analyzed using a simpler 2-dimensional model. However, the existence of turbulence and swirl flow in the actual core should enhance axial development of the temperature profile.

- d. The magnitude of the hot spot under the spacer, when adjacent to the channel wall, increases as the wire wrap thickness decreases, since the pin is being denied coolant in a critical thermal area. However, as very small pin-to-wall spacings are approached (e.g., 10 mils and less), the spacer ΔT decreases since there is now almost a negligible amount of coolant between the unsupported pin and the channel wall. Thus, the ΔT under the wire wrap undergoes a maximum somewhere between 50 and 10 mils.

Because a large contact area was used for the wire wrap, and axial conduction in the cladding around the wrap neglected, the value of this hot spot ΔT will not exceed a value somewhere in the neighborhood of 50°F.

- e. The contact coefficient between the spacer and the cladding will increase the hot spot ΔT , possibly by as much as 50% to 75%, but an internal gas gap in the wire wrap has almost a negligible effect on the hot spot ΔT .

7.5.2 Hydraulics

A review of the available experimental pressure drop data for singly wire wrapped rod bundles, arranged in a triangular array, has been completed. A method of extrapolating the data was developed and is being checked out. The experimental data that has been used is that from the CVTR⁽¹⁵⁾ tests, Karlsruhe,⁽¹⁶⁾ and some recent data obtained at PNL.⁽¹⁷⁾ The data was plotted as the ratio of the pressure drop for the wire wrapped bundles to that for an identical bundle without wire wraps. The primary variables were found to be the

FIGURE 7-5. CLADDING TEMPERATURE AT VARIOUS DISTANCES FROM INLET

FIGURE 7-6. CLADDING TEMPERATURE FOR UNSUPPORTED PIN

FIGURE 7-7. CLADDING TEMPERATURE AT 11.50 INCHES FROM INLET

TABLE 7-4

TEMPERATURE DIFFERENTIALS FOR AN EDGE PIN

	<u>Pin-to-Wall Spacing</u>		
	<u>50 Mils</u>	<u>25 Mils</u>	<u>10 Mils</u>
ΔT across unsupported pin 11.50 inches from inlet ($^{\circ}\text{F}$)	106	45	-45*
ΔT across unsupported pin with complete axial thermal development ($^{\circ}\text{F}$)	670	330	35
ΔT_s^{**} under spacer at the wall ($^{\circ}\text{F}$)	17	32	24
ΔT_{sh}^{**} under spacer at the wall with contact coefficient to the cladding ($^{\circ}\text{F}$) (h-1000)	34	48	40

* Negative sign indicates $\theta = 180^{\circ}$ temperature is greater than $\theta = 0^{\circ}$ temperature.

** This is the temperature perturbation under the wire, as compared to an unsupported pin (see Figure 7-7).

number of wraps per foot, the rod pitch-to-diameter ratio, the distance of the outer row of rods from the channel wall and the number of rods per bundle. A secondary Reynolds number effect was also observed.

Using the assumption that the pressure drop can be separated into a component due to friction along the wires and pins and another due to the drag over the wires, the following expression was written:

$$\frac{\Delta P}{\Delta P_o} = \alpha + \beta n^2$$

where:

α = theoretically derived frictional term

β = experimentally derived drag term

n = number of wire wrap turns per foot of rod length

ΔP = wire wrapped bundle pressure drop

ΔP_o = the experimental or calculated pressure drop for an identical bundle without wire wraps. ΔP_o was calculated by a method described in Reference 18.

7.5.3 Fuel Swelling

7.5.3.1 Model Development

The intragranular swelling* caused by the accumulation of fission gas within grains has

* Swelling within a grain.

been calculated assuming that bubble spacing during growth can be predicted by the expression:

$$S = \sqrt[6]{\frac{1}{18} \frac{D}{G}} r_o^2 a^2$$

where:

D = Diffusion constant for fission gas assumed:

$$= 6.6 \times 10^{-6} \exp(-42,200/RT) \text{ (cm}^2/\text{sec)}$$

G = Gas generation rate atoms gas/atom site-second

r_o = Fission gas bubble radius (cm)

a = Lattice parameters (cm)

Using this spacing assumption, the swelling due to intragranular bubbles is

$$\frac{\Delta V}{V} \text{ (Gas in grain)} = \frac{237 r_o^2}{r_f} \left[\frac{Q}{D} \right]^{1/2}$$

where:

r_f = outer radius of fuel (cm)

Q = linear power (kW/ft)

The fission gas bubble radius (r_o) was determined by assuming the gas pressure in the bubble to be restricted by surface tension, fuel viscosity, and local hydrostatic pressure. The calculated intragranular swelling as a function of bubble size and temperature is shown in Figure 7-8.

During the period, a study was made to determine the growth that could be expected as bubbles traverse to grain boundaries. It was assumed that the volume after coalescence is determined by equating the amount of gas in two bubbles which coalesce with the amount of gas in the bubble produced, assuming the

pressure before and after coalescence is determined by surface tension forces. This assumption leads to the result that after N encounters with equal-sized bubbles, the final bubble will have a radius $\sqrt{N+1}$ times the radius of the bubble encountered. As the sweeping bubble grows in size it sweeps out a larger and larger cross section. Consequently, the rate of encounters with equally spaced bubbles will increase with distance swept. The relation between the radius of the final bubble (r_f) with that encountered (r_i) is

$$\ln \left[\frac{r_f}{r_i} \right]^2 = \pi r_i^2 N_v d$$

where:

N_v = number of bubbles per unit volume

d = distance travelled

Again assuming that bubbles are spaced in a FCC lattice according to the previous expression the final radius of the bubble will be related to the size of the bubble encountered and the distance moved by

$$r_f = r_i \exp \left[\frac{6.29 r_i d}{a} \left[\frac{G}{D} \right]^{1/2} \right]$$

The above expression was solved as a function of temperature and $\phi = 6.29 \psi d$ (Figure 7-9). Here $\phi/6.29d$ corresponds to a probability of coalescence once a bubble is encountered. Typically, the grain diameter (maximum d) is between 5 microns and 40 microns. For a bubble to increase by an order of magnitude while traversing a grain, either the probability of coalescence must be about 0.02 or the coefficient of the diffusion constant should be increased. An increase in size of more than an order of magnitude while traversing a grain should result in the bubble breaking away from the boundary and continuing on to the fuel center.

FIGURE 7-8. PREDICTED FUEL SWELLING DUE TO INTRAGRANULAR BUBBLES AT A LINEAR POWER OF 16 kW/ft AS A FUNCTION OF BUBBLE SIZE AND TEMPERATURE. (RADIUS OF THE FUEL (r_f) = 0.11 INCH, DIFFUSION CONSTANT (D) = $6.6 \times 10^{-6} \text{ EXP } (-42200/RT) \text{ cm}^2/\text{sec.}$)

FIGURE 7-9. PREDICTED SIZE OF BUBBLES REACHING GRAIN BOUNDARIES AS A
 FUNCTION OF FUEL TEMPERATURE AND ϕ , WHERE $\phi = \text{DISTANCE}$
 TRAVELLED TIMES PROBABILITY OF COALESCENCE TIMES 6.29

The experimental evidence which is beginning to accumulate suggests that the diffusion coefficient may be larger than that used in this analysis. British investigators have recently suggested that the diffusion constant used here may be low by approximately 200 times. Consequently, the probability of coalescence is probably greater than 0.02 and the equation which predicts growth by coalescence may be brought into better correlation by increasing the diffusion constant (D).

7.5.3.2 Physical Properties Studies

During the period, a literature search was made to find a suitable coefficient of friction for use between the fuel and cladding in order to better predict axial fuel expansion. Although no measurements were found between UO_2 or (U, Pu) O_2 and stainless steel, a value of 1.2 between alumina and stainless steel was found.⁽¹⁹⁾ The data obtained at 400°C to 600°C is thought to approximate the coefficient of friction between fast reactor fuel and stainless cladding.

7.5.3.3 Correlation with Experimental Data

A statistical method has been developed and tested to analyze the Task F irradiation data in terms of the effect of fabrication and operating variables on the fuel swelling induced cladding diametral expansion.

The post-irradiation profilometer traces of the Task F pins show an increase in diameter along the length of the pin. Fuel cladding irradiation swelling contributes significantly to this cladding diameter increase. The cladding swelling contribution can be estimated from the recent Task L studies. If it is postulated that these are the only significant strain mechanisms, the fuel swelling contribution is obtained by difference. The operating variables (burnup, linear power, cladding temperature, etc.), the fabrication variables (pellet density, fuel-cladding gap, cladding material and thickness,

etc.), and the location of each pellet are known as a function of length along the pin. It is thus possible to associate a cladding strain due to fuel swelling with values of the variables at each pellet location.

The statistical techniques of multiple linear regression have been adapted to this analysis. The cladding strain at any pellet location could be predicted from the arithmetic mean of the data using the standard deviation:

σ_0 = standard deviation of the mean

$$= \sqrt{\frac{1}{N} \sum_j (y_j - \bar{y})^2}$$

where:

y_j = measured percent $\Delta D/D$ at jth pellet location

$\bar{y}$ = arithmetic mean percent $\Delta D/D$

The FITTER subroutine solves the equation for the coefficients of up to nine variables of a multiple linear correlation equation,

$$y = A_0 + A_1X_1 + A_2X_2 + \dots + A_9X_9$$

and calculates two measures of goodness-of-fit of the correlation to the data:

- a. The statistical correlation coefficient is a measure of the ability of the correlation to predict the data relative to predicting the data from the arithmetic mean:

r = statistical correlation coefficient

$$= \sqrt{1 - \left(\frac{\sigma_1}{\sigma_0}\right)^2}$$

σ_1 = standard deviation of the correlation

$$= \sqrt{\frac{1}{N} \sum_j (y_j - y'_j)^2}$$

y'_j = calculated percent $\Delta D/D$ at the j th pellet location.

The closer the correlation coefficient to unity, the better is the correlation (statistically) than is the arithmetic mean. Of two correlations, the one with the larger r gives, of course, the better fit of the data.

- b. The design correlation coefficient is a measure of the correlation relative to a simplified design correlation:

q = design correlation coefficient

$$= \sqrt{1 - \left(\frac{\sigma_1}{\sigma_r}\right)^2}$$

σ_r = standard deviation of the reference design correlation.

A simplified equation in terms of the variables having the greatest effect on the results is often most useful in preliminary design parametric studies. This form of the correlation coefficient shows quantitatively, on a full-scale of zero to unity, the gain in goodness-of-fit possible with a more extensive correlation.

The statistical correlation method has been successfully demonstrated on one Task F pin. The method is now ready to be applied to a group of pins and statistically significant functions of the variables established.

7.5.3.4 Development of Fuel Swelling

Work continued on writing and debugging a computer program to predict fuel swelling based upon the model developed in Section 7.5.3.1. Fuel swelling is assumed to be produced from fission gas bubbles within grains, bubbles located at grain boundaries, and solid fission products.

There is assumed to be a maximum bubble size that can exist within grains. This maximum size is determined by the local temperature and local temperature gradient. If the bubble size is calculated to be less than the critical size, the local swelling is calculated from the sum of that due to solid fission products and intragranular bubbles. If the bubble size is larger than critical and the fuel temperature is less than that which causes pore sweeping to the fuel center to occur, the swelling is assumed to be caused by bubbles located at grain boundaries and solid fission products. In this region initial fuel porosity is assumed to close before swelling begins. In the portion of the fuel hotter than that required for sweeping to the fuel center, the initial porosity is assumed to move to the fuel center and not be available to accommodate swelling. In this region swelling is assumed to come from solid fission products only. Writing and debugging of a simplified computer program has been completed and trial runs to investigate different variables are beginning.

SECTION VIII

TASK L—CLADDING DEVELOPMENT

8.1 GENERAL

The technical and economic feasibility of ceramic-fueled fast reactors depends on

the attainment of high levels of performance of the fuel system, especially the survival of the cladding at temperatures and specific powers of practical interest. The present

status of testing and analysis of potential cladding material with regard to alloys and environmental variables indicates good prospects that adequate initial performance of cladding for fast reactor fuel elements can be obtained by using carefully specified materials within the range of compositions of commercial processing methods. No material, however, has yet been tested under fully typical conditions. Data are not yet available to establish optimum compositions or structures, although some tentative trends have been suggested. This lack of test experience to support the rational selection of preferred cladding specifications is regarded as a major remaining obstacle to the attainment of the reliable operation of an LMFBR fuel system.

The general objectives of the cladding development program for the next 4 years are to contribute to the attainment of the desired levels of fast reactor fuel element performance which, in turn, are required by the desired fuel-cycle economics. These objectives of specific activities are:

- a. Determine the performance characteristics—including mechanical properties, fracture characteristics, radiation damage, and structural stability—of the most likely cladding material candidates under test conditions simulating LMFBR environmental parameters. The candidate materials are within the range of compositions and structures which can be produced by present commercial processes.
- b. Obtain data and 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.

- c. Determine the in-depth material behavior required to establish performance limitations (such as allowable stress levels, creep rates, maximum strains, local temperatures), on selected alloys, based on initial fast-flux test results, and recommend preferred material specifications and design limits for LMFBR use.

8.2 MATERIAL IRRADIATION PERFORMANCE

8.2.1 Series L-2, L-6, L-14, and L-15 — EBR-II Materials Capsules

Irradiation of L-6, L-15 series of material capsules is continuing in EBR-II. The L-14 series capsules have passed the target total fluence of 2.0×10^{22} n/cm² and were discharged in December. Four of the L-2 material capsules, removed earlier from S/A XG05 are still waiting reinsertion into another S/A for further exposure. The other L-2 capsule (L-2I) was transferred to S/A XO39 and accumulated additional exposure in row 4. This capsule was discharged in December and will be examined with the L-14 capsules. As of December 31, 1968, capsule L-2I, and the capsules L-6, L-14, and L-15 have accumulated the following total fluences: 5.9, 4.9, 3.0 and 1.4×10^{22} n/cm² respectively. (Unless otherwise noted, all reported fluences are based on EBR-II physics predictions.) Complete descriptions of these series and their experimental objectives have been given previously. (20, 21)

8.2.2 Series L-10 EBR-II Material Capsules

A description of these ten capsules has been presented earlier. (20) These capsules were discharged from EBR-II in October 1968 and were shipped in early January.

The L-10 material capsules were irradiated in the XO16 subassembly in row 4 of EBR-II for $\sim 7,894$ MWd exposure or ~ 175 EFPD*. An estimated total fluence of 2.8×10^{22} n/cm² was accumulated in the peak core position. Post-irradiation examination of the L-10 capsules is scheduled for the next reporting period.

8.2.3 Uniaxial Testing of Irradiation Materials

The objective of this program is to evaluate the mechanical behavior of EBR-II irradiated cladding materials using uniaxial testing techniques including both tensile and ring ductility tests. Initial short term tests have been performed and will be extended to tests at lower strain rates later in the program.

8.2.3.1 Ring Ductility Tests

Ring ductility tests were conducted this quarter on irradiated tubular specimens from capsules L4F (Type 304) and L4D (Type 316).

Specimens in these capsules were given a pre-irradiation heat treatment of 1650°F for 24 hours to precipitate and agglomerate carbides at grain boundaries.

Capsule L4D was irradiated in the EBR-II to a peak fluence of 3.5×10^{21} n/cm² ($E_n > 1$ MeV).** These tubular sections were irradiated at temperatures of 1000°F to 1300°F for ~ 2860 hours. Capsule L4F was irradiated to a peak fluence of 3.1×10^{21} n/cm² ($E_n > 1$ MeV),* at temperatures of 1000°F to 1300°F for 2000 hours. A complete description of these capsules has been presented earlier with results regarding the post-irradiation

testing of tubular and flat tensile samples. (22,3) The ring test technique used was described in a previous quarterly report. (3)

The experimental design consisted of testing 0.250-inch o.d. rings 0.075-inch wide in triplicate at four strain rates. Failure strains ($\Delta D/D_0$) were obtained from pre- and post-test diameter measurements using an optical comparator. The average strain rate was calculated by dividing the failure strain by the time to failure and averaging the values for all of the tests at each machine cross-head rate.

The results (Table 8-1 and Figure 8-1) showed a general decrease in the circumferential strain with decreasing strain rate for both materials with Type 304 exhibiting more ductility than Type 316. For strain rates of 0.035 to 0.00032 min⁻¹, typical $\Delta D/D_0$ values for Type-304 core specimens ranged from 20% to 31%, whereas values for similar Type-316 specimens ranged from 5% to 12%. When these results are compared to earlier tensile test results on similar irradiated specimens, it was observed that the ring test ductility ($\Delta D/D_0$) was greater than the tensile test ductility ($\Delta L/L_0$) (Figure 8-2).

Work is now completed on the L-2', L-4, and L-4' capsules and a topical report for these capsules is in preparation.

8.2.3.2 Tensile Tests

Tensile tests were performed on specimens cut from the XG05 hex-tube material to determine the uniaxial mechanical properties for highly-irradiated Type-304 stainless steel ($\sim 10^{22}$ n/cm², > 1 MeV). The specimens were

* EFPD = Effective full power days at 45 MW.

** These fluences were determined from flux wire activation analysis.

* These fluences were determined from flux wire activation analysis.

TABLE 8-1

1300°F RING DUCTILITY RESULTS FOR
CARBIDE AGGLOMERATED TYPES 304 AND 316

	Specimen Number	Neutron Exposure $\times 10^{21} \text{ n/cm}^2$ ($E_n > 1 \text{ MeV}$)	Average Strain Rate min^{-1}	Failure Strain %
C. A.				
Type 304	2	0.8	0.035	31
	3	2.4	0.035	27
	5, 7, 9	3.0-3.1	0.035	25, 30, 31
	6, 10	3.0-3.1	0.00037	20, 29
C. A.				
Type 316	1, 5, 9	0.2-0.3	0.35	40, 53, 46
	2, 6, 10		0.035	29, 32, 33
	3, 7, 11		0.0031	23, 21, 20
	8, 12		0.00037	17, 18
	13, 14, 15	2.2-2.4	0.035	20, 19, 15
	16, 20, 24	3.0	0.35	17, 21, 19
	17, 21, 25		0.035	12, 12, 10
	18, 22, 26		0.0031	7, 9, 5
	19, 23, 27		0.00037	6, 11, 9
	28, 29, 30	3.5	0.035	22, 22, 24

cut from 2-inch sections of the hex tube with a Tensilkut machine to a geometry similar to Task-L material specimens (Figure 8-3). Specimens were cut from ten axial locations: five in the core region, one just below and one just above the core and three in the upper blanket. The test temperature was similar to the irradiation temperature, 700°F to 790°F. The results are shown in Table 8-2 and presented graphically as a function of axial location in Figure 8-4.

In general, the 0.2% offset yield stress was more than tripled from about 25,000 psi to 30,000 psi to about 78,000 psi to 101,000 psi for the core region. The total elongation

correspondingly decreased from the range of 26% to 29% to the range of 2.6% to 3.5% for the core region. In the upper blanket region, the yield stress decreased with increasing distance from the core and approached the unirradiated values (31,000 psi) at 28 to 30 inches from the core centerline. The total elongation increased with increasing distance from the core and also approached the unirradiated values (25%) at 28 to 30 inches from the core center. Although the irradiation temperature varied from 700°F at the bottom to 790°F at the top of core region, the effects of this 90°F rise in temperature were not apparent on either the strength or the ductility of the hex tube.

FIGURE 8-1. EFFECTS OF DEFORMATION RATE ON RING DUCTILITY AT 1300°F FOR IRRADIATED CARBIDE AGGLOMERATED TYPE 304 AND 316

FIGURE 8-2. COMPARISON OF POST IRRADIATION DUCTILITY AS MEASURED IN RING DUCTILITY AND TENSILE TESTS FOR VARIOUS FLUENCE LEVELS AT 1300°F

FIGURE 8-3. TENSILE SPECIMEN CUT FROM XG05 HEXAGONAL TUBE

The width of each tensile specimen was measured with an optical comparator before and after tensile testing at the fracture. The percent reduction in width was used as an indication of the reduction in area of the specimens. The values obtained ranged from 14% to 23% in the core region and 18% to 27% in the blanket region, indicating that substantial localized ductility existed below 790°F for the hex-tube material. This amount of necking suggests that the embrittlement as measured by total elongation, observed for the hex tube, was caused by a decrease in the work hardening ability of the material because of irradiation hardening and not the elevated temperature type of embrittlement observed at the higher temperatures which is attributed to grain boundary weakening. The latter type of embrittlement results in very little if any necking.

The strain hardening exponent (n) was also determined from the true stress-strain curves using the relationship $\sigma = K \epsilon^n$, where

σ = true stress, ϵ = true strain, K and n are constants. Typical n values for the unirradiated specimens were 0.20 to 0.22 compared with values of 0.005 to 0.014 for the core region specimens. The n values remained fairly constant throughout the core region and increased with distance above the core in the upper blanket region similar to the total elongation curve shown in Figure 8-4. The lower n values add further support that the cause of embrittlement observed in the irradiated hex-tube material was caused by a decrease in work hardening ability.

The PNL thimble tube tensile data⁽²³⁾ are also shown in Figure 8-4 to illustrate the similarities in both sets of data. The thimble tube received approximately the same peak total fluence. Except for the lower core region, the yield strength data was remarkably similar for both the thimble and hex tube. The degree of radiation hardening observed in the lower core region appears to be less in the hex tube (Row 4) than the thimble tube

TABLE 8-2

TENSILE PROPERTIES OF THE XG05 HEX TUBE

<u>Axial Location*</u> (inches)	<u>Fast Fluence</u> (n/cm, >1MeV)	<u>Temperature</u> (° F)	<u>Stress (ksi)</u>		<u>Elongation (%)</u>	
			<u>Yield</u>	<u>Ultimate</u>	<u>Uniform</u>	<u>Total</u>
23	6.5×10^{19}	790	40/41	63/69	17/18	20/21
15	6.0×10^{20}	790	51/68	72/77	6.1/7.9	8.1/11
9	3.0×10^{21}	790	72/77	86/94	2.7	4.7
7	5.0×10^{21}	790	86	87	0.9	2.6
5	7.4×10^{21}	775	85/85	92/96	1.1/1.3	3.4/3.9
3	9.0×10^{21}	765	98/107	99/107	0.7/0.8	2.8/3.2
-7	5.0×10^{21}	700	78/97	96/97	0.7/1.2	3.0/3.7
+7	5.0×10^{21}	790	86	87	0.9	2.6
-5	7.4×10^{21}	730	88/94	88/95	0.8/1.2	2.7/3.1
+5	7.4×10^{21}	790	85/85	92/96	1.1/1.3	3.4/3.9
-1	9.6×10^{21}	750	96/101	96/102	0.6/0.8	2.8/3.0
+1	9.6×10^{21}	760	83/93	98/100	1.0/1.4	3.4/3.5
Unirradiated	0	700-800	25/30	59/63		26/29
Unirradiated	0	842	24	62	26	27

* Distance from center plane of core.

FIGURE 8-4. TENSILE PROPERTIES OF THE XG05 HEXAGONAL TUBE

(Row 5). It is believed this difference is caused by the large scatter in the hex-tube data as shown in Figure 8-4 and is not a real significant difference.

8.2.3.3 Electron Microscopy Examination of Tensile Specimens L-2' and L-4'

Both optical and electron metallography have been used to elucidate differences observed in the short term tensile properties of irradiated Type 304 for two microstructural conditions. Two of the tensile specimens examined, No. 212 and No. 218, were mill-annealed (1800°F for 15 minutes) prior to irradiation. The third specimen examined, No. S7, was carbide agglomerated (mill-annealed plus 1650°F for 24 hours). Both mill-annealed specimens were irradiated in L-2' capsules to 6.2×10^{21} n/cm² (>1 MeV),* whereas, the carbide agglomerated specimen was irradiated in the L-4' capsules to 3.0×10^{21} n/cm² (>1 MeV).* All specimens were tensile tested at 1300°F. The strain rate for specimens No. S7 and No. 218 was 0.02 min⁻¹, and for specimen No. 212, 0.0002 min⁻¹.

Earlier examinations⁽⁵⁾ of optical photomicrographs showed that incipient grain boundary cracking occurred throughout the slowly strained mill-annealed specimen, while cracking was localized at the fracture in the rapidly strained specimen. The carbide-agglomerated specimen exhibited grain distortion, suggesting that considerable intragranular deformation was realized near the fracture. Large agglomerated carbides were also observed in grain boundaries of this specimen similar to those present prior to irradiation.

Earlier transmission electron microscopy (TEM)⁽⁵⁾ provided evidence of polyhedral cavities in these irradiated specimens. These

cavities were initially correlated with the density changes observed in these materials.⁽²⁾ Figure 8-5 is a typical TEM micrograph of the gage length from specimen No. 218 and shows the cavities but the contrast is such that dislocations are also visible. The dislocations have interacted with the cavities forming jogs and intersections. There is a slight denudation of cavities near the grain boundary. Similar results were observed for specimen No. 212, except the dislocation density in this specimen was noticeably less.

It is believed that the radiation induced strengthening observed for these specimens arises from the interaction of cavities with dislocations. The cavities modify the dislocation mobility. These observations lend support to the work at PNL⁽²⁴⁾ which showed that above 1200°F radiation induced strengthening was associated with cavities or the combined action of cavities and the dislocation network.

To reveal the fine microstructure in more detail, chromium-shadowed carbon replicas of optical metallographic surfaces from these specimens were also examined. Specimen No. 218 has a continuous carbide network in grain boundaries as well as in coherent and incoherent twin boundaries (Figure 8-6). Short intergranular cracks have formed in some of the surface grain boundaries but not in twin boundaries. No cracks were found with a depth of more than one grain.

Specimen No. 211 (same condition as No. 212) exhibited a general carbide distribution similar to No. 218. However, because of the slow strain rate during the tensile test, surface cracks were found in every grain boundary (Figure 8-7). These cracks were much deeper than those observed in specimen No. 218 and often extended for several grains. The present electron microscopy results support an earlier hypothesis that the differences in the stress-strain curves for specimen No. 211 and No. 218 were associated with time-dependent propagation of incipient cracks in the slowly-strained sample

* These fluences were determined from flux wires activation analysis.

FIGURE 8-5. TRANSMISSION ELECTRON MICROGRAPH FROM GAGE SECTION OF FAILED SPECIMEN NO. 218 (Mill-Annealed Type 304)

FIGURE 8-6. SURFACE MICROSTRUCTURE OF FAILED SPECIMEN No. 218 (Mill-Annealed Type 304)

FIGURE 8-7. SURFACE MICROSTRUCTURE OF FAILED SPECIMEN NO. 211 (Mill-Annealed Type 304)

versus localized and rapid propagation of cracks in the sample tested at the 0.02 min^{-1} .⁽⁵⁾

In contrast to irradiated mill-annealed Type 304, the carbide-agglomerated specimen No. S7 contained well rounded individual carbides in the grain boundaries (Figure 8-8). A few short intergranular cracks were found on one surface but not on the other. Thus, the detailed examination using replication electron microscopy agrees with the earlier observation that the agglomerated carbides at grain boundaries were retained after the irradiation exposure. The following discussion is offered as an explanation for the greater resistance to radiation embrittlement of carbide-agglomerated Type 304 over mill-annealed Type 304.

Precipitates within the grains may restrict dislocation mobility (matrix strengthening) so that stress concentrations may be established at grain-boundary triple points properly oriented. If grains are stronger than the grain boundaries, deformation will result from shear stresses in boundaries. This shearing then produces a stress at a grain junction which can exceed the strength of the boundaries and open a microcrack. Continued sliding causes the crack to grow and propagate. Matrix precipitation may arise from elevated-temperature exposure or be strain-induced.

An additional explanation for the improved resistance of carbide-agglomerated structures could be involved with tramp elements in grain boundaries. Trace impurity elements, absorbed at grain boundaries, can reduce the surface energy for fracture (γ). The Stroh relation⁽²⁵⁾ shows the importance of γ on the minimum shear stress (σ_s) required to initiate a wedge crack,

$$\sigma_s = \left(\frac{12\gamma G}{\pi L} \right)^{1/2}$$

where G is the shear modulus, and L is the grain boundary length between obstacles such as triple points of carbides.

It can be seen that a reduction in γ will reduce σ_s . However, carbides in grain boundaries can act as a "sponge" to collect the trace elements and effectively remove them from boundary surfaces, thus increasing γ which increases σ_s . Carbides also, in effect, reduce L by acting as obstacles to grain boundary sliding. Observations⁽²⁶⁾ have been made on irradiated Type 304 that suggest grain boundary carbides can increase the stress at which wedge-type cracks (obtained from grain-boundary sliding) are nucleated; the propagation rate was also reduced. The degree of radiation embrittlement was reported to be reduced.

8.3 CLADDING SWELLING PROGRAM

The objective of this program is to establish the relationship between cladding swelling rates and the irradiation variables presently thought to affect cladding swelling: flux, fluence, temperature, and alloy microstructure and chemistry.

The measurements used to establish cladding swelling include immersion density changes, dimensional changes, and structural changes as observed in Transmission Electron Microscopy (TEM).

A comprehensive density survey of two XG05 capsule tubes which contained Task F fuel pins was completed this quarter. The examinations were performed on specimens spanning the entire length of Capsule Tube F2T and the core region only of F2R. Because of the detailed post-irradiation dimensional examination performed on both capsule tubes, these specimens provided an opportunity to evaluate the diametral changes resulting from neutron radiation induced swelling.

Both tubes were welded and drawn Type-304 stainless steel with a nominal outer diameter and wall thickness of 0.375-inch and 0.020-inch, respectively. The as-received tubing microstructure was that of a fully annealed

**FIGURE 8-8. SURFACE MICROSTRUCTURE OF FAILED SPECIMEN
No. S7 (Carbide Agglomerated Type 304)**

material, with an average ASTM grain size of 8. The certified chemistry of heat No. 136272, from which all F2 capsule tubes were fabricated, is shown in Table 8-3.

TABLE 8-3

CERTIFIED CHEMISTRY OF TYPE 304TUBING, HEAT NO. 136272

CARBON	0.07
MANGANESE	1.55
PHOSPHORUS	0.027
SILICON	0.48
CHROMIUM	18.46
NICKEL	9.35
SULPHUR	0.019
IRON	Balance

The material was irradiated in the as-received condition. The radiation history is shown graphically in Figure 8-9. Metallographic examination results of a representative capsule tube after irradiation were reported previously.⁽²²⁾

After irradiation and prior to sectioning, continuous profilometry scans were made along the entire length of both capsule tubes. The chart recordings revealed that measurable diametral swelling had occurred in the core-exposed regions of both tubes. Diametral increases as large as 3.65 mils relative to the nominal diameter were measured. Since these tubes were neither restrained in the axial direction nor mechanically strained radially during irradiation, it is assumed that this damage is caused solely by cladding swelling. The density measurements discussed below were performed to test this assumption and to provide a numerical correlation between diametral expansion and total volume increase.

Both capsule tubes were sectioned into 3-inch segments according to the general sectioning plan shown in Figure 8-9. The density of each segment was determined by the immersion technique.

The results of these measurements are listed in Table 8-4. The general trend of volume increase with fluence is similar to that shown by previous work in this program and elsewhere. The higher fluence data in fact strengthens the validity of extrapolations to total fluences as high as 7.5×10^{22} (the Cawthorne and Fulton data point on M316L).⁽²⁷⁾

The diametral expansion of the tube follows a trend similar to the volume expansion. The correlation between diametral and volume increase is more clearly shown in Figure 8-10. From the slope of the curve, the following relationship is suggested:

$$\frac{\Delta D}{D_0} = \frac{3}{8} \frac{\Delta V}{V_0}$$

Although this implies that diametral expansion is slightly greater than one-third of total volume expansion, the one-third relationship is within the error band. The assumption of isotropic swelling in an unstrained tube appears to be valid.

Although it appears that swelling is isotropic in the absence of stress, no substantial evidence has yet been advanced to show that the imposition of stress influences the directionality of cladding swelling. The density examination of fuel pin cladding which has undergone significant diametral change should provide such evidence. Work is currently in progress to establish an acceptable sample preparation technique.

Surveillance of capsule tubes will continue as higher fluence materials become available. Additional data on unstressed capsule tubes

FIGURE 8-9. RADIATION HISTORY AND SECTIONING PLAN FOR A REPRESENTATIVE XG05 FUEL PIN CAPSULE TUBE (F2T)

TABLE 8-4

DENSITY MEASUREMENTS ON F2 FUEL PIN CAPSULE TUBES

Tube No.	Total Fluence	Density (gm/cm ³)*	Average	
			$\frac{\Delta\rho}{\rho_0}$ (%)	$\frac{\Delta D}{D_0}$ (%)
F2T-2	2.9×10^{20}	7.8942	0.15	0
F2T-3	5.0×10^{20}	7.8899	0.20	0.03
F2T-4	1.5×10^{21}	7.8895	0.21	0.03
F2T-5	3.3×10^{21}	7.8821	0.30	0.05
Control**	--	7.9058	--	
F2T-7a	2.0×10^{22}	7.8410	0.68	0.29
F2T-7b	3.5×10^{22}	7.7821	1.43	0.59
F2T-7c	4.6×10^{22}	7.7298	2.09	0.80
F2T-7d	4.8×10^{22}	7.7465	1.88	0.72
F2T-7e	3.8×10^{22}	7.7830	1.42	0.48
Control	--	7.8948	--	
F2R-5	3.5×10^{22}	7.7833	1.41	0.51
F2R-4	4.6×10^{22}	7.7319	2.06	0.77
F2R-3	4.8×10^{22}	7.7410	1.95	0.73
F2R-2	2.8×10^{22}	7.7884	1.35	0.51

* Average of four determinations.

** Although the control sample is the same in both cases, a control run is made with each group of analyses. Hence the difference.

FIGURE 8-10. VOLUME CHANGE IN F2 FUEL PIN CAPSULE TUBES AS A FUNCTION OF DIAMETRAL CHANGE

should not alter the correlation of volume with diametral expansion presented here but are essential in extending the swelling-versus-fluence curve to allow more accurate extrapolation to end-of-life conditions.

8.4 MATERIALS MANAGEMENT AND SPECIFICATION PROGRAM

8.4.1 Quality Control of Tubing

Detailed quality characterization of all tubular material slated for in-reactor experimental programs is the foremost responsibility of the material management program. The purpose of such characterization is threefold: (a) to provide detailed dimensional data that will allow a more custom fuel pin assembly, (b) to establish the pre-irradiation condition of the cladding so that accurate assessment of radiation damage can be made, and (c) to assess variances in

quality of tubing, within and between lots, to determine appropriate areas for improvement.

Identical orders for seamless tubing were placed with four vendors to provide a broad range of tubing for this quality assessment and, at the same time, to satisfy the needs of current irradiation programs. Tube vendors responding to this request included: Superior Tube Company, Handy and Harman Tube Company, Trent Tube Company and Matthey Bishop Inc. Each vendor delivered 1000 feet of Type 304H and 2000 feet of Type 316H in the mill-annealed condition. The size of all tubing was 0.250-inch outer diameter, 0.015-inch wall.

Detailed inspection of representative quantities of each tubing heat, which includes chemistry, structure, dimensions, surface condition and mechanical properties, is in progress.

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APPENDIX

PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Number 1—94, from July 1959 through October 1968.

GEAP-3888	FCR Development Program - First Quarterly Report, October—December 1961.	GEAP-4931	FCR Development Program - Fifteenth Quarterly Report, May—July 1965.
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GEAP-3981	FCR Development Program - Third Quarterly Report, April—June 1962.	GEAP-5098	FCR Development Program - Seventeenth Quarterly Report, November 1965—January 1966.
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CONTRIBUTORSTASK B—Defected Fuel Behavior

D. L. Brown	R. E. Murata
N. W. Brown	P. E. Novak
J. T. Cochran	H. C. Pfefferlen
F. R. Coffman	D. E. Plumlee
G. D. Collins	L. E. Pohl
W. L. Conant	N. K. Schad
M. D. Dermer	K. M. Seidel
S. K. Evans	M. D. Swadley
R. H. Eymann	D. M. Switick
M. G. Farschon	G. W. Tunnell
D. D. Freeman	W. O. Wade
J. J. Hagerty	E. E. Walker
G. L. Howe	L. W. Waters
R. W. Jones	D. L. Zimmerman
T. E. Lannin	

TASK C—Safety Experimentation

R. J. Burrill	H. C. Pfefferlen
M. D. Dermer	R. E. Smith
R. H. Eymann	G. L. Spence
J. H. Field	G. L. Stimmell
T. Hikido	M. L. Thompson
R. H. Hollingsworth	R. C. Thompson
R. W. Jones	G. R. Thomas
T. E. Lannin	W. O. Wade
S. M. Lukezic	D. L. Zimmerman
L. E. Pohl	

Program Planning and Coordination

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

TASK E and F—Fuel PerformanceEvaluation and Integral Fast Flux
Irradiations

R. R. Asamoto	L. S. Lee
A. L. Darzin	J. E. Lewis

G. C. Berg	G. F. Melde
P. E. Bohaboy	E. J. McAlpine
D. A. Cantley	R. C. Nelson
A. E. Conti	K. J. Perry
C. N. Craig	H. J. Rosenbaum
E. S. Darlin	B. F. Rubin
R. Erwin	C. M. Ryer
R. H. Eymann	D. W. Sandusky
G. R. Hull	N. C. Shirley
T. A. Koppel	F. H. Scott
E. F. Kurtz	M. L. Thompson
T. E. Lannin	D. L. Zimmerman

TASK G—Core Neutronics, Thermal
Hydraulics, and Safety Analysis

F. G. Aquino	M. D. Kelly
D. J. Bender	H. R. Kleijn
D. C. Bullington	P. Magee
C. L. Cowan	W. G. Meinhardt
S. C. Crick	S. Oldberg
R. N. Evatt	L. N. Price
J. H. Field	R. Protsik
D. D. Freeman	R. S. Seymour
P. Greebler	J. D. Stephen
D. A. Greene	R. G. Stuart
C. L. Hillsman	R. C. Thompson
D. P. Hines	J. E. Turner
B. A. Hutchins	P. C. Vaughan
A. Hynsbedt	S. D. Weber

TASK L—Cladding Development

J. J. Albanes	T. A. Lauritzen
C. F. Barrett	A. Montes
L. Brown	R. P. Rhoades
H. J. Busboom	H. Saethre
F. A. Comprelli	W. K. Wagner
R. Erwin	A. Withop
R. H. Eymann	U. E. Wolff
J. P. Ferguson	D. L. Zimmerman
D. H. Frisby	

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