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FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

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LIST OF ABBREVIATIONS, SYMBOLS, AND ACRONYMS

Å	=	angstroms
Ann.	=	annealed
at %	=	atom percent
BCD	=	binary coded decimal
BEHAVE		fuel modeling code
BISYN	=	A two dimensional, diffusion-theory computer code based on spatial synthesis techniques
B.P.	=	boiling point
BPI	=	bits per inch (computer tape density)
BUBBLE	=	coolant boiling and channel voiding code, a subroutine of FREADM
cm	=	centimeters
C.W.	=	cold worked
CSEWG	=	Cross Section Evaluation Working Group
DEMON	=	diffusion theory flux solution computer code
EBR-II	=	Experimental Breeder Reactor II
ENDF/B-II	=	Evaluated Nuclear Data File/B, version II
ENDRUN-2	=	computer code for cross section processing version 2
ETOE-MC ²	=	computer codes for cross section processing
FCI	=	fuel-coolant interactions
FFTF	=	Fast Flux Test Facility
FORTTRAN	=	standard computer programming language
FREADM	=	Fast Reactor Excursion and Accident Dynamics Model
ft/sec	=	feet per second
FUMBLE	=	fuel management computer code for start-up cycles
h_{gap}	=	fuel-cladding gap conductivity

GEAP-10028-38

GETR	=	General Electric Test Reactor
gm	=	grams
i.d.	=	inside diameter
in./in./h	=	inches per inch per hour
I.P.	=	integrated power
K	=	thermal condition
Kcal/mole	=	kilocalories per mole
k_{eff}	=	effective neutron multiplication factor
keV	=	kilo electron volts
kW/ft	=	kilowatts per foot
ℓ	=	angular momentum states
L_{bf}	=	pounds force
L_{bm}	=	pounds mass
LMFBR	=	Liquid Metal Fast Breeder Reactor
MARS	=	disassembly calculation
MeV	=	million electron volts
MeV-GE	=	General Electric adjusted fluence for EBR-II ⁽¹⁾
mg	=	milligram
μm	=	micrometers
msec	=	millisecond
mole	=	gram molecular weight
MWd/Te	=	megawatt-day per ton (metric, tonne, 2205 lb)
ν	=	number of neutrons emitted per fission
nvt	=	time integrated neutron flux
o.d.	=	outside diameter
% Δk	=	percent reactivity addition
%TD	=	percent of theoretical density of fuel

P_L	=	linear power
P_{max}	=	peak power
ppm	=	parts per million
psi	=	pounds per square inch
psia	=	pounds per square inch absolute
psig	=	pounds per square inch (gauge)
psia	=	pounds per square inch (absolute)
psig	=	pounds per square inch (gauge)
R	=	Roentgens, unit of radiation dosage
S/A	=	subassembly
SEFOR	=	Southwest Experimental Fast Oxide Reactor
SS	=	stainless steel
(ST)	=	solution treated
SUR	=	strong unbroken ring
σ_c	=	capture cross section
σ_f	=	fission cross section
SYN	=	combined synthesis-fuel cycle code system
TDOWN-2	=	computer code for cross section processing, version 2
TEM	=	transmission electron microscopy
THTD	=	transient heat transfer code (version D)
THTE	=	transient heat transfer code (version E)
TOFF	=	transient overpower fuel failure
TREAT	=	Transient Reactor Test Facility
VENUS	=	disassembly calculation code
VERA-11	=	critical assembly 11A in VERA facility
VNC	=	Vallecitos Nuclear Center

VOIDUM	=	systems hydraulics model, a subroutine of FREADM
vol %	=	volume percent
$\bar{\xi}$	=	average logarithmic energy decrement
W/cc	=	watts per cubic centimeter
wt %	=	weight percent
XPORT	=	a computer routine to prepare code for shipment to outside users
watts/cc	=	watts per cubic centimeter
wt %	=	weight percent
ZEBRA-3	=	critical assembly 3 in ZEBRA facility
ZPPR 2	=	critical assembly 2 in Zero Power Plutonium Reactor facility
ZPR3-11	=	critical assembly 11 in ZPR-3 facility
ZPR3-12	=	critical assembly 12 in ZPR-3 facility
ZPR3-48	=	critical assembly 48 in ZPR-3 facility
ZPR-6	=	Zero Power Reactor facility 6

HIGHLIGHTS

- A power spike during the irradiation of the first unencapsulated fuel pin experiment may explain the extent of the intergranular attack observed at the cladding inside surface.

Analysis of the irradiation history of the F3A experiment indicates that the presence of a high worth driver subassembly in an adjacent position during two EBR-II runs caused cladding temperatures substantially higher than expected.

- A high burnup EBR-II specimen was exposed to a transient in TREAT. This test resulted in substantial fuel melting in the highest burnup specimen tested to date.

The C4H high burnup specimen (irradiated in EBR-II to $>50,000$ MWd/Te) completed TREAT testing. Post-test neutron radiographs show no indication of gross failure, but capsule cover gas analysis indicates loss of cladding integrity.

- Out-of-pile thermal gradient experiments with fission product additives caused cladding reactions comparable with in-pile experiments.

Intergranular fission product cladding reaction, as evidenced by microprobe examination, showed the same relative degree of fission product attack and distribution as that observed in fuel pins from S/A F3A.

- Twenty-one critical quality assurance and quality control procedures required for forced convection capsule fabrication were completed and transmitted to AEC-RDT. The hold on capsule B9D was removed and fabrication was restarted.
- Modifications have been made to the Evaluated Nuclear Data File/B, Version II, which provide good comparisons between calculated and measured eigenvalues for fast benchmark criticals.

Calculation of the Doppler coefficient in SEFOR using ENDF/B-II, gave a value 15% higher than the measured value, but within experimental error.

- Both general reaction and intergranular attack at the cladding inside surface were detected during metallographic examination of fuel pins that reached $\sim 88,000$ MWd/Te in EBR-II.

Intergranular attack to a depth of ~ 4 mils was observed in annealed Type-316 stainless steel cladding which operated at 1140°F . This examination also revealed the presence of two distinct phases, believed to be concentrations of fission products, at the edge of the fuel around the central void.

- The greater restraint of fuel growth offered by 15-mil cladding thickness relative to 10-mil cladding is evident in thermal flux results.

A capsule with 10-mil cladding irradiated to $\sim 78,000$ MWd/Te in the GETR showed approximately 50% greater increase in diameter compared with a similar capsule with 15-mil cladding.

- Tests were successfully completed to establish the operational characteristics of the failure detection system to be used on forced convection capsule B9D.

A mock-up of the B9D failure detection system was operated for 257 hours. During the test, 44 samples were taken. Post-test examination of the system showed that no sodium or incipient sodium plugs were present in the sample line openings, the sample lines, or the filters.

- The primary loop transient hydraulics calculations in the FREADM code were reformulated to permit evaluation of accidents to either tank or piped primary systems.

The momentum and continuity relationships governing core and system thermal-hydraulics were rewritten to allow direct simulation of a tank or piped primary with pumps in either the hot or cold leg of the latter arrangement.

- Comparison of data from Type-316 stainless steel samples at two fluence levels indicates that swelling at fluences above 3×10^{22} is primarily due to growth of existing voids rather than further nucleation.

Measurements on Type-316 stainless steel samples at 3×10^{22} and 7×10^{22} n/cm² E > 0.1 MeV-GE revealed a mean void diameter increase from 365 to 640 Å, while void number density doubled (2.1×10^{14} to 4.4×10^{14} /cm³) with the resulting increase in swelling from 0.7 to 8.7%.

- A study was completed to determine the relative pressure drop effects of grid versus wire-wrap spacers.

An economic comparison of wire-wrapped and grid-spaced fuel assemblies in a 1000-MWe LMFBR shows an economic incentive of ~0.1 mill/kWh for the low pressure drop grid spacer relative to the normal wire wrap design, exclusive of possible burnup incentives.

ABSTRACT

This report is the thirty-eighth in a series of quarterly progress reports published under the Fast Ceramic Reactor Development Program. Current activities and technical progress are reported on a number of continuing analytical and experimental tasks directed toward the development of liquid metal cooled fast breeder reactors. Progress during the period of February 1971 through April 1971 is described for tasks related to defected fuel testing, safety experimentation, fuel materials properties, fuel testing in EBR-II, core design analysis, reactor safety analysis, reactor safety analysis, cladding development, physics, and fuel element design analysis.

1. INTRODUCTION

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

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

Progress during the period of February through April 1971 on the currently active tasks of this program is described in subsequent sections.

This is the thirty-eighth 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 Appendix A.

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2. SUMMARY

2.1 TASK B—DEFECTED FUEL TESTING

Preparation of 21 critical quality control and quality assurance procedures was completed and copies were transmitted to AEC-RDT. On the basis of the completion of the QA/QC procedures, the fabrication hold imposed on capsule B9D was removed and work was resumed. An audit of the fabrication facilities and implementation of QA/QC procedures required for disassembly and inspection of parts assembled prior to the work stoppage was carried out prior to restarting work on the capsule. The capsule failure detection system was successfully tested. No sodium or incipient sodium plugs were present in the sample line openings, the sample lines, or the filters.

Physics analysis of the flux filter for capsule B10A was continued.

Radial heat pipe development work was continued. Tests indicate that use of the heat pipe to remove heat quickly from the fuel test section of a forced convection capsule to prevent damage to the fuel pins is feasible.

Transducer development work for capsule B9B was completed. Two transducers were irradiated to a total gamma dose of 1.9×10^8 R. No decalibration was observed.

A study was completed to identify major design problems and peripheral equipment required to carry out in-core forced convection capsule irradiations in the GETR, using unirradiated and preirradiated fuel pins.

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

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

TREAT irradiation of the C4H fuel specimen was completed. This is the highest burnup ($\sim 50,000$ MWd/Te in EBR-II) mixed-oxide fuel pin to attain fuel melting during TREAT irradiation (~ 50 vol %). Mild cladding failure occurred, with no external fuel movement visible.

Remote sodium filling and final assembly operations were completed on capsule C4G, but an open circuit in the heater assembly occurred during capsule heatup at TREAT and has required partial disassembly and rework.

Fabrication has been initiated for capsules C4E, F, and M. C4M will be a FFTF fuel pin supplied by HEDL and preirradiated to $\sim 50,000$ MWd/Te in EBR-II.

Continued analysis of the C4X test flowmeter output indicates cyclic sodium voiding occurred, and that flowmeter sensitivity is better than 1 in./sec.

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

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

Fabrication of component parts for the mock-up IB tests has been initiated. Fabrication of the C10X capsule is estimated to be 95% complete, and operation is scheduled for early May. The Design Basis Specification for capsule C10A has been completed and is undergoing Quality Assurance review. All of the approximately 150 subroutines in THTE have been compiled and errors have been corrected. The THTD to THTE data translation program was completed and checked out.

2.4 TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

The fission product-cladding reactions developed in out-of-pile experiments are in close agreement with the reactions observed in in-pile experiments. Microprobe examination of these out-of-pile samples indicates that Cs, Te, and O_2 react with the cladding grain boundaries in a manner similar to irradiated sample F3B-3. In similar investigations testing the relative resistance of several cladding alloys to Cs_2O attack, Type-321 stainless steel resisted attack, while Types 316 and 347 reacted.

Capsule E2R, designed for investigating rate of central hole formation, was inserted into GETR for its initial exposure. No central hole was found after a 1-minute irradiation.

Microprobe examination of cladding exhibiting intergranular attack indicates that molybdenum and oxygen penetrate most deeply along the grain boundaries; cesium and tellurium migrate to a lesser depth.

The current status of Task E irradiations is shown in Table 2-3.

2.5 TASK F—FAST FLUX TESTS AND SUPPORT

Metallographic examination of six encapsulated fuel pins from group F3B at $\sim 88,000$ MWd/Te indicated both general reaction and intergranular attack at the cladding inside surface. Intergranular attack to a depth of ~ 4 mils was observed in the peak temperature zone ($1140^\circ F$) of the annealed 316 SS cladding of pin E1H-26. In addition, two

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Table 2-1
TASK B-IRRADIATIONS

Capsule	Objective	Design Parameters							Operating Conditions				Status	
		Fuel			Composition				Cladding Material	Irradiation Facility	Cladding Surface Temp (°F)	Peak Rating (kW/ft)		Peak Burnup (MWd/Te)
		Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in-diam)	Density (Smear) (% TD)	PuO ₂ (%)	O/M							
B8B (2 pins)	Fuel sodium compatibility Sodium bonded fuel central temperature versus exposure time	0 216	2	0 010	80	25	2 00	347	V Raft GETR	800	14	14 500	Examination completed	
		0 216	2	0 010	83	25	1 93	347						
B8C (2 pins)	Fuel sodium compatibility versus burnup Fission gas release versus burnup Fuel sodium-clad compatibility	0 144	2 2	0 006	86	25	1 97	316	V Raft GETR	1100	21	113 500 57 000	Examination completed	
		0 204	2 25	0 017 *	84	25	1 97	316						
B8D	Fuel sodium compatibility versus density, stoichiometry and Pu content	0 214	1 75	0 0065	80 88	40	1 97	316	V Raft GETR	950	17 5	41 000	Examination completed	
		0 214	1 75	0 0065	80 88	25	1 97					43 500		
		0 214	1 75	0 0065	80 88	25	1 94					41 400		
B9A (2 pins axially aligned)	Defected fuel pin Fuel swelling versus stoichiometry in flowing sodium	0 216	5	0 005	89	25	1 99	316	Special V Raft GETR	1019 971	23 4	8 600	Examination completed	
		0 216	5	0 005	89	25	1 95					8 600		
E1J B1	Thermal expansion clad defect mechanism	0 221	5	0 0007	94	25	2 00	316	Trail Cable	835	17 5	50	Examination completed	
E1J B2	Thermal expansion clad defect mechanism	0 221	5	0 0007	92	25	2 00	316	Trail Cable	860	19 0	300	Examination completed	
B9C (2 pins)	Swelling behavior of defected fuel pins in flowing sodium	0 216	4 25	0 004	89 6	25	1 97	316	Special V Raft GETR	1015	20 0	~7300 **	Test eliminated	
		0 216	4 25	0 004	89 6	25	1 97	20" CW 316 Annealed		1030	20 0	~7300 **		
B9D (2 pins)	Two defected fuel pins Fuel swelling in clean sodium (< 5 ppm Oxygen)	0 215	4 25	0 005	89 5	25	1 97	316 20" CW	Special V Raft GETR	1100 1060	15 0	~5500 *	Fabrication	

* Large gap required to accommodate sampling device
Planned irradiation

CLAP 10068 38

Table 2 2
TASK C 1—EXPERIMENTAL TESTS

Design Parameters

Capsule	Objective	Diameter (inch)	Fuel Length (inch)	Fuel Density (% TD)		Fuel Composition			Cladding Material
				Pellet	Smear	%PuO ₂	%U-235	O/M	
C4A	Transient failure threshold for zero burnup, mixed oxide fuel as a function of axial restraint density and form (pellet versus powder)	0 2159	14 23	93 8	90 4	19 4	0 57	1 983	316SS
C4B		0 2155	14 21	94 5	91 2	19 4	0 57	1 983	316SS
C4C		0 2164	14 19	83 1	80 4	20 9	0 91	1 983	316SS
C4D		0 2154	14 22	83 7	80 1	20 9	0 91	1 983	316SS
C4I		powder	13 48	powder	83 5	20 2	0 89		316SS
C4J		powder	13 45	powder	83 5	20 2	0 89		316SS
C4E	Comparison with C4A and C4B test results (high-density pellet fuel)	0 2159	13 46	83 2	80 5	28	92 7	1 974	316SS
C4F		0 2160	13 51	84	80 5	28	92 8	1 975	316SS
C4G	Comparison with C4A and B to determine effect of burnup on failure threshold (transient magnitude will be varied)	0 2154	13 50	94 4	90 5	28	89 3	1 973	316SS
C4H		0 2152	13 45	94 4	90 4	28	89 3	1 976	316SS
C4K	Comparison with C4I and J to determine effect of burnup on failure threshold of powder fuel (transient magnitude will be varied)	0 2205	13 50	Powder	84 1	28	93 0	1 972	316SS
C4L		0 2206	13 47	Powder	84 2	28	93 0	1 972	316SS
C4M	Failure threshold for low power (~10 kW/ft) high burnup fuel	0 215/ 0 217	13 5	84/94	80/90	30/20	~93 0	1 97/ 1 98	316SS
C4X	Proof test of redesigned C4 TREAT capsule (remote assembly techniques and TREAT characteristics) to be used with C4E—H and K—M	0 2186	13 48	96 2	91 0	0	92 8	2 000	316SS
C5A	Mobility of transient melted fuel, investigate potential for fuel relocation under prototypical conditions of length, restraint, and fission gas	0 214	24	95	91 5	0	12	2 000	316SS
C5B		0 214	24	95	91	0	12	2 000	316SS
C6A 1	Transient performance of defected or sodium bonded fuel, relative performance limits and failure mechanisms of pellet and powder fuel under transient irradiation conditions Sodium logged powder fuel is to maximize fuel sodium heat transfer (transient magnitude will be varied)	0 212	24	86	80	0	10 0	2 000	316SS
C6A 2		0 2203	24	powder	80	0	10 0	2 000	316SS
C6A 3		0 212	24	86	80	0	10 0	2 000	316SS
C6B 1		0 212	24	86	80	0	10 0	2 000	316SS
C6B 2		0 2203	24	powder	80	0	10 0	2 000	316SS
C6B 3		0 212	24	86	80	0	10 0	2 000	316SS

NP Not Planned

P Planned

I Initiated

NC Nearly Completed

C Completed

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

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

†† Actual fuel pin has yet to be chosen

Table 2-2
TASK C-1—EXPERIMENTAL TESTS

		Status							
				TREAT		Examinations [†]			
				Transient		Nondestructive			
		Irradiation							
Capsule	Fabrication	Pre-irradiation	Calibration	Test	Before Transient	After Transient	Destructive	Report	
C4A								GEAP-13562	
C4B									
C4C									
C4D									
C4I									
C4J									
C4E	C	C*	P	P	C	P	P		
C4F	C	C*	P	P	C	P	P		
C4G	C	C*	P	P	C	P	P		
C4H	C	C*	C	C	C	I	P		
C4K	C	C*	P	P	C	P	P		
C4L	C	C*	P	P	C	P	P		
C4M	††	P	P	P	P	P	P		
C4X	C	NP	C	C	C	C	I		
C5A		**						GEAP-13543	
C5B		**							
C6A-1	C	NP	C	C	C	C	C		
C6A-2	C	NP	C	C	C	C	C		
C6A-3	C	NP	C	C	C	C	C		
C6B-1	C	NP	C	C	C	C	C		
C6B-2	C	NP	C	C	C	C	C		
C6B-3	C	NP	C	C	C	C	C		

Table 2 3

TASK E-IRRADIATIONS

	Capsule	Objective	Design Parameters								Operating Conditions			Status	
			Diameter (inch)	Length (inch)	Fuel		Composition		Cladding Material	Irradiation Facility	Cladding Surface Temp (° F)	Peak Power (kW/ft)	Burnup (MWd/Te)		
					Density (% TD)		PuO ² (%)	O/M							
					Pellet	Smear									
26	E1E	Compare integral conduc- tivity at high burnup	0 218	1 9	94 7	94 5	20	2 00	347	MTR	875	20 0	85 000	Report in Publication	
	E1F		0 218	1 9	94 6	94 8	20	2 00		MTR	654/875	20/28	77 000		
	E1J 1	Effect of restructuring and stoichiometry on integral conductivity to melting	{	0 217	5 0	92 0	90 5	25	1 94	316	GETR	1100	23 0	2 000	Report in Preparation
	E1J 2			0 217	5 0	92 0	90 5	25	1 94	316	GETR	1100	23 0	2 000	
	E1J 3			0 217	5 0	92 0	90 5	25	1 94	316	GETR	1100	23 0	2 000	
	E1J 4			0 217	5 0	92 0	90 5	25	1 97	316	GETR	1100	23 0	2 000	
	E1J 5			0 217	5 0	92 0	90 5	25	1 97	316	GETR	1100	24 0	2 000	
	E1J 6			0 217	5 0	92 0	90 5	25	1 97	316	GETR	1100	23 0	2 000	
	E1J 7			0 217	5 0	92 0	90 5	25	2 00	316	GETR	1100	24 0	2 000	
	E1J 8			0 217	5 0	92 0	90 5	25	2 00	316	GETR	1100	23 0	2 000	
	E1J 9	0 217	5 0	92 0	90 5	25	2 00	316	GETR	1100	23 0	2 000			
	E2C 1	High burnup reference fuel radiated at near reference thermal conditions	0 218	5 0	84 5	82 2	25	2 00	347	GETR	1060	20 0	121 000	Examination	
	E2C 2		0 218	5 0	94 3	91 4	25	2 00	347	GETR	1060	20 0	113 000	Complete	
	E2E	Fuel swelling Four pins at three power levels	{	0 100	3 0	98 5	95 4	25	2 00	304	MTR	785	11 6	199 000	Report in Publication
	E2F			0 100	3 0	96 9	95 1	25	2 00	304	MTR	655	9 4	172,000	
	E2G			0 100	3 0	97 0	95 2	25	2 00	304	MTR	560	7 9	255 000	
	E2H			0 100	3 0	96 6	94 8	25	2 00	304	MTR	785	11 6	330 000	
	E2J 1	High burnup Annular versus solid pellet	0 218	5 0	90 4	89 5	25	1 98	316	GETR	1100	21 0	124 000	Examination	
E2J 2	0 218		5 0	96 0	88 5	25	1 98	316	GETR	1100	21 0	125 000	Complete		
E2R	Center void formation rate at lower burnup	0 216	10 0	94 0	91 0	25	1 98	316	GETR	1000	18 6	~0	Irradiation in Progress		
E5A 1	Powder versus pellet at low burnup	0 218	4 9	86 8	84 3	25	2 00	347	GETR	1100	27 0	2 000	Examination		
E5A 2		0 218	4 9	84 1	84 1	25	2 00	347	GETR	1100	27 0	2 000	Complete		
E5B 1	Powder versus pellet at high burnup	0 218	4 9	85 8	84 1	25	2 00	347	GETR	1400	20 0	138 000	Examination		
E5B 2		0 218	4 9	84 1	84 1	25	2 00	347	GETR	1400	20 0	138 000	Complete		

distinct phases, believed to be fission products, were observed at the edge of the fuel around the central void in pin E1H-30.

Interim examination of two capsules with 10- and 15-mil cladding thickness irradiated in a thermal flux to $\sim 78,000$ MWd/Te indicated that significantly greater restraint of fuel growth is achieved with the thicker cladding.

Gamma scans of five encapsulated fuel pins from the group F5 experiment at $\sim 55,000$ MWd/Te showed peaks at the top of the fuel column of pin F50, and below the fuel column at the interface between the insulator pellet and the end plug of pin F5A. This indicates gross movement of some fission products within the pins.

Analysis indicates that cladding temperatures in the F3A experiment (unencapsulated fuel pins to $\sim 50,000$ MWd/Te) were substantially higher than expected during part of the irradiation. The perturbation was produced by the presence of a high-worth driver subassembly in an adjacent position for two EBR -II runs. This temperature spike may explain the extent of intergranular attack observed at the inside surface of the cladding.

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

2.6 TASK G-6—CORE DESIGN ANALYSIS

A reference, optimized, grid-spaced, FFTF driver fuel assembly was selected and approved by RDT. A memo describing the technical basis for the selection was completed, and was reviewed by RDT. The preliminary design for the FFTF grid-spaced diagnostic assembly F9F for irradiation testing in EBR was initiated.

2.7 TASK G-10—SAFETY METHODS AND ANALYSIS

All major subroutine packages for FREADM-2 are near completion. These include.

- TOFF - Transient overpower fuel failure model
- VOIDUM - Updated primary loop hydraulics calculation
- BUBBLE - Improved sodium boiling and channel voiding model

In addition, disassembly calculation packages obtained from other sources are operational and ready for coupling to FREADM. These include:

- MARS - Updated version of R-Z modified Bethe-Tait (already coupled)
- VENUS - Lagrange hydrodynamics in R-Z geometry

2.8 TASK G-12—ANL SAFETY/CORE DESIGN GUIDANCE

Priority development needs were compared with specific ANL programs, and preliminary recommendations were discussed with cognizant ANL personnel. A GE engineer will be assigned to work at ANL to provide improved liaison.

2.9 TASK L—CLADDING DEVELOPMENT

Transmission electron microscopy (TEM) observations on the F2 series of fuel pin cladding indicate that Type 347 contains a greater number of smaller voids than Type 316 after similar irradiation conditions. The higher void density in Type 347 is attributed to the extensive intragranular precipitates in the alloy (3.4×10^{22} n/cm², $E > 0.1$ MeV-GE).

Comparison of data for Type 316 irradiated to fluences of 3.1 and 7.1×10^{22} n/cm², $E > 0.1$ MeV-GE at 525 to 545°C (978 to 1013°F) indicates that void nucleation rates are decreasing and that the increased swelling in the higher fluence specimen is due primarily to void growth.

Immersion density measurements of samples from the L-2 series capsules enable a comparison to be made of the swelling of various austenitic alloys. Peak swelling occurs at 500 to 520°C (932 to 968°F) for Types 316 and 347, and at 460 to 480°C (860 to 896°F) for Incoloy-800 and Type 304. The observed peak swelling for Incoloy-800 is about twice that for Type 316.

Biaxial creep tests are being performed on Type-304 stainless steel samples (capsule tubes) after fluence to 2.4×10^{22} n/cm², $E > 0.1$ MeV-GE. A total of 12 specimens have failed after testing up to 2000 hours. None of the specimens showed greater than 1% strain to failure. Instantaneous loss of pressure has not been observed. Failure is caused by gradual loss of pressure, apparently through microcracks forming a leak path through the tube wall. High fluence specimens ($\sim 7 \times 10^{22}$ n/cm², $E > 0.1$ MeV-GE) are now on test at 482°C (900°F). The rate and extent of swelling annealing at 593°C (1100°F) and 704°C (1300°F) are being evaluated before attempting strain measurements on specimens with high $\Delta V/V$ values.

Uniaxial tensile tests on specimens of Incoloy-800 and Types-304, -316, -321 and -347 stainless steels after fluence of up to 6.8×10^{22} n/cm², $E > 0.1$ MeV-GE are in progress. Initial results on these specimens [tested at 704°C (1300°F)] indicate ductility values ranging from 0.1 to 1%.

Table 2 4

TASK F FAST FLUX IRRADIATIONS *

	Objective	Total Number of Pins in Group	Date In	Projected Date Out	Actual Date Out	Number of Pins In S/A	S/A	Position
F0	Low power swelling rate of high density fuel at various burnup levels	5	07/16/65 07/16/65 07/16/65 09/14/70 09/14/70	02/27/72 01/30/73	08/15/68 08/08/70 08/08/70	1 2 2 2 2	XG02 XG03 XG04 XG03A XG04A	7A1 7D1 7B1 8D2 8A1
F1	Short time performance and proof test	6	05/06/65		05/23/65	6	XG01	4F2
F2	Medium power scoping irradiation. Major parameters include smear density stoichiometry cladding coolant interactions and burnup	21	09/03/65 09/03/65 02/23/69	05/18/70	03/06/68 02/20/67 05/18/70	9 12 4	XG05 XG06 X050	4C2 4E2 4C2
F3A	Bare pin irradiation to evaluate fuel cladding and cladding coolant interaction	16	08/15/68 10/04/69		06/30/69 05/18/70	16 16	X040 X040A	5B2 5B2
F3B	Density and stoichiometry effects for encapsulated fuel pins	18	11/21/67		05/18/70			
F4	Low density fuel behavior at high heat ratings	9	05/09/66 08/15/68 01/10/70		06/28/67 12/24/68 08/08/70	9 7 7	X011 X039 X070	2F1 3C2 3E1
F5	Fuel performance at peak for cladding temperatures	19	05/28/69		08/08/70	19	X064	4F2
F6	Fuel cladding interactions under accelerated swelling	4	03/24/66		04/24/69	4	X010	7F3
F7	Premirradiated axially restrained fuel prior to transient test. Main variable is fuel density	6	12/23/67 01/23/69		12/23/67 05/18/70	2 4	X015 X050	4A2 4C2
F8A	Effect of density on fuel swelling rate	16	01/13/67 01/13/67 05/24/70	07/01/72	08/02/69 01/31/70	7 9 9	X019 X020 X081	6D2 6B5 7F6
F8B	Effect of void deployment on fuel cladding reaction	19	07/25/68 02/24/71 02/24/71	09/21/75 03/24/74	08/24/70	19 9 19	X036 X117 X118	7E1 8B2 8D7
F9A	High power high burnup unencapsulated fuel pin irradiations	37	02/20/69 10/27/70	06/11/72	08/18/70	37 37	X043 X043A	4D2 6D3
F9B	Medium power high burnup unencapsulated fuel pin irradiations	37	05/23/69	10/29/72		37	X062	7F3
F9C	High power high burnup unencapsulated fuel pin irradiations	37	04/03/69 10/27/70	07/03/72	07/16/70	37 37	X056 X056A	5C2 6B2
F9D	Medium power, high burnup unencapsulated fuel pin irradiations	37	04/24/69	03/19/73		37	X058	7B6
F9E	High power high burnup, unencapsulated fuel pin irradiations in a grid spacer subassembly	37						
F10A	Cladding performance at peak cladding temperatures	19						
F10B	Cladding performance at peak cladding temperatures	19						
F10C	Fuel pin performance at peak cladding temperatures	19						
F11A	Cladding performance at peak FCR cladding temperatures	37						
F20	Irradiation to Peak Operating Limits	19						

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

1 Values are the calculated axial average powers. Ratio of peak to average power is 1.125

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

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

Table 2-4 (Continued)

TASK F FAST FLUX IRRADIATIONS

Group	Range of Average Pin Powers ⁽¹⁾ kW ±t	Target Burnup		Ave Calc Burnup ⁽²⁾		Peak Calc Fluence ⁽³⁾ (inv t X 10 ⁻²)		Status
		At %	MWd/Tp	At %	MWd/Tp	Total	E 0.1 Mev	
F0	66	3.91	35000	3.91	29831	0.276	0.199	Post exam complete Reconstituted into XG03A Reconstituted into XG04A In reactor
	5.39 5.81	8.95	80000	6.85	61505	0.595	0.433	
	5.35 6.14	11.18	100000	7.62	68102	0.662	0.458	
	5.57 6.18	8.95	80000	7.33	65512	0.634	0.462	
	5.41 6.14	11.18	100000	8.06	72048	0.704	0.517	
F1	12.1 14.31	0.22	2000	0.21	1859	0.017	0.015	Reported in Geap 5570
F2	11.28 13.74	8.39	75000	6.69	59800	0.569	0.501	Post exam complete Reported in Geap 13549 Post exam in progress
	8.34 14.65	5.59	50000	5.02	44897	0.422	0.362	
	10.50 12.58	11.19	100000	11.75	105054	1.034	0.887	
F3A	17.59 14.32	5.62	50000	2.88	25601	0.236	0.201	Reconstituted into X040A Post exam in progress
	17.25 13.91	5.62	50000	5.00	45977	0.430	0.366	
F3B	11.19 12.49	13.49	120000	8.86	8821	0.754	0.647	Post exam in progress
F4	17.57 14.30	5.65	50000	4.64	42755	0.298	0.256	Reconstituted into X039 Reconstituted into X070 Interim exam
	17.17 13.55	9.04	80000	4.85	42963	0.400	0.347	
	17.14 13.00	9.04	80000	8.25	8078	0.695	0.596	
F5	17.26 13.05	11.23	100000	5.71	508.4	0.476	0.408	Removed from reactor
F6	5.45 9.17	5.59	50000	5.00	51014	0.480	0.394	Post exam complete
F7	17.27 12.32	5.64	50000	5.40	47879	0.456	0.390	Terminated Terminated
	17.17 11.94	5.63	50000	5.43	48277	0.449	0.384	
F8A	5.36 8.95	5.62	50000	4.99	44161	0.412	0.42	Post exam complete Reconstituted into X081 In reactor
	8.20 8.91	11.25	100000	6.15	54614	0.524	0.438	
	8.08 8.03	11.25	100000	8.3	69562	0.677	0.556	
F8B	4.41 6.27	11.24	100000	4.73	42026	0.401	0.307	Removed from reactor In reactor In reactor
	4.46 5.55	11.24	100000	4.46	39614	0.379	0.279	
	4.06 6.22	5.62	50000	4.82	42861	0.409	0.306	
F9A	7.8 13.03	1.25	00000	5.57	49503	0.464	0.397	Reconstituted into X043A In reactor
	10.67 12.64	11.25	100000	6.10	54259	0.510	0.43	
F9B	9.9 10.55	11.24	100000	5.72	50881	0.477	0.407	In reactor
F9C	10.26 11.27	11.24	100000	5.67	50457	0.472	0.402	Reconstituted into X056A In reactor
	10.00 12.05	11.24	100000	6.15	54705	0.514	0.437	
F9D	6.94 9.61	11.24	100000	5.46	48556	0.459	0.376	In reactor
F9E								Fabrication in progress
F10A								Shipped to EBR II
F10B								Shipped to EBR II
F10C								Awaiting approval in principle
F11A								Fabrication complete
F20								Approval in principle received

Table 2-5
SUMMARY OF TASK F THERMAL FLUX IRRADIATIONS

Capsule Number		Fuel Design Parameters*				Operating Conditions**				Status
		Diameter (inch)	Length (inch)	Density (% TD)		O/M	Cladding Surface Temp (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)	
F12A (E5F)	Linear fuel movement versus burnup	0.214	10.0	Pellet 92.0	Smear 87.1	1.98	850	17.0	44,000	Report in Process
F12C (E5C 1)	Powder, pellet, and annular fuel comparison at high burnup	0.215	16.0	87.5	82.9	1.95	1160	19.5	137,000	Report in process
F12D (E5D 1)		Powder	16.0	81.8	81.8	1.97	1160	19.5	141,000	
F12E (E5E 1)		0.215 (Annular)	16.0	95.0	83.3	1.96	1160	19.5	130,000	
F12P (E2P)	Effect of radial restraint on fuel swelling (10 mil and 15 mil clad thickness)	0.219	2.0	92.0	91.0	1.98	1160	15.0	~85,000	Continuing irradiation
F12Q (E2Q)		0.229	2.0	92.0	91.0	1.98	1160	15.0	~85,000	

* In all capsules the cladding is Type 316 stainless steel and the fuel is (U 0.25 Pu)O₂

** The irradiation is conducted in the GETR

Stress-rupture testing of unirradiated defected tubing of 10 and 20% cold-worked Type-316H stainless steel has shown that defects as deep as 10% of the wall do not degrade or otherwise alter the mechanical properties of thin-wall tubes tested at 704°C (1300°F) with hoop stresses as high as 13,000 psi.

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

2.10 TASK N—CORE PHYSICS ANALYSIS

Modifications to the nuclear data in the Evaluated Nuclear Data File/B, Version II, have improved the agreement between calculated and measured eigenvalues and reaction rate ratios. A 29-group file has been generated from the modified data to provide a basis for reliable LMFBR design calculations. Preliminary tests with this file give $k_{eff} = 1.002$ for ZPR3, Assembly 48.

The ANL critical experiment programs for ZPPR and ZPR-6 were reviewed with ANL personnel. The primary modification requested by General Electric for the evaluation of physics design quantities is the addition of in-pin foil irradiations at more positions.

A calculation of the isothermal Doppler coefficient in SEFOR, using ENDF/B-II data, gave 0.0093 for $(-T dk/dT)$. This is 15% higher than the measurement, but within experimental error. It is expected that the data modifications to ENDF/B-II will reduce the value.

The fuel management code FUMBLE, for analyzing startup fuel cycles, was sent to the Argonne Code Center, and a topical report describing it was issued (GEAP-13599).

The input card formats subcommittee of the *de facto* committee on Computer Code Coordination has specified a standard "free form" type format.

2.11 TASK P—FUEL ELEMENT DESIGN ANALYSIS

A mechanism for producing cladding strain during a mid-life power increase is now modeled in BEHAVE—I. Inclusion of this capability has resulted in good agreement between the computed and measured axial diameter profiles for fuel pin F0E. Recent materials test data are shown to be in support of the cumulative strain to failure criterion used in STATUS.

Table 2-6
TASK L—CAPSULES IN EBR II

Group	Objective	Total Pins	Number of Specimens		Date In	Date Out	Row S/A	Core Position	Reactor Exposure (MWd)	Target Fluence (n/cm ² X 10 ²³)	S/A Average Fluences*** (n/cm ² X 10 ²³)		Status
			Tube	Flat						Total	Total	0.1 MeV	
L 2'	PI* Mech Prop Eval of Type 304, 316L 321 347 & Incoloy 800	5	15	80	09/03/65	02/20/67	4 XG06	4E2	9,317	0.4	0.33	0.29	Reported in Geap 10066
L 4	PI Mech Prop Eval of Type 316	2	10	20	03/29/66	11/14/66	4 X009	4A2	5,355	0.2	0.19	0.16	Reported in GEAP 10066
L 4'	PI Mech Prop Eval of Type 304, 321, 347, & Incoloy 800	5	25	50	07/17/66	04/10/67	2 X014	2D1	3,674	0.2	0.16	0.14	Reported in GEAP 10066
L 10	PI Mech Prop Eval of Type 304, 316, 321, 347 & Incoloy 800	10	70	60	01/13/67	09/00/68	4 X016	4D3 & 4F3	7,894	0.2	0.28	0.24	Reported in RMU Memo
L 14	PI Mech Prop Eval of Type 304, 316, 321, 347 & Incoloy 800	19	133	114	10/10/67	12/00/68	4 X025	4E2	8,399	0.2	0.29	0.25	Post Exam Complete
L 21	PI Mech Prop Eval of Type 321	1	3	16	09/03/65 08/15/68	03/06/68 12/24/68	4 X605 3 X039	4C2 3C2	12,632 2,532	0.75	0.54	0.46	Post Exam in Progress
L 2	PI Mech Prop Eval of Type 304, 316, 347 and Incoloy 800	4	12	64	09/03/65 02/23/69	03/06/68 05/18/70	4 X605 4 X050	4C2 4C2	12,632 11,706	0.75	0.90	0.75	Post Exam in Progress
MT 3 [†]	PI Mech Prop Eval of Incoloy 800, Inconel 625 and Hastelloy X	4	None	48	12/13/65	06/26/69	4 XA08	4F2	19,280	1.0	0.64	0.55	
L 6	PI Mech Prop Eval of Type 304, 316, 321 347 and Incoloy 800	3	63	126	12/06/66 4/23/69 10/02/69	01/02/69 06/26/69 09/10/70	2 X018 4 X018A 4 X018B	2B1 4E2 4E2	10,687 2,334 9,851	2.0	0.78	0.67	At FEF EBR II
L 15	PI Mech Prop Eval of Type 304, 316 321 347 & Incoloy 800	4	None	288	08/15/68 08/24/70	12/24/68	3 X039 4 X080	3C2 6E5	2,532 2,525	0.3	0.17	0.12	In Reactor
MT 4 [†]	PI Mech Prop Eval of Incoloy 800, Inconel 625 and Hastelloy X	1	None	48	12/13/65 08/24/70	06/26/69	4 XA08 6 X080	4F2 6E5	19,280 2,525	2.0	0.72	0.60	In Reactor
L 16A	Meas γ heating rates for L 16 Type Capsule Design	19					4	4C2		<0.01			Fabrication in Progress
L 16B	Meas Swelling in Type 316 and 304 Annealed and CW	19	76	None	11/16/70	01/00/76**	4 X098	4C2		0.3	0.02		In Reactor
L 16C	Meas Temp Dependence of Swelling in Clad Alloys	7		None									Design in Progress
L 16D	Reencapsulation of previously irradiated specimens	3	36	None							0.90 ^{††}	0.75 ^{††}	Design in Progress

* Post Irradiation

** Projected date out assuming EBR II continues to operate at 62.5 MW with 50% Load Factor

*** Fluence calculated as described in GEAP 10028-37

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

†† Fluence prior to reencapsulation

3. TASK B—DEFECTED FUEL BEHAVIOR

Task Leader: C. N. Craig*

3.1 GENERAL

The technical and economic feasibility of fast reactors depends on a thorough knowledge of the behavior of defected fuel and its consequences to reactor design and safety. Defective fuel elements must be expected in any reactor, even despite extensive preventive measures. Moreover, to reach the full economic potential of the LMFBR, high plant availability must be achieved. It must, therefore, be experimentally demonstrated that an LMFBR can continue to operate safely with a substantial number of failed fuel pins.

The specific purpose of Task B is to (a) determine experimentally whether failure of a fuel pin under normal or near-normal LMFBR operating conditions will result in propagation of failure to adjacent pins, (b) provide experimental and analytical data to permit selection of fuel design parameters and operating conditions which limit the consequences of fuel failure, (c) provide experimental data on failure kinetics to permit modeling and prediction of fuel failure behavior for the projected range of LMFBR operating conditions, and (d) identify the fuel and fission product species distribution in the reactor coolant and cover gas as a function of the failure mode.

3.2 DEFECTED FUEL TESTING

3.2.1 Forced Convection Capsule B9B

Capsule B9B is designed to investigate the behavior of failed fuel at higher burnups. The capsule will contain one fuel pin which will be defected at power after an exposure of approximately 25,000 MWd/Te.

Design work on capsule B9B was interrupted temporarily to provide increased effort for preparation and implementation of quality assurance and quality control procedures required for capsule B9D.

3.2.2 Forced-Convection Capsule B9C

Capsule B9C is designed to investigate the performance of defected hypostoichiometric mixed-oxide fuel operating at typical LMFBR conditions in flowing sodium. During final checkout of the capsule, a failure in the instrumentation system was detected, forcing cancellation of the experiment.

Disposition of capsule B9C was initiated in an attempt to salvage the main capsule components. Major components considered salvageable are the fuel pin, flowmeter, and pump helix. The detailed scope of work for disassembly and salvage was completed, and effort will be initiated as soon as nuclear safety requirements are established and the necessary procedures prepared and approved. The work to be completed during disassembly is:

- Freeze capsule sodium.
- Test sodium for alpha contamination.
- Section capsule.
- Melt sodium from salvageable components.
- Dispose of remaining capsule parts.

Should alpha-contamination be detected in the capsule sodium, this would indicate failure of the fuel pin. The capsule would then be disposed of without further disassembly.

3.2.3 Forced-Convection Capsule B9D

Capsule B9D is designed to investigate the performance of fuel pins defected in high-temperature (1050°F) flowing sodium after several days of full-power operation.

Efforts continued on the preparation and implementation of new and improved quality assurance and quality control procedures to be followed during fabrication, irradiation, and examination of capsule B9D. Preparation of 21 critical quality assurance and quality control procedures was completed, and copies were transmitted to AEC-RDT. Review and approval of 23 additional procedures required during fabrication and irradiation of the capsule are presently in progress.

On the basis of the completion of the 21 critical QA/QC procedures, fabrication of parts for the capsule sodium fill system, jigs, and fixtures was initiated. Preparation of quality report forms, quality control planning, and inspection forms for all parts in the capsule was completed.

* Refer to Appendix B for listing of contributors.

Each of the nine basic weld designs used in the B9D capsule was qualified to new or revised and updated procedures. Two welders will be qualified for each weld design. Qualification of the new brazing technique, using an infrared heat source, is nearing completion.

Audits were conducted of the fabrication facilities and for implementation of QA/QC and compliance with the recommendations of the October RDT QA audit. During fabrication of the capsule, similar scheduled and unscheduled audits will be conducted. A detailed plan to be followed during the disassembly and inspection of the partially assembled capsule, was reviewed and approved. Disassembly and inspection of the parts was initiated.

A development engineer and quality assurance engineer have been assigned to the fabrication site to monitor operations during fabrication of the capsule.

A full-scale mock-up of the failure detection system (FDS) was tested. The test apparatus has been described previously.¹ In the test, the sodium was maintained at 1000°F for 257 hours. During this period, 44 cover gas samples of ~90 cc each were drawn through filters and lines which duplicated those to be incorporated in the capsule. Subsequent examination of the filters and lines showed no evidence of plugging.

3.2.4 Forced-Convection Capsule B10A

Capsule B10A is to be the first in-core, multiple-pin, forced-convection capsule. The purpose of the capsule is to investigate pin-to-pin failure propagation for pins operating at typical LMFBR operating conditions.

Physics analysis for the B10A capsule flux filter was continued. The analysis has indicated the need for two filters: one integral with the reactor* filler piece, and the other in the capsule. The filter in the capsule is required to absorb those neutrons that are thermalized as they pass through the cooling water annulus surrounding the capsule. Preliminary calculations based on a design with a 0.020-in.-thick cadmium filter in the capsule indicate that the power difference between the center and outer pins in a seven-pin array can be matched within approximately 2 kW/ft. Additional calculations will be made to evaluate other filter materials and extensions of the analysis from a one-dimensional radial array to a two-dimensional analysis. These calculations will increase the accuracy with which the power distributions can be estimated in the seven-pin array.

3.3 TECHNICAL SUPPORT FOR FORCED-CONVECTION CAPSULE TESTS

3.3.1 Heat Pipe Development

Development and testing of a radial heat pipe was continued for use as a "thermal switch" to prevent overheating the fuel pin in the event of pump shutdown. A preliminary series of tests to determine wettability of three different wick designs in sodium was completed. Two of the wicks were fabricated from stainless steel screen, 100 and 200 mesh, respectively. The third design uses a porous sintered metal coating. The bottom of each wick was immersed in 850°F sodium to a depth of 1-1/2 in. Complete wetting of the 200-mesh and sintered metal wicks occurred after approximately 1 hour.

A radial heat pipe was fabricated with a 200-mesh stainless steel mesh wick containing 6 cc of lithium as the working fluid. The heat pipe was evacuated prior to welding shut. Thermocouples were attached in pairs at three different axial positions to both inside and outside walls. A rod heater was placed in the center of the heat pipe and bonded to the heat pipe with copper powder. The temperature of the heat pipe was increased by raising the heat flux at the surface of the wick to 0.77 cal/cm²-sec. Figure 3-1 shows how the temperature difference between the evaporator (inside) and condenser (outside) walls changed with evaporator temperature and the heat flux across the wick. Testing is continuing.

3.3.2 Pressure Transducer Testing

Three semiconductor strain gages** were purchased for use in forced convection capsule B9B. These transducers will be used to measure pressure within the fuel pin during operation and after defecting of the fuel pin.

The three transducers were calibrated upon receipt from the vendor. Two of the transducers were then exposed to 1.9×10^8 R gamma (exposure expected during B9B experiment) in the Vallecitos Laboratory cobalt pit under ambient conditions. All three of the transducers were then recalibrated. The transducers, both before and after irradiation, were calibrated at 150, 350, and 500°F up to their full-scale reading of 250 psi. No change in response of the transducers relative to their pre-irradiation calibration was measured. Pre- and post-irradiation calibration values are plotted in Figure 3-2. The vendor's calibration curve is shown for comparison.

* General Electric Test Reactor (GETR)

** Tycho Bytrex Model HG-2501

Figure 3-1. Performance Curve for Heat Pipe

3.3.3 Technical Feasibility Study

A study was made to identify major design problems and peripheral equipment required to carry out in-core, forced-convection capsule irradiations in the GETR using unirradiated and pre-irradiated fuel pins. Significant results indicated by the study are summarized as follows:

- Capsule length is a critical design parameter. Length of the capsule is limited by the reactor crane height.
- Space limitations may require removal of the capsule during each refueling of the reactor. If the capsule must be removed during refueling, either the annular linear induction pump (ALIP) or the helical induction pump (HIP) type of pump may be used. Other considerations may dictate pump type used. The ALIP requires a smaller sodium inventory; however, considerable development of this type of pump may be required.
- A new cask or extensive modifications of a present cask are required. Shielding provided by the cask presently used is not sufficient for a multipin capsule.
- Fabrication of multipin forced-convection capsules using pre-irradiated fuel pins is not feasible unless the requirement to maintain the sodium within the capsule molten after filling can be waived, or a procedure for filling the capsule at the reactor can be developed.

Figure 3-2. Calibration Data for Transducer No. 26906 at 350°F

4. SAFETY EXPERIMENTATION (TRANSIENT OVERPOWER)

Task Leader: G. R. Thomas*

4.1 GENERAL

The purpose of Task C-1 is to evaluate fuel failure thresholds and mechanisms in transient overpower accident situations in an analytical and experimental program of safety testing. This information is required for the timely development of the demonstration plant, FFTF, and the commercial LMFBR. Specifically, these investigations will include study of the transient behavior of oxide fuel as a function of geometry, density, burnup, linear power history, fuel form, and sodium ingress (defected and/or sodium bonded). In addition, tests will be conducted to determine the nature of fuel movement in the molten state and if such movement is controllable, and to observe the effect of a prototypic range of fuel and blanket lengths and the effect of flowing sodium on both failure thresholds and post-failure events.

4.2 HIGH-BURNUP FUEL EXPERIMENTS

This subtask investigates the transient overpower failure thresholds and subsequent fuel-sodium interactions for pre-irradiated fuel specimens of varying fuel smeared density (80 to 90%), fuel form (pellet and powder), and prior irradiation history. Current specimens allocated for these tests have accumulated ~ 50,000 MWd/Te fast-flux burnup in the EBR-II reactor. Included in this inventory are six GE pins irradiated at 13 to 15 kW/ft, and two HEDL-supplied FFTF pins (probably PNL-2-14 and 15), which operated at a lower power level of ~10 kW/ft.

4.2.1 Capsule Fabrication and Assembly

Capsule C4G was remotely sodium-filled, assembled into a TREAT pressure vessel, and shipped to the TREAT site. Subsequently, the capsule was returned to the Vallecitos Center for correction of an open circuit in the internal-heater system, encountered midway through capsule heatup (preliminary to TREAT irradiation). A structural failure in a spring clip, designed to maintain positive electrical contact between the pressure vessel head and heater assembly, had resulted in the open heater circuit. Repairs are nearly complete and this capsule is scheduled for TREAT irradiation early next quarter.

Capsule component fabrication and minor design modifications have been initiated for the C4E and C4F (low-density pellet) fuel specimens and the C4M fuel specimen (first HEDL-supplied FFTF fuel pin). The interim examination and de-encapsulation of the C4E and -F fuel specimens (removal from their EBR-II capsules) will be completed early next quarter.

4.2.2 TREAT Irradiation

TREAT irradiation of the C4H capsule was completed. This is the highest burnup (~ 50,000 MWd/Te in EBR-II) oxide fuel pin to attain fuel melting during transient irradiation. The actual TREAT irradiation parameters were very close to those requested, and the predicted amount of fuel melting (fractional amount above solidus temperature) is ~ 50 vol %. Post-test neutron radiographs show virtually no indication of failure, but analysis of capsule cover gas indicates failure did occur. The capsule is presently at VNC for further nondestructive tests and destructive examination.

4.2.3 Post-Irradiation Analysis

Continued analysis of flowmeter output from the C4X test indicates that in addition to the detection of the cyclic formation and collapse of sodium vapor bubbles, the flowmeter is capable of measuring flow induced by thermal expansion of the components in the fueled region of the capsule (flow velocities less than about 1 in./sec).

One of the principal objectives of flowmeter output analysis for the Series 4 tests is to obtain a time-dependent voiding history of the originally stagnant sodium in the fueled region of the capsule. The C4X transient flowmeter output provided an "unperturbed" sodium voiding history for such a study. Specifically since the C4X fuel pin did not fail, sodium vapor was the only nonliquid component in the inner sodium region (no dispersed fuel or gas components).

* Refer to Appendix B for listing of contributors.

Integration of the flowmeter velocity trace (Figure 4-1) produced the total volume flow of sodium through the flowmeter (which measured mass weighted velocity), shown in Figure 4-2. The flow oscillations between ~ 3.3 to ~ 4.2 sec shown in Figure 4-1 appear in Figure 4-2a as volume perturbations in the same time interval. They represent the *primary evidence of sodium boiling*¹. The magnitude of the volume perturbations indicate that the maximum bubble volume was about 0.06 in. The shape of the oscillations in Figure 4-2a (and 4-2c) implies the existence of only one major bubble.

From Figure 4-2a, the oscillatory bubble volume was about one tenth of the magnitude of the total volume which had passed through the flowmeter at time boiling began (~ 3.3 sec), i.e., there was a substantial flow component not associated with sodium boiling. This additional flow was accounted for by a relatively simple model, describing the time-dependent thermal expansion of the fuel pin, the surrounding dummy fuel pin array, and the inner sodium (Figure 4-2b). As seen in Figure 4-2c, this model accounted for all extraneous volume up to initiation of boiling at ~ 3.3 sec. The necessary time dependent temperatures for the pins and sodium were obtained from the THTD heat transfer code² analysis which yielded the best fit to the measured thermocouple data for the transient. This fit is seen in Figures 4-3 and 4-4. Note the measured temperature perturbations which occurred at the time of boiling initiation (~ 3.3 sec).

Additional investigation of the net volume curve (Figure 4-2c) is planned. The non oscillatory volume contribution from ~ 3.4 to 3.8 sec could be attributed to several causes, e.g., a non-condensing sodium vapor bubble nucleus, the base volume of two or more bubbles growing in overlapping cycles, or error in thermal volume calculations (due to temperature perturbations as seen in Figures 4-3 and 4-4). This volume contribution could also be due to a diametral increase of the C4X cladding in the fueled region (a deformation of ~ 0.004 in. would account for the extraneous volume through ~ 3.8 sec). If this were the case, the flowmeter output for the time span of ~ 3.4 to 3.8 sec would provide a direct indication of *time-dependent* cladding deformation. New light will be thrown on this possibility by fuel pin profilometry during the C4X destructive examination.

As seen in Figure 4-2c, the volume through the flowmeter continues to rise after 3.8 sec. This may imply the development of a longer termed natural convection flow in the capsule (flow velocities of 1 to 2 in./sec). The capsule design would permit such a low velocity circulatory sodium flow, but sufficient data were not available on this experiment to confirm it (premature shutdown of TREAT data recording equipment). Additional data were collected on the ensuing C4H test transient, and this natural convection possibility will be investigated.

4.3 COMPARATIVE TESTS

The primary purpose of these tests is to evaluate relative fuel pin performance as a function of such parameters as prior irradiation history, fuel length, and possible differences between UO_2 and UO_2PuO_2 during identical transient overpower conditions.

A topical report is in process describing the results of tests comparing the transient performance of defected or sodium filled fuel pins.

A preliminary design study has been completed to investigate the possibility of comparative tests using three capsules with pre-irradiated fuel side by side in one TREAT vessel. A memo concerning the value and feasibility of such tests is being prepared for transmittal to RDT.

4.4 MARK II LOOP TESTS

These tests will evaluate fuel performance under flowing sodium conditions and in multiple pin configurations.

Work has continued on thermal hydraulic calculations and Mark II loop design layouts for evaluation of TREAT transient test parameters. A first draft of a comprehensive design package describing the Series 8 tests in TREAT has been completed and is in review.

Figure 4-1. C4X Test Transient Measured Sodium Flow

Figure 4-2a. Sodium Volume Through Flowmeter

Figure 4-2b. Sodium Volume through Flowmeter Due to Thermal Expansion (Calculated)

Figure 4-2c. Corrected Sodium Volume through Flowmeter

Figure 4-3a. Capsule C4X, Second Test Transient Results, Sodium Temperature Histories (Calculated versus Measured) (Expanded Time Scale)

Figure 4-3b. Capsule C4X, Second Test Transient Results, Sodium Temperature Histories (Calculated versus Measured)

Figure 4-4a. Capsule C4X, Second Test Transient Results, Sodium Temperature Histories (Calculated versus Measured) (Expanded Time Scale)

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

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

Task Leader: H. C. Pfefferlen*

5.1 GENERAL

A major safety related area that is of concern to both FFTF and the LMFBR programs is the potential for rapid reduction or loss of coolant flow to localized portions of the core. Such coolant starvation might be a result of local blockages, fuel distortion, debris, or failure of the pumps. The fuel response to such starvation, the propagation of malfunctions to neighboring pins, and the time characteristics of signals (such as temperature and flow) are a primary concern in the design of the fuel bundles and instrumentation system. Firm data in these areas must be available to provide reliable input to accident analyses for licensing purposes.

The effects of loss of flow have been extensively studied analytically, ranging from estimates of core voiding reactivity to the local heat transfer and dynamics of single channel boiling. However, very little experimental data are available on the local effects of transient undercooling and boiling in a confined channel. A need exists to

- Run experiments out of-pile to establish the basic parameters of bubble dynamics
- Develop theoretical models by which these results can be extended to the prediction of more realistic situations
- Run experiments in pile, with oxide fuel and flowing sodium to verify these models and assess any secondary events

The purpose of this task is to conduct a program of such tests, both in pile and out of pile, utilizing the Forced Convection Capsule (FCC) and/or the Mark II loop as the basic in pile vehicle.

5.2 MOCK-UP 1B

Mock up 1B will be used to investigate the initial stages of a loss of flow accident in an out of pile loop experiment utilizing the same facility as the Mock up 1A test. This investigation includes the flow transient as well as the initial stage of void formation. The test will be performed in a concentric loop³ which utilizes three replaceable, 24 inch-long electric heaters to simulate fuel rods.

Fabrication of component parts necessary for the operation of the Mock-up 1B experiment has been initiated. Capsule preparation, including cleaning and instrumentation checkout, has also been initiated.

Heater vendors have commented on the specification and drawings. These comments are being evaluated to determine their effect on heater quality and cost (heater delivery is expected in June). Planning for the heater control and instrumentation equipment has been initiated, and a cost estimate was prepared. A test schedule will be prepared as soon as the heater delivery date is firm.

5.3 CAPSULE C10X

Capsule C10X is an in-pile vehicle for the study of the latter stages of the loss-of-flow accident. This is a small, nonflowing capsule in which boiling will be initiated, and the subsequent interaction of sodium and fuel will be investigated. Results from this test will have intrinsic value, in addition to providing vital information for the design of later loss of-flow tests in the FCC and/or Mark II loop.

Fabrication of capsule C10X is approximately 95% complete. All NaK filling operations have been successfully completed and verified by x rays. The "Design Report and Data Package" on this capsule was completed and is undergoing final review by project, safety, and operations personnel. Irradiation is scheduled for early May to coincide with a GETR midcycle shutdown.

5.4 CAPSULE C10A

Capsule C10A is the first in a series of in-pile loss-of-flow tests utilizing the Forced Convection Capsule (FCC). The purpose of this series of tests is to investigate the complete loss-of-flow accident from the initial flow reduction, through voiding, and into the area of secondary events from sodium molten fuel interactions. In these tests, every effort will be made to include the influence of fuel, blanket, and plenum, as well as flow restrictions and boundary conditions.

The Design Basis Specification for capsule C10A has been completed and is currently undergoing Quality Assurance review.

* Refer to Appendix B for listing of contributors

5.5 MODEL DEVELOPMENT

The THTE program is being utilized to provide the multichannel heat transfer and fluid flow modeling capabilities required for the analysis of single- as well as multi-pin transient undercooling experiments. The THTE program is the fifth generation program in the THT series of heat transfer programs for the solution of complex, three-dimensional transient and steady-state heat transfer problems. The primary extensions available in the THTE program are the computation of multi-fluid-channel heat transfer and multichannel one-dimensional pressure drop for compressible or incompressible fluids.

Work on the checkout and updating of the THTE program continued. Thirty-five subroutines were compiled and all detected errors were corrected. This completes the compilation of all the approximately 150 subroutines in THTE. An option is being added to provide computation and printout of the fuel enthalpy distribution to simplify application of the program to the analysis of experimental conditions. The THTD to THTE data translation program was completed and checked out. This program will perform approximately 95% of the reformatting required to perform a THTE analysis with a THTD model.

5.6 LIAISON

Plans and results of this program are interchanged with those of other organizations concerned with safety. A meeting was held at Argonne on February 4-5, 1971, with ANL Safety Division personnel to gain a better understanding of the ongoing ANL safety programs. A joint RDT/HEDL/ANL/GE meeting was held at BRDO, Sunnyvale, on March 17, 1971, to discuss loss-of-flow test facilities. At this meeting, the capabilities of the TREAT Mark II/R Loops and the GE Forced Convection Capsules were compared.

6. TASK E—FUEL PROPERTIES AND PERFORMANCE EVALUATION

Task Leader: B. F. Rubin*

6.1 GENERAL

The purpose of Task E is to establish the physical, mechanical, and thermal properties of ceramic fuels and fuel elements. These properties are obtained by in- and ex-reactor testing and are utilized to model or predict the behavior of fuel elements under steady-state or failure conditions. The reactions of fuel and cladding with fission products and sodium are investigated with an electron probe microanalyzer, x-ray diffractometer, and by simulated out-of-pile testing. Methods of limiting these reactions are being explored. The migration of impurities, fission products, and fuel cations is being studied in irradiated samples.

6.2 FUEL-FISSION PRODUCT-CLADDING REACTIONS

Previous quarterly progress reports described this program which is intended to investigate volatile fission product migration and the combination and reaction of fission products with cladding over the temperature range of 500 to 1150°C.

6.2.1 Out-of-Pile Thermal Gradient Capsule Testing

6.2.1.1 Fission Product Additions

The heat treatment and elemental migration in the thermal gradient capsules containing mixed-oxide fuel, cesium oxide, and tellurium were described previously.⁴ Microprobe examination of the cladding section showed intergranular attack, with depletion of the grain boundaries in alloying elements. This is depicted in the specimen current image at the top left of Figure 6-1. The figure shows that cesium penetrated more deeply into the cladding than the other fission products. Oxygen also showed significant intergranular reaction, while tellurium remained confined to the fuel-cladding interface.

Figure 6-1 also shows that the fission product attack of cladding in F3B-3 (Figures 6-2 and 6-3) followed a similar pattern. These results indicate that out-of-pile capsule tests successfully produce fission product reactions typical of those observed in irradiated samples.

6.2.1.2 Fuel Cladding Modifications

A program of fuel and cladding modification has been initiated, which has the objective of developing a fuel element that is resistant to fission product attack. Six thermal gradient capsules have been prepared and heat-treated for 100 hours. Cs₂O was added to stoichiometric fuel in rods containing stabilized and unstabilized cladding alloys 316, 321, and 347 in several grain sizes. These are identified as Test Set A in Table 6-1. Another set, Test Set B, is designed to investigate the effects of cold work. This is in keeping with the findings of Warzee, et al.,⁵ who noted that cold working increased the corrosion resistance of Type-304 stainless steel in superheated steam at 400 to 600°C by the formation of a protective surface layer. An additional capsule will contain extremely low-carbon 304L material (0.008%) to determine whether sensitization, with chromium depletion around grain boundaries, is associated with intergranular fission product attack. Capsules 1 through 6 have been examined metallographically in the unetched condition. Type-321 stainless steel was not attacked at any temperature, while Type-316 stainless steel was corroded above ~900°C, and Type-347 above ~750°C. When the Cs₂O was added at the cold end rather than at the hot end of a Type-316 tube, the attack occurred at 575 to 600°C, instead of above ~900°C. These results represent the first relative comparison of candidate cladding alloys in a common environment.

6.3 ELECTRON MICROPROBE EXAMINATION OF IRRADIATED SPECIMENS

Microprobe examination of irradiated fuel and cladding specimens is being done to establish the nature and extent of cladding attack and the identification and distribution of fission products and other elements.

* Refer to Appendix B for listing of contributors.

Figure 6-1. Specimen Current and X-Ray Images of Chromium, Cesium, Oxygen, and Tellurium, Showing that Intergranular Attack During Out-of-Pile Tests Reproduces the In-Pile Phenomenon

PHOTOMICROGRAPH

NEGATIVE SPECIMEN CURRENT IMAGE

Te X-RAY IMAGE

Cs X-RAY IMAGE

O X-RAY IMAGE

Mo X-RAY IMAGE

500X

Figure 6-2. Photomicrograph, Specimen Current, and X-Ray Images of Molybdenum, Cesium, Tellurium, and Oxygen in Grain Boundaries of the Cladding of F3B-3

NEGATIVE SPECIMEN CURRENT

Mo X-RAY IMAGE

Cs X-RAY IMAGE

Te X-RAY IMAGE

250X

Figure 6-3. Specimen Current and X-Ray Images of Molybdenum, Cesium, and Tellurium in a Fuel-Cladding Gap Area of Transverse Sample F3B-3

Table 6-1
THERMAL GRADIENT CAPSULES
TESTING CLADDING MODIFICATIONS

	Capsule ^(a)	Cladding Alloy	Condition	Variation
Test Set A	1	321	Annealed	
	2	316	Annealed	ASTM 8 Grain Size
	3	347	Annealed	ASTM 8-1/2 Grain Size
	4	347	Annealed	ASTM 10 GrainSize
	5	316	Annealed	ASTM 5-1/2 Grain Size
	6	316	Annealed	Cs ₂ O at Cold End ^(b)
Test Set B	7	304	12% ColdWorked	
	8	304	20% Cold Worked	
	9	316	30% Cold Worked	
	10	316	50% Cold Worked	
	11	316	Annealed	Contains Extra Low Carbon (0.008%) Type 304 Inserts
	12	316	10% Cold Worked	

(a) All capsules loaded with (U-25 wt% Pu)O_{2.00} and Cs₂O and heat treated 100 hours in 550 to 1150°C thermal gradient.

(b) All other capsules contained Cs₂O at hot end.

6.3.1 Microprobe Examination of a Specimen from Pin F3B-3

The examination of a transverse sample 0.6 in. from the top of the fuel column in pin F3B-3, was completed. The sample was an annular pellet of (U - 25 wt % Pu)O_{2.00} irradiated to a measured burnup of 3.8 at.% at a calculated cladding temperature of 1050°F.* Microprobe measurements of the radial oxygen and xenon concentrations are shown in Figure 6-4. This figure also includes microstructural features of the pellet and indicates that the original dimension of the central hole was reduced 20 mils diametrically. No microstructural feature distinguishes the original diameter and no compositional interface is indicated in the microprobe traces. Vapor deposition is believed to account for the center hole closure of this pellet, which was located in the cooler region of the fuel rod. This sample also indicated an increase in plutonium concentration from 25 to about 40% Pu at the center, as shown in Figure 6-5. The fuel of greater than 25 wt % PuO₂ composition represents 10% of this pellet volume. The deposition of a plutonia-rich vapor is contrary to expectations for a fuel with an O/M ≥ 2.00. However, Rand and Markin⁶ have pointed out that in substoichiometric mixed oxide, the vapor phase would contain more plutonia than the original composition. Plutonia enrichment, to a much lower extent (increase from 20 to 26 wt % PuO₂) was determined in the microprobe trace of F2C located at approximately the same axial location.⁴ This pin was also stoichiometric, and operated to a burnup of about 5.3 at.% and at comparable linear powers.

* This pin was from S/A X040 which experienced an overpower condition for at least 11% of its irradiation history. Cladding temperatures in these periods for sample F3B-3 are estimated to have been 1200°F.

Figure 6-4. Photomicrograph of Specimen F3B-3 and Corresponding Radial Distributions of $Pu/(Pu+U)$, Oxygen, and Xenon

Figure 6-5. Plutonium Migration in Sample F3B-3 Plotted versus $(r/R)^2$ to Show Volume Relationship

Figure 6-2 illustrates the fission product distribution at a region where intergranular attack of the cladding took place. Oxygen and molybdenum penetrated deepest into the cladding. Cesium and tellurium were also present within the grain boundaries, but they were not the leading elements.

Figure 6-3 is an enlargement of a portion of the area shown in Figure 6-4. Molybdenum again seems to be a leading penetrant, while cesium appears primarily at the interface. Tellurium also showed slight penetration. It may be concluded that cesium, molybdenum, and oxygen are the prime fission products that react with cladding, although their specific or relative rates are not yet delineated.

Oxygen within the grain boundaries and across the cladding was also measured in this specimen from F3B-3. Figure 6-6 shows the relative oxygen intensity across the cladding obtained by point counting at 5-mil intervals.

Figure 6-7 shows x-ray rate-out traces which indicate a continuous scan of oxygen activity that can be related to the cladding microstructural features. The upper left-hand photograph is an oxygen trace across the entire cladding wall and shows the same oxygen concentration near the fuel interface as shown in Figure 6-6. The remaining three photographs include the traces over individual grain boundaries. These results show a significantly higher oxygen level in the grain boundaries than in the cladding matrix, indicative of oxide fission product(s).

6.4 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR

6.4.1 Capsule E2R—Rate of Center Hole Formation at Reactor Startup

The E2R capsule was inserted into the GETR trail cable facility for its first irradiation cycle on April 4, 1971. The capsule contained solid pellets of 88 and 94% TD mixed-oxide fuel. The purpose of this experiment is to investigate fuel restructuring (rate of center hole formation) at reactor startup rates of 1 kW/ft/min and center temperatures of 2500°C. The initial exposure was for 1 minute at full power which, assuming no center hole, required 18.3 kW/ft to attain that temperature. Thermocouple readings indicated that the capsule was inserted at the desired startup rate and that a peak power of 18.8 kW/ft was reached. Figure 6-8 shows the power-time history of this irradiation cycle. The capsule was neutrographed to determine the presence and size of a central hole. None was found, despite contrast enhancement to emphasize any hole. The capsule was reinserted for a 10-minute irradiation cycle at 18.4 kW/ft and the capsule will be re-neutrographed and examined for central hole formation.

6.5 TRITIUM DISTRIBUTION

Since tritium is a potential hazard in fuel reprocessing, attention is being directed to its concentration within irradiated fuel elements and its distribution between fuel, cladding, and fission gas. Samples of several fuel elements irradiated in EBR-II are being prepared for a study of tritium distribution within irradiated elements. Scoping analysis has indicated that no tritium was present in the fission gas sample of fuel pin F3B-1.

Figure 6-6. Oxygen Intensity Across Cladding of Sample F3B, with Higher Concentrations on Fuel Side

6.6 CLADDING TEMPERATURE DETERMINATION BY Kr^{85} RELEASE

Because fission product attack on the cladding is temperature dependent and may impose limitations on the operation of a reactor, a method for confirming calculated cladding temperatures is desirable. Swanson, Butler, and Clarke⁷ reported a method of determining the highest cladding temperature late in life. In this method, the cladding is heated to evolve krypton, which has entered the cladding by recoil and/or diffusion; the krypton release is monitored as the cladding is heated, and the temperature of release is considered the maximum late-in-life cladding temperature. Because of some uncertainty about the operating temperatures of the F3A test specimens (see Task F), a program utilizing this technique is being implemented.

Two alpha-cleaned cladding samples from F2V (fuel burnup of about 13 at. %) were analyzed for krypton. In these first experiments, no gas was detected. The lack of gas in these samples is believed to be caused by the removal of the inner 2 to 3 mils of cladding, which may be the area of major krypton concentration.* A modified procedure is planned in which the sample will be analyzed in RML and alpha-cleaning will not be mandatory.

6.7 THERMAL SIMULATION OF COLUMNAR GRAIN GROWTH

This program is intended to determine the temperatures at which microstructural changes such as columnar grain growth take place in mixed-oxide fuel. The results will provide temperature markers that will improve the interpretation of photomicrographs and the analysis of fuel thermal behavior. The temperature of columnar grain growth in mixed oxides of several stoichiometries will be determined. The furnace previously used for thermal-conductivity measurements is being utilized with a thermal gradient of $2500^{\circ}\text{C}/\text{cm}$.

6.7.1 UO_2 Testing

An initial test was run with UO_2 heated in the furnace for 1.5-, 5-, and 10-hour periods. Metallographic examination showed that columnar grains had formed in all three samples. Figure 6-9 illustrates the structure in the sample heated 10 hours. The large columnar grains and chevron-shaped voids are clearly evident, along with some fuel melting. Pellet surface temperatures were estimated at about 1200°C , extrapolating from the readings of black body holes, while the center temperature exceeded the melting point of about 2840°C . The thermal gradient was calculated

* The cladding samples were alpha-cleaned beforehand.

- a LINE OF THE X-RAY TRACES
- b BACKGROUND
- c X-RAY RATE OUT

Figure 6-7. Oxygen as Oxide Concentrated in Grain Boundaries of Fuel Pin F3B-3

to be $2300^{\circ}\text{C}/\text{cm}$. Columnar grains formed at about 2250°C in this 89% TD UO_2 pellet. A temperature of about 2100°C was reported for this microstructural change by Asamoto, et al.⁸ Short-term GETR Trail Cable tests of UO_2 pellets, irradiated 1 and 24 hours, indicated columnar grain growth temperatures between 2200 and 2370°C .⁹

6.7.2 Mixed Oxides

Stoichiometric pellets were heat-treated under the same conditions, but the tungsten heater rod experienced significant oxidation and vaporization, which compromised the findings. Additional pellets of two hypostoichiometric compositions will be tested in appropriate furnace atmospheres to determine temperatures of microstructural change as a function of stoichiometry.

Figure 6-8. Power-Time History for First Irradiation Cycle, 1 Minute at Power, Capsule E2R

6-11/6-12

GEAP-10028-38

Figure 6-9. UO₂ Heat Treated in a Steep Thermal Gradient to Develop Columnar Grains

7. TASK F—FAST FLUX TESTS AND SUPPORT

Task Leader: J. D. Stephen*

7.1 GENERAL

The purpose of Task F is to apply the more promising results of developments in basic fuel technology to the design, fabrication, and integral testing of fuel elements in a fast flux. Important fuel performance characteristics, such as fission gas release, fuel cladding interactions, fuel thermal performance and subassembly interactions, are determined under conditions simulating as closely as possible those expected in a fast ceramic reactor.

The test parameters for Task F are selected by a combination of design analyses which rely on basic properties data and on the results of pertinent experiments (such as those in Task E and Task L). Model analysis (under Task P) has been applied quantitatively to interpretation of the results, and to aid in the selection of ranges of parameters for further tests. The results of prior fast flux irradiation tests are interpreted to guide the design of future tests. The selection of test parameters is also influenced and guided by the work under Task B on defected fuel testing, Task C, which focuses on the preferred range of parameters for safety considerations of LMFBR designs, and Task G-6, which is concerned with subassembly design and performance. Demonstration of high fuel burnup is required over a range of power levels. Fuel parameters that are under study in a fast flux environment include density, configuration, fabrication process history, stoichiometry, void deployment, external restraint, and pin spacer concepts. Fast flux tests are performed with both encapsulated and unencapsulated pins.

Selected irradiation tests in thermal test reactors are being performed in support of the fast flux irradiations. These tests include high risk tests, instrumented capsules, capsules operated at high power with fuel temperatures at or above melting, and capsules designed to determine mixed oxide fuel swelling by measuring fuel pin diameters without the complications of cladding swelling.

A summary of the fast flux irradiation tests in the EBR II is given in Table 2-4. Calculated values for total fluence greater than 0.1 MeV, and fuel burnup values in MWd/Te and atom percent (taking into account fuel depletion), are incorporated into the summary table. The fluence, power, and burnup values given in Table 2-4 were calculated by using the fission rates provided in the latest revision¹⁰ of the "Experimenters' Guide for Irradiations in EBR-II" as a basis.

Irradiation tests in thermal reactors that are being performed under Task F are listed in Table 2-5.

7.2 SCOPING AND STATISTICAL CONFIDENCE IRRADIATIONS

7.2.1 Group F3B (Fuel Void Deployment and Clad Alloy Studies)

The destructive examination of 6 of the 18 F3B encapsulated fuel pins, which operated at ~ 15.5 kW/ft peak to ~ 88,000 MWd/Te peak, is continuing. The fabrication parameters, operating conditions and pin diameter increases are summarized in Table 7-1.

The gamma scan of the failed pin E1H-33 indicated an accumulation of fission products near the failure. The neutron radiograph showed many areas of high neutron cross section in the central void in the failure zone.

Profilometer measurements showed that the growth of pin E1H-30 was twice that of pin E1H-26. The fabrication and operating parameters of the two pins were similar, except for the cladding heat treatment. The 13% cold-worked Type-304 stainless steel cladding of pin E1H-30 received a carbide agglomeration heat treatment (see Table 7-1) after cold working to enhance the creep properties.

Transverse metallographic sections of all six pins have been examined. Both general reaction and intergranular attack were observed at the cladding inside surface. Intergranular attack to a depth of ~ 4 mils was found in the peak temperature zone of the annealed Type-316 stainless steel cladding of pin E1H-26. The Type-304 cladding of pin E1H-30, which received the carbide agglomeration heat treatment, exhibited a wide reaction layer on both specimens examined. In the area of peak flux, the depth of the general reaction was 2 mils maximum, and in the area of maximum cladding temperature the general reaction reached ~ 3 mils. There was intergranular attack in both areas, to a maximum depth of ~ 1 mil in the peak cladding temperature area.

Two distinct phases were observed at the edge of the fuel around the central void in pin E1H-30. The lighter of the two had a dendritic structure. Microprobe examination of similar phases observed in pin F2G of the F2 experiment¹ showed high concentrations of fission products.

* Refer to Appendix B for listing of contributors.

Table 7-1

SUMMARY OF PARAMETERS OF THE SIX FUEL PINS EXAMINED
FROM THE GROUP F3B IRRADIATION

	Fuel Pin Number					
	E1H-21	E1H-23	E1H-26	E1H-29	E1H-30	E1H-33
Fuel (U-0.25Pu)_{0.2}						
Stoichiometry, O/M	1.97	1.97	1.97	1.97	1.98	1.98
Fabrication Process	Co-ppt	Co-ppt	Phy. Mixed	Co-ppt	Co-ppt	Co ppt
Configuration	Powder	Annular Pellet *	Solid Pellet	Solid Pellet	Solid Pellet	Solid Pellet
Average Pellet Density, % T.D		96.4	89.9	95.2	96.5	95.7
Smeared Density, % T.D.	82.3	88.5	86.9	91.8	93.2	92.7
Diametal Gap Range, mils		2.9-4.9	2.4-4.5	3.6-4.4	3.1-4.6	2.6-4.0
Cladding						
Alloy	316SS	316SS	316SS	304SS	304SS	304SS
Fabrication Method	Welded, Drawn	Welded, Drawn	Welded, Drawn	Seamless	Seamless	Seamless
Thermal Treatment	Annealed **	Annealed **	Annealed **		Carbide Agg *** HT	Annealed
Mechanical Treatment				13% CW	13% CW	
Wall Thickness Range, mils	14.7 15.6	15.2 15.8	14.9 15.7	15.8 16.4	15.9 16.5	14.0 14.7
Irradiation Conditions						
Calculated Peak Burnup, Atom %	9.4	9.9	9.6	9.2	9.3	9.1
Calculated Peak Burnup, 10 ³ MWd/Te	83.9	87.7	85.5	81.7	82.5	81.3
Calculated Average Burnup, Atom %	8.5	8.9	8.6	8.2	8.3	8.2
Calculated Peak Linear Power, kW/ft	14.2	15.7	15.4	15.4	15.4	15.6
Calculated Peak Fluence, (>0.1 MeV) 10 ²² nvt	6.1	6.5	6.3	6.0	6.0	6.0
Calculated Peak Clad Temperature at fuel-clad interface	1105°F	1150°F	1140°F	1140°F	1140°F	1140°F
Dimensional Changes						
Maximum Fuel Pin Diameter Increase, mils (%)	2.2 (0.8)	3.3 (1.3)	6.5 † (2.6)	4.0 (1.6)	8.0 (3.2)	-
Fuel Pin Length Increase, inches	0.076	0.063	0.102	0.022	0.101	--
Fuel Column Length Increase, inches	-	0.195	0.044	0.311	0.374	0.232

* Initial diameter of central hole was 0.038-inch

** Solution annealed

*** Carbide agglomeration heat treatment: 24 hours at 900°F in He, 216 hours at 1300°F in He, rapid air cool after cold work

† The high value may be attributed to an area of smaller than typical fuel-cladding gap

7.2.2 Capsules F12P and F12Q (Effect of Radial Restraint on Fuel Swelling in a Thermal Flux)

An interim examination of the F12P Q experiment is completed. The two capsules have reached $\sim 78,000$ MWd/Te (8.8 at.%) in a thermal flux under near identical conditions except for cladding thickness. The maximum diameter increases of F12P (15-mil cladding thickness) and F12Q (10 mil cladding thickness) were 4.4 mils and 7.4 mils, respectively.

Figure 7-1 shows the average diameter increase and the rate of average diameter increase as a function of burnup. The greater restraint of fuel growth of the thicker cladding is evident. The rate of average diameter increase for 10-mil cladding is decreasing with burnup, while for 15-mil cladding, the rate appears to be increasing. Several possible mechanisms to explain this behavior are being investigated, including (a) strengthening the cladding by strain-hardening, and (b) utilizing the fuel region void to accommodate fuel growth.

The capsules have been reinserted in the GETR for further irradiation.

7.3 ELEVATED CLADDING TEMPERATURE IRRADIATIONS

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

Examination of five encapsulated fuel pins, which operated at ~ 16 kW/ft peak to $\sim 55,000$ MWd/Te (6.1 at.%), is in progress. Three of the fuel pins will be destructively examined. The other two capsules are to be returned to EBR-II to be reconstituted into a new subassembly with the fourteen remaining Group F5 capsules for continued irradiation toward a goal burnup of $> 100,000$ MWd/Te (11.25 at.%).

Neutron radiographs showed that none of the Group F5 fuel pins had failed. Profilometer measurements along the capsules showed a maximum diameter increase of ~ 2.3 mils. There was no apparent effect of the spiral spacer wires on the profiles.

Gross gamma scans of the five capsules revealed (a) a peak at the bottom end plug/insulator pellet interface of fuel pin F5A, and (b) a peak at the top of the fuel column of fuel pin F5Q. Incremental gamma scans showed the F5A peak to be Cs^{137} , Zr^{95} , and Nb^{95} , and the F5Q peak to be Cs^{137} and Nb^{95} . Single-channel scans of fuel pin F5Q for Cs^{134} , Cs^{137} , Nb^{95} and Zr^{95} showed the Cs^{137} peak, and also showed that Cs^{134} was distributed uniformly along the fuel pin from the bottom end plug to approximately the upper spacer above the spring.

7.3.2 Groups F10A, F10B (Cladding Performance at Peak FCR Cladding Temperatures)

The data packages for the F10A and F10B experiments were reviewed by the ANL/EBR-II Project, and approved. The encapsulated fuel pins were shipped to EBR-II and are ready for assembly into subassemblies X021 and X022.

7.4 GRID SPACER IRRADIATIONS

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

Fabrication of the fuel pins for the F9E experiment is complete. Fabrication of the subassembly was initiated this quarter. All grid spacer material has been received and characterized. The outer channel and noseplate were received from EBR-II, and the inner channel is expected shortly. Fabrication of the top end fixture, lower shield, and orifice plate is complete. Preparation of the data package was initiated.

7.5 IRRADIATIONS TO PEAK OPERATING LIMITS

Approval-in-principle was received from RDT for the F20-1 experiment. This subassembly of 19 encapsulated pins will operate in EBR-II for a short period with fuel center melting. The objectives of the experiment are to develop power-to-melting parameters and to study the effects of operation with fuel center melting. Both new and pre-irradiated fuel pins will be included. Fuel preparation and preparation of the fabrication specifications are under way.

7.6 UNENCAPSULATED FUEL PIN IRRADIATIONS

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

The F3A subassembly contained fuel pins F3B through F3X.

7.6.1.1 Burnup and Cladding Temperature

Cladding temperatures were substantially higher than expected for part of the irradiation of this subassembly, which may explain the intergranular attack observed at the inside surface of the cladding. This conclusion is based upon measured burnup values that have been correlated with distance from a high worth EBR-II driver subassembly (70%

Figure 7-1. Diameter Change in 10-mil and 15-mil Cladding after 10,000 MWd/Te

enriched versus the normal 54%) inserted in an adjacent channel during runs 34A and 38B. Consistently higher burnup was observed at shorter distances from the high worth subassembly

Analyses of fission product neodymium and heavy metal depletion were made on F3B-1, F3B-2, F3B 3, F3B-4, F3B-5, and F3B-7 to determine the actual burnup achieved by these pins. The results (Table 7 2) show good agreement between the Nd and heavy metal depletion values, with the heavy metal depletion results being consistently ~1 15% higher than the Nd results. The small difference between the two independent techniques is probably due to systematic errors in input parameters such as fission yields or relative fission rates for the higher atomic weight Pu isotopes.

The average discrepancy between the measured and the calculated burnup (11 9%) shown in Table 7 2 is lower than the discrepancy observed in previous irradiations (~14 4%). This change in burnup discrepancy indicates that the F3A pins either operated at a slightly higher power than expected throughout the irradiation, or operated at a substantially higher power for a short period. During runs 34A and 38B, a 70% enriched driver subassembly was inserted at location 4B1, which is immediately adjacent to the 5B2 location of the F3A pins. This high worth driver contained about 30% greater fissionable loading than the standard 54% enriched driver, and would be expected to increase the fission rates for some distance around location 4B1.

The burnup discrepancy is plotted in Figure 7 2 as a function of distance from the center of the high worth subassembly at position 4B1. There is a definite trend of higher burnup (hence higher power) occurring closer to the high worth driver.

Burnup measurements were taken at five locations along pin F3B-3 to verify the axial flux distributions predicted by the method suggested in the EBR II Guide for Experimenters¹⁰. The normalized results and results from previous experiments (Figure 7 3) correspond well with the predicted curve.

7.6.1.2 Fission Gas Release

The fission gas release data for pins F3B 1, F3B 2, F3B 3, F3B 4, F3B 5 and F3B 7 are given in Table 7-3. The fractions Xe and Kr released/generated for those pins are substantially higher than observed previously at similar power level and burnup. A sample of the gas from pin F3B 1 was analyzed for tritium. None was detected.

7.6.2 Group F9C (Unencapsulated Fuel Pin Irradiations to High Burnup)

At the request of RDT, the fuel pins F9C experiment will be committed to the EBR II Mark E program to obtain data on the consequences of failure of high burnup, mixed oxide fuel pins. Data packages for the reconstitution of these pins into two Mark E subassemblies are being prepared.

7.6.3 Group F11A (Unencapsulated Fuel Pin Performance at Peak FCR Cladding Temperatures)

Preparation of the data package for the F11A experiment continued.

Table 7-2

BURNUP MEASUREMENTS – GROUP F3A

Sample	Distance from Core Bottom (inches)	Calculated Local Burnup* (C) (At.%)	Measured Burnup (At.%)		Burnup Discrepancy (%)	
			Nd	HM **	$\left(\frac{C-Nd}{Nd}\right)$	$\left(\frac{HM-Nd}{Nd}\right)$
F3B-1H	6.6	5.341	4.707	4.653	13.47	-1.15
F3B-2E	4.6	5.456	4.905	4.847	11.23	-1.18
F3B-3C	0.5	4.668	4.231	4.183	10.33	-1.13
F3B-3E	3.7	5.538	4.961	4.904	11.63	-1.15
F3B-3H	7.4	5.693	5.157	5.098	10.39	-1.14
F3B-3J	10.6	5.023	4.520	4.469	11.13	-1.13
F3B-3N	12.8	4.915	3.829	3.786	28.36	-1.12
F3B-4D	6.6	5.480	4.898	4.841	11.88	-1.16
F3B-5D	6.6	5.381	4.798	4.742	12.15	-1.17
F3B-7E	7.2	5.505	4.810	4.755	14.45	-1.14
Average						-1.15 ± 0.03

*C = (Calculated Peak Burnup) X [(Local/Average) Ratio per Figure C-1 in EBR-II Guide(10)] ÷ (1.12).

**HM = Heavy Metal (U + Pu Depletion) Burnup Measurements.

Table 7-3

FISSION GAS RELEASE - GROUP F3A

Pin	Total (Xe+Kr) Released ^(a) cm ³ at STP	Total (Xe+Kr) Produced ^(b) cm ³ at STP	% (Xe+Kr) Released
F3B-1	75.1	101.2	74.2
F3B-2	103.2	104.2	99.0
F3B-3	82.4	107.9	76.3
F3B-4	76.3	103.7	73.6
F3B-5	82.0	95.8	85.6
F3B-7	95.9	104.9	91.4

(a) Based on gas chromatographic results

(b) Based on calculated average burnup per the EBR-II Guide for Experimenters⁽¹⁰⁾ assuming all U which is not U²³⁵ to be U²³⁸ and using 0.248 Xe+Kr atoms per fission

Figure 7-2. Burnup Discrepancy in X040A Fuel Pins as a Function of Distance from Center of High-Worth Subassembly at 4B1

Figure 7-3. Axial Fission Rate Profile—Comparison of Calculated and Experimental Values

8. TASK G-6—CORE DESIGN ANALYSIS

Task Leader: P. R. Pluta*

8.1 GENERAL

The primary purpose of Task G-6 is to develop a grid-spaced fuel assembly for the fast breeder reactor. Another purpose of this task is to contribute to the understanding of fuel performance, core design, and thermal hydraulic problem areas, and to provide guidance for effective planning and coordination of experimental programs.

The spacer concept offers potential advantage over the present wire wrap fuel assembly in

- Increased accommodation of differential structural swelling.
- Reduced core steel volume fraction
- More uniform fuel rod circumferential temperature distribution.
- Reduced pressure drop

The engineering analyses, out-of-pile mechanical and hydraulic testing, and supporting irradiation diagnostic test assemblies in EBR-II will result in an optimized grid-spaced driver fuel assembly and an LMFBR assembly being tested in the FFTF. This program, including prototypical fuel assembly tests in the FFTF, will establish the technical bases for the selection of the grid-spacer support system as the preferred system for the LMFBR.

8.2 FFTF GRID-SPACED DRIVER FUEL ASSEMBLY

The purpose of this subtask is to assess the performance of various 217 rod grid spacers for irradiation in FFTF, to select a reference grid design, and to qualify an optimized grid-spacer fuel assembly for irradiation test in FFTF.

The reference optimized grid-spaced FFTF driver fuel assembly selected was approved by RDT. A memo describing the technical basis for the selection of the reference FFTF grid-spaced driver fuel assembly design was completed and reviewed with RDT.

A study of bundle vibration characteristics was conducted to determine the minimum bundle size that adequately simulates full 217-rod assembly vibration. Since each rod vibrates individually almost independent of the number of grids and with very little feedback from other rods or grids, the 37-rod bundle is believed to constitute an adequate simulation for vibration, fretting, and wear in a hot sodium flow test to simulate local effects in a 217-rod bundle. A long-term, in-line, sodium flow test of a 37-rod grid-spaced bundle is being designed. A study to determine the test requirements for a full-scale, 217-rod, water flow test is now being conducted.

To assess the effect that the grid-positioning channel dimples or notches in the channel wall have on the strength of the channel, an analysis of the resulting stress concentration has been initiated. The dimple cross section and the finite element representation are shown in Figure 8-1. A finite element plane-strain analysis has been employed to determine the elastic stress concentration that the discontinuity will cause. The representation and the solution are limited to the section transverse to the axis of the duct for tensile loads only. This solution may be useful in the investigation of fatigue loading from cyclic pressure and thermal gradients through the thickness of the duct wall. The solution indicates an elastic stress concentration of ~ 2.5 times the stress in the undisturbed section. Material properties were chosen for Type-316 stainless steel at a temperature of about 1000°F.

Further analyses will include the longitudinal cross section of the dimple for loads in the axial direction of the duct, bending in the section, and an elastic-plastic analysis to determine strain concentrations. Investigations will be based on the geometry illustrated in Figure 8-1, and on a similar geometry with the external protuberance removed.

A preliminary study of the potential economic incentives for the use of grid spacers in commercial LMFBR plants was completed. Grid spacer advantages considered included lower steel fraction, reduced hot-spot temperatures, lower pressure drop, and increased burnup capability.

8.3 F9E GRID SPACER IRRADIATION TEST

The purpose of this subtask is to design and analyze a grid-spaced LMFBR diagnostic test assembly for irradiation in EBR-II (0.25-in. o.d. fuel rods at $P/D = 1.28$, high swelling stainless steel).

The design evaluation of the F9E diagnostic experiment was completed.

* Refer to Appendix B for listing of contributors

8.4 F9F GRID-SPACER IRRADIATION TEST

The purpose of this subtask is to design and analyze a grid-spaced FFTF diagnostic test assembly for irradiation in EBR-II.

A preliminary design of the F9F assembly has been initiated. The following design parameters are being evaluated:

- Rod-to-channel edge spacing to achieve prototypical rod bowing and grid-to-channel differential swelling
- Rod-to-grid contact loading at beginning of life
- Rod-to-grid self welding, wear, or galling
- Various grid beam thicknesses
- Backup grid spacer design

8.5 OUT-OF-PILE GRID SPACER MECHANICAL TESTS

The purpose of this subtask is to perform mechanical tests directed toward the overall evaluation of the grid spacer and its interaction with the fuel rods and the channel wall.

8.5.1 Thermal Stability Test

To determine the thermal stability of a wave beam and straight dimpled beam grid spacer, a thermal cycling test was conducted. Four welded grids—two of which were inserted in channel sections—and five grid beams were thermally cycled between 1100 to 1200°F and ambient for 50 cycles. Soaking times of 10 to 15 minutes at each temperature were sufficient to attain uniform temperature throughout the specimens. The major objective of the test was to assess the effect of thermal cycling on the shapes and dimensions of the grid beams and welded grids. A second test involved several grid beams and welded grids which were soaked at 1300°F for 1000 hours in a furnace.

After the tests were completed, the grids were inspected. No beam relaxation, grid distortion, or surface damage was observed.

8.6 FUEL ELEMENT FAILURE, COOLANT BOILING, AND OPERATING SAFETY LIMITS

The objective of this task is to make a comprehensive assessment of the information on the effects of fuel element failure, coolant boiling, and operating safety limits needed to design LMFBR fuel assemblies. During this quarter, a topical report on the work accomplished was completed, and final review was initiated.

Figure 8-1. Stress Contours in Channel Wall Adjacent to Grid-Spacer Dimple (Nominal Membrane Stress of 10,000 psi)

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9. TASK G-10—SAFETY METHODS AND ANALYSIS

Task Leader: D. D. Freeman*

9.1 GENERAL

The purpose of Task G 10 is to develop and use analysis techniques for evaluation of the safety of an LMFBR. Results will form a basis for establishing the design and safety criteria that define the range of reactor characteristics and test parameters upon which experimental programs must focus.

9.2 MODEL DEVELOPMENT

9.2.1 Fuel Failure

The purpose of this subtask is to develop computer models for failure of fuel pins and movement of fuel in transient overpower or loss of flow accidents.

The transient overpower fuel failure model (TOFF) has been extended from a single-node model to treat the full fuel pin. Calculations to be performed in TOFF II are summarized in Figure 9 1. The fuel expulsion calculation is currently being extended beyond the initial results reported for the last quarter.¹ Coupling to the FREADM failure triggering logic is summarized in Figure 9 2.

Checkout of the TOFF calculation will be facilitated by using the THTE heat transfer calculation to generate fuel temperatures and melt fractions as a function of time in a fuel pin geometry equivalent to that used in the FREADM code. Sample data have been generated for the following conditions, using the model shown in Figure 9 3.

- a. A steady-state temperature profile was determined for the following parameters
 - 15 kW/ft
 - 26 ft/sec (sodium velocity)
 - T_{inlet} 725°F
 - 30,000 Btu/h ft² °F heat transfer coefficient to the sodium
- b. Starting with the above steady state situation, the following overpower transient condition was imposed:

The resulting temperatures and material melt fraction, as a function of time, which THTE calculates, were saved on magnetic tape to serve as input for the TOFF program.

The release of fission gas from the plenum, down through the fuel and out a defect, has been modeled. The model presently analyzes the transient flow of gas from a plenum through a small-diameter pipe. The analysis assumes that the gas expands adiabatically and includes a friction loss term. Initial results indicate that the transient gas expansion in the fuel pin could be treated as a quasi steady-state process.

* Refer to Appendix B for listing of contributors

Figure 9-1. TOFF II Logic

Figure 9.2 FREAMD TOFF II Coupling Logic

9.2.2 Loop Hydraulics and Channel Voiding

The VOIDUM routine is a FORTRAN program for determining transient flow conditions throughout the primary coolant system under hypothetical LMFBR accident conditions. VOIDUM, a subroutine of FREAMD,¹¹ is being updated.

The routine was originally written for a tank type coolant system. It has been completely revised so that the dynamic response of a piped coolant system can also be simulated. Two types of piped systems are included, with the pumps located either in the hot leg or in the cold leg. Figure 9.4 is a schematic diagram of the tank-type system and the piped system with the pumps in the cold leg. Figure 9.5 shows the piped system with the pumps in the hot leg.

The modification to the routine is largely internal. The set of governing equations, for continuity and unsteady motion, forms a matrix of 54 X 54 as compared with the previous maximum size of 48 X 48. Of the total number of variables, 42 are acceleration terms and 12 are pressure terms—an increase of two acceleration terms and four pressure terms over the previous model. All 54 equations have now been updated. Terms that are not common to the three different types of systems are preceded by a multiplying factor which takes the value of either 1.0 or 0.0, depending on which type of system is being considered.

The solution to the matrix is obtained by using the same procedure as before. The matrix is reduced to a 12 X 12 pressure matrix formed by eliminating the acceleration terms. Crout's method is then used to solve for the pressures. The acceleration terms are obtained from back substitution, and the velocity and displacements are obtained through integration. Figure 9-6 is a flow chart of the entire routine.

The change on the input requirement has been kept to a minimum. The additional values needed in the NAMELIST input are as follows:

Geometry

AHXDP1	= Cross-sectional flow area of IHX1 discharge piping
AHXDP2	= Cross sectional flow area of IHX2 discharge piping
ZHXDP1	= Change of elevation of IHX discharge piping
XHXDP1	= Length of IHX discharge piping

Figure 9-3. THTE/TOFF Model

Figure 9-4. Schematic Diagram Showing Tank-Type or Piped Cooling System with Pumps in Cold Leg

Figure 9-6. Flow Chart of Routine VOIDUM

Control Variables

HLOOP	=	Flag to select piped coolant system with pumps in hot leg
CLOOP	=	Flag to select piped coolant system with pumps in cold leg

The additional variables that will be shown in the output of loop type coolant systems are

VDHDP1	=	Acceleration of coolant at IHX1 discharge pipe
VDHDP2	=	Acceleration of coolant at IHX2 discharge pipe
VHXDP1	=	Velocity of coolant at IHX1 discharge pipe
VHXDP2	=	Velocity of coolant at IHX2 discharge pipe
PHX01	=	Pressure at outlet of IHX1
PHX02	=	Pressure at outlet of IHX2
PUMIN1	=	Pressure at pump 1 inlet
PUMIN2	=	Pressure at pump 2 inlet

For example, if this routine is used to determine the transient pressure and/or velocity of the piped coolant system, the input would have

HLOOP = 1 0

in addition to all other input values as needed. The routine would correct the 54 governing equations for this system internally.

9.2.3 Probabilistic Modeling

The purpose of this study is to gain insight into the dominant safety question stemming from fuel failure propagation. The understanding of propagation mechanisms is to be expressed quantitatively by evaluating probability trees. This understanding forms the basis for evaluating the worth of proposed research on key safety questions.

This study was initiated with a state of the art review of fuel coolant interaction (FCI). Two distinct cases of FCI are of interest: fuel assembly meltdown due to loss of cooling and transient overpower accidents. Review of available literature and in house analysis indicates that pressurization due to FCI in loss of cooling accidents is theoretically much less a hazard than in transient overpower accidents. The sodium boiling that precedes meltdown on loss of cooling provides a very compressible sodium vapor environment and very limited quantities of liquid sodium available for rapid boiling on mixing with molten fuel. Short pressure pulses of only a few hundred psi are theoretically possible compared with longer pressurization of a few thousand psi under transient overpower conditions. Significant damage propagation due to FCI on local loss of cooling appears highly unlikely.

Based on the fuel failure propagation fault tree previously generated under Task G 6,^{3 12} a detailed probability tree was drawn, starting with a fuel pin failure and including the possible end state of core disassembly. Preliminary estimates of transition probabilities were made based on current understanding of propagation mechanisms. With the assumption of no safety system action, the probability of progressing to a core disassembly accident was 5×10^{-6} . Safety System reliability would reduce this probability by a factor of $\sim 10^{-6}$.

The probability of progressing to a core disassembly accident from an intermediate point involving a local flow blockage due to fuel debris was < 0.003 , assuming multiple failures of fault detection instrumentation and no control action. The probability tree for this progression is shown in Figure 9.7.

9.2.4 Core and Reactivity Modeling

The purpose of this subtask is to improve modeling of whole core LMFBR accidents.

Test cases have been prepared for the VENUS Lagrange hydrodynamics calculation. Energy release will be computed using equation of state information recently developed for UO_2 using Hirshfelder's method.¹³ Comparison will be made with modified Bethe-Tart calculations and with VENUS calculations using alternate equation of state data. A comparison of the equation of state using Hirshfelder's method for mixed oxide with Ackerman's¹⁴ low temperature data for UO_2 is shown in Figure 9.8.

Figure 9-7. Probability Analysis of Local Flow Blockage

Figure 9-8. Equation of State Comparisons

9.2.5 FREADM Update

The purpose of this subtask is to incorporate state of the art modeling of LMFBR accident events into the FREADM accident analysis system.

All major subroutine packages to be incorporated into FREADM-2 are near completion. These include updated VOIDUM, TOFF, and BUBBLE calculations.

9.2.6 Parametric Analysis

The purpose of this subtask is to provide analytical backup to model development and experiment design and analysis.

Transient bundle thermal-hydraulics calculations have been specified to compute the buildup of thermal gradients near channel walls in loss-of-flow accidents. Results of this work will provide a further indication of the importance of radial condensation in sodium boiling resulting from loss of flow.

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10. TASK G-12—ANL SAFETY/CORE DESIGN GUIDANCE

Task Leader: D. B. Sherer*

10.1 GENERAL

Task G-12 has the goal of providing improved liaison with development programs at Argonne National Laboratory, and providing design-oriented input for the development planning in order to strengthen support of the Demonstration Plant in safety and related areas. This task was initiated last quarter.

10.2 REVIEW OF CURRENT PROGRAMS

A review of several of the safety-related ANL programs has been made. This included a trip to ANL to discuss these areas of investigation in order to gain a better understanding of their scope, schedules, and the expected output from these programs. The following areas of ANL work were the subject of this review, and continue to be of primary interest to Task G-12:

- Coolant Dynamics
- Fuel Coolant Interactions
- Fuel Dynamics
- Fuel Element Failure Propagation
- Post-Accident Heat Removal
- Containment Program
- Core Structural Safety
- Accident Analysis and Safety Evaluation
- Instrumentation System to Protect Core Integrity
- High-Temperature Physical Properties and Equation of State of Reactor Materials
- Fuels and Materials Studies, including
 - In-Reactor Creep Studies
 - In-Pile Stainless Steel Swelling
 - Behavior of Reactor Materials
 - Oxide Fuel Studies
- Heat Transfer and Fluid Flow
- Instrumentation and Control

10.3 DEFINITION OF NEEDS

The needs of the GE Demonstration Plant for information on core design and safety development have also been reviewed. Comparisons of these needs with the expected output from the ANL programs have indicated several areas where further interaction appears profitable. Areas of particular interest are:

- Fuel Coolant Interaction
- Fuel Element Failure Propagation
- Fuel Dynamics
- Instrumentation and Control

The comparison of the expected outputs from the ANL efforts and the Demonstration Plant needs has led to certain preliminary conclusions whereby the ANL work would more directly support the Demonstration-Plant needs. These recommendations have been discussed with cognizant ANL personnel. Further study and interaction with ANL is necessary in order to establish firm recommendations.

10.4 TECHNICAL LIAISON

Arrangements have been initiated for assignment of an engineer to work at ANL. This will produce a direct involvement in the technical programs of interest, and will provide improved liaison between GE and ANL,

*Refer to Appendix B for listing of contributors.

11. TASK L—CLADDING DEVELOPMENT

Task Leader: H. J. Busboom*

11.1 GENERAL

The technical and economic feasibility of ceramic-fueled fast reactors depends on the attainment of high performance of core structural members and the fuel system, especially the survival of the cladding at temperatures and specific fuel pin powers of practical interest

The objectives of the cladding materials development program are

- Determine the performance characteristics—including mechanical properties, fracture characteristics, radiation damage, and structural stability—of the most likely cladding material candidates and core structural materials under test conditions simulating the LMFBR environment.
- Investigate the rate of alloy swelling and the environmental and metallurgical factors that influence and control it in LMFBR primary alloys, and the factors that are potential candidates.

11.2 MATERIALS IRRADIATION PERFORMANCE

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

11.2.1 Fluence Calculations for EBR-II Irradiations

The method for calculating neutron fluence values for materials irradiations in EBR-II¹ has been updated to include the core configuration for the 62.5-MW core. The revised EBR-II Users' Guide¹⁰ was used to determine normalized radial profiles for total fluence and fluence greater than 0.1 and 1 MeV. The profiles are similar to those obtained for the 50-MW core, with the exception that the profile for fluence greater than 1 MeV showed a sharper decrease between rows 5 and 7 in the 62.5-MW core. This information is being incorporated into the procedure used to calculate neutron fluence for any radial and axial position in EBR-II.

11.2.2 Series L-6, L-15, and MT-4—Materials Capsules

The three L-6 capsules (Mark-B7 type) have been discharged and are in the EBR-II Fuel Evaluation Facility hot cell. These capsules will be shipped to the Vallecitos Nuclear Center during the next quarter. The L-6 capsules have accumulated a total fluence of 8.3×10^{22} n/cm²-GE in S/A X018B.

The four L-15 capsules and the MT-4 capsule (Mark-A19 type) are in the EBR-II core, Row 6 (S/A X080). The total fluences accumulated by the L-15 and MT-4 series capsules to date are 1.7×10^{22} n/cm²-GE and 7.2×10^{22} n/cm²-GE, respectively.

11.2.3 Series L-16 Materials Capsules

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

11.2.3.1 L-16A Capsules

The 19 L-16A capsules are designed to accommodate various temperature monitoring devices and flux wires to measure the temperature of irradiation and neutron flux in a typical L-16 type Mark-A19 subassembly. Assembly of the capsule inserts is in progress.

11.2.3.2 L-16B Capsules

The 19 L-16B capsules are in the EBR-II core, row 4 (S/A X098). The initial irradiation exposure for this S/A began in March 1971 and is scheduled for a total fluence of 3×10^{22} n/cm²-GE.

*Refer to Appendix B for listing of contributors

11.2.3.3 L-16D Capsules (Materials Re-Irradiations)

These three capsules are being designed to accommodate highly irradiated ring specimens for continued irradiation to higher fluences. Selected cladding specimens will be taken from the high burnup Task F fuel pins and Task L materials capsules. Design of these capsules is continuing.

11.3 CLADDING ALLOY SWELLING

The objective of this program is to measure cladding alloy materials and subassembly hardware volume changes in available irradiated specimens to determine the relationships between swelling, fluence, irradiation temperature, alloy microstructure, and stress history. The techniques used to measure cladding swelling are liquid immersion density measurements, dimensional measurements (profilometry), and transmission electron microscopy (TEM).

11.3.1 L-2 Series Specimens

Immersion density measurements have been performed on specimens taken from the L-2 series test capsules after irradiation in EBR-II to a peak fluence of $7.4 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. The disassembly of these capsules was briefly described in the thirty-sixth quarterly report in this series.⁴

The L-2 series capsules were irradiated in row 4 of EBR-II. The four high-burnup capsules described here (L2A, L2C, L2E, and L2G) were initially in subassembly XG05 but were transferred to X050 when the reactor power was increased to 50 MW. The accumulated exposure has been calculated to be $7.4 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. The fluence values calculated for these capsules are 6 to 10% greater than estimated by the HEDL Dosimetry Center on the basis of run 31F data.

The L-2 capsules were designed for irradiation in the EBR-II type Mark-A19 subassembly. The outer capsule tubes were standard 0.375-in. o.d. X 0.020-in. wall Type 304 stainless steel supplied by Argonne National Laboratory. The capsule tube contained a train of three tubular specimens each 5-5/32 in. long X 0.250 in. o.d. X 0.015 in. wall (Figure 11-1). These were joined with "umbilical tubes" 1/8-in. o.d. X 0.020-in. wall and were similarly connected to a reservoir situated in the upper region of the capsule. The tubes were filled with sodium which was pressurized by helium contained in the reservoir. The annulus between the specimen train and the capsule was filled with argon. Each tubular specimen contained tensile specimens attached to a tungsten strip. The tungsten had the purpose of acting as a gamma heating element to raise the specimens to the capsule design temperature. Temperature sentinels, each containing three different melt alloys, were situated in the upper end of each tubular specimen. Two tubular specimens were located within the EBR-II core region, with the third positioned just above it. One-inch-long immersion density specimens were cut from the sentinel region of the in-core specimens and from the lower end of the above-core specimen, as shown in Figure 11-1.

All of the specimens in a particular capsule were of the same alloy. The materials tested were:

- Capsule L2A-Incoloy-800
- Capsule L2C—Type-316L stainless steel
- Capsule L2E—Type-347 stainless steel
- Capsule L2G—Type-304 stainless steel

The analyses of these materials are listed in Table 11-1, together with initial grain size. All of the alloys were purchased as-welded, and drawn tubing in the mill-annealed condition; they were not further heat-treated before irradiation.

The tubular specimens were designed to operate at 650°C (1200°F), with internal pressures equivalent to 80% of the stress rupture limits of the alloy in a 10,000-hour test at 650°C (1200°F). These conditions gave design hoop stresses of 8 to 10 ksi. The tensile coupons were designed to operate at the same temperature and provide post-irradiation mechanical properties data. The design was based on EBR-II gamma heating rates for tungsten, calculated in 1965. More recent work has shown that gamma heating is greater than estimated so that the specimens initially operated at temperatures in excess of the design temperatures. This overheating had the additional effect of raising the internal pressure over the design value which, coupled with the weakening of the specimens by overheating, caused failure of the pressurized tubes usually at one or more weldments. In capsules L2E and L2G, considerable diametral deformation of the tubes occurred before failure, 12% $\Delta D/D$ being measured on an L2E specimen. Capsules L2A and L2C did not show gross diametral increases, suggesting that these specimens failed at a low internal pressure. Evidence from pre-irradiation x-ray radiographs and from the manner of disassembly strongly suggests that L2C was not pressurized at all due to a faulty weldment at the bottom of the specimen train.

This early-in-life loss of pressure, coupled with the egress of sodium, substantially complicates the heat transfer analysis.

Figure 11-1. Schematic Diagram of L-2 Series Capsule (not to scale)

Table 11-1

ANALYSES OF L-2 SERIES TEST ALLOYS

Alloy	Incoloy 800	Type 316L	Type 347	Type 304
Capsule	L2A	L2C	L2E	L2G
Heat Number	4167	133310	99973	125653
Element				
Ni (wt %)	32.00	13.31	9.32	9.15
Cr (wt %)	20.44	17.39	18.20	18.42
Fe (wt %)	46.09	Balance	Balance	Balance
C (wt %)	0.05	0.02	0.05	0.06
Mn (wt %)	0.99	1.80	1.16	1.55
Si (wt %)	0.25	0.66	0.72	0.41
Cu (wt %)	0.33	0.07	0.20	0.27
Mo (wt %)	ND	2.20	0.14	0.50
P (wt %)	ND	0.017	0.023	0.025
S (wt %)	0.007	0.008	0.009	0.010
B (ppm)	12	9.5	12	16
N (ppm)	930	500	560	540
Other	0.51 Ti + 0.40 Al		0.91 (Nb+Ta)	
Grain Size (ASTM)	10	6-7	10	6-7

During disassembly of each capsule, sodium was found in the annulus between the specimens and the capsule tube, showing that the internal pressure had failed the specimens and forced sodium out. No trace of sodium was found in any of the individual specimen tubes or sodium reservoirs of capsules L2A, L2E, or L2G. This indicated that the internal helium pressure had ejected virtually all of the sodium from within the specimens into the annulus. The presence of traces of sodium within the lower two specimens of L2C, but not within the upper specimen or reservoir, reinforced the opinion that the faulty bottom weld had prevented pressurization so that sodium had simply run out until equilibrium levels occurred in the specimens and in the annulus. Thus, temperature calculations were performed for a number of possible situations within the capsule; i.e.,

- a. Beginning of life design condition
 - Specimen train filled with pressurized sodium
 - Annulus filled with argon
 - Reactor power = 45 MW
- b. Second condition, after specimen rupture
 - Specimen train part-filled with helium part-filled by sodium
 - Annulus part filled by sodium, part filled with argon
 - Reactor power = 45 MW

This condition was evaluated by considering the following possibilities:

- (1) Annulus filled with argon, specimen filled with helium
 - (2) Annulus filled with sodium, specimen filled with helium
 - (3) Annulus filled with sodium, specimen filled with sodium
- c. End of life condition. Examined by repeating conditions b.(1), (2), and (3) with the reactor power at 50 MW.

Calculations indicated that if all of the sodium contained in the specimens and reservoir (0.3 to 0.4 cu. in.) was ejected into the annulus, then the sodium level in the annulus would be approximately halfway up the midsection of the specimen train. Thus, the upper two density specimens (Figure 11-1) would always be surrounded by an argon-filled annulus which would largely govern the temperature of the specimen. The presence of helium or sodium within the specimen tube has little effect on the tube temperature, providing the argon-filled annulus is maintained. The temperatures of the density specimens taken from the two upper sections were reasonably well known.

If the design configuration had been maintained throughout the irradiation, the density specimen from the bottom specimen tube would have operated at a high temperature where little or no swelling would be expected. Also, the tensile specimens in the tube would be at a temperature well above the range in which swelling occurs. Thus, the extensive swelling observed in these components indicated that they operated at a lower temperature and thereby confirmed the presence of sodium in the specimen train-capsule annulus.

Table 11-2 lists the swelling data obtained from these specimens, together with the associated irradiation temperatures and accumulated fluences. The temperatures are essentially the same for equivalent sections in the various capsules. L2C shows a slight variation due to the specimen train's being situated slightly higher in the capsule than the others.

For purposes of comparison and analysis, the data have been plotted as a function of temperature after normalizing to the intermediate fluence level of $5 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. Figures 11-2 through 11-5 show the data for each material and compare them with other results for the same alloys. As might be expected from the difficulties involved in assessing the precise irradiation conditions, the data are not entirely consistent. However, some interesting trends are evident.

Figure 11-2 shows the L2C Type 316L data converted to $\% \Delta V/V$, together with immersion density results from the F2V, F2X, and F2H fuel pins. The data scatter is considerable, and can be ascribed to assumptions in normalizing as well as to difficulties in temperature estimation and variations in material composition. Analysis of swelling data for Type 304 has shown that the fluence dependence of stainless steel swelling is, in turn, dependent on the temperature during irradiation.¹⁵ In Figure 11-2, a uniform $(\phi t)^{1/8}$ was assumed, and a plot of only the F-2 density data¹⁸ has shown a more acute fluence dependence than this. Thus, more sophisticated normalizing would reduce the scatter. However, it is evident that Type-316 stainless steel undergoes a swelling maximum of 4 to 5% $\Delta V/V$ at 500 to 520°C (932 to 968°F) after an exposure of $5 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. The Type 347 data in Figure 11-3 show a similar temperature dependence, with a higher maximum swelling value. The L2A data in Figure 11-4 show extensive swelling for Incoloy-800, this capsule did not exhibit bulging and thus probably failed at a low stress, so that little or no stress effect would be expected.

Figure 11-5 shows the Type-304 data from L2G, together with a correlation and the 95% confidence intervals for a further observation based upon over 150 data points for Type 304 obtained from unstressed specimens. The low-temperature points are within the 95% data limits, but the other two points show dramatically more swelling than anticipated. The L2G specimens showed large $\Delta D/D$'s ($\sim 12\%$) from the internal pressurization. If this pressurization continued longer than first believed, extensive stress-enhanced swelling may have occurred at the high temperatures.

The trends found in Figures 11-2 through 11-5, together with information from previously available data, led to the schematic representation of the swelling of various alloys in the essentially unstressed condition that is shown in Figure 11-6. It is shown that different alloys exhibit differing temperature dependences and differing extents of swelling.

11.3.2 TEM Examination of F2 Claddings

Examination of cladding samples from the F2 series of fuel pin irradiations by transmission electron microscopy (TEM) has been initiated. The TEM specimens were prepared from sections of the alpha-decontaminated fuel cladding by cutting 3-mm-diameter discs from the cladding with an electro-discharge machine (EDM). In this way, the discs were not deformed during the cutting operation. The resultant curved discs were thinned down to foil thickness by twin-jet electro-polishing in a specially designed specimen holder. Void number densities and size distributions were measured for each foil. Results are available for specimens irradiated at 525 to 545°C (976 to 1013°F) from pins F2H and F2V (Type-316 cladding) and F2T (Type-347 cladding).

The most striking feature observed is the large amount of small precipitate particles found within the grains. This precipitation, which is heavier in Type 347 than in Type 316, shows a definite crystallographic orientation with respect to the matrix material. Voids are frequently associated with these particles, and the void number density for Type 347 is greater than in the Type 316. Identification of the precipitate particles is not complete, but indications are that it is σ -phase. Much larger precipitate particles of M_{23}C_6 are randomly distributed throughout the grain boundaries.

Figure 11-2. Type-316 and -316L Stainless Steel Immersion Density Data

Figure 11-3. Type-347 Stainless Steel Immersion Density Data

Figure 11-4. Incoloy-800 Immersion Density Data from L2A

Figure 11-5. Type-304 Stainless Steel Immersion Density Data from L-2 Capsules

Figure 11.6 Swelling of Various Austenitic Alloys

Table 11-2

SWELLING DATA FROM L-2 SERIES CAPSULES

Specimen	Alloy	Fluence $\text{n/cm}^2 \times 10^{22}, E > 0.1$ MeV-GE	Nominal Calculated Temperatures		Swelling $-\Delta\rho/\rho\%$
			$(^{\circ}\text{F})$	$(^{\circ}\text{C})$	
L2A 1A	1800	4.03	870	465	5.8
L2A 2A1	1800	5.22	1025	550	5.4
L2A 2A2	1800	5.5	1170	632	5.2
L2A 3A1	1800	7.34	730	388	4.4
L2A-3A2	1800	7.31	710	376	3.5
L2C 1A	316L	4.03	840	448	0.3
L2C-2A1	316L	5.22	995	535	3.7
L2C 2A2	316L	5.5	1140	616	1.4
L2C 3A1	316L	7.34	730	388	3.0
L2C-3A2	316L	7.31	710	376	2.6
L2E-1A	347	3.73	870	465	1.5
L2E 2A1	347	4.8	1025	550	1.7
L2E-2A2	347	5.05	1175	635	1.9
L2E 3A1	347	6.72	730	388	2.5
L2E-3A2	347	6.7	710	376	0.4
L2G-1A	304	3.73	870	465	4.3
L2G-2A1	304	4.8	1025	550	6.6
L2G 2A2	304	5.05	1165	630	6.8
L2G-3A1	304	6.72	730	388	4.2
L2G-3A2	304	6.7	710	376	3.1

The void size and number density data are given in Table 11-3 together with earlier data for Type 304 obtained from the GE MT3 materials capsule.¹⁶

The void number density (materials irradiated to fluences of $3\text{--}4 \times 10^{22}$, $E > 0.1\text{ MeV-GE}$) increases by two orders of magnitude from Type 304 (Figure 11-7), through Type 316 (Figure 11-8), to Type 347 (Figure 11-9), while the average void diameter decreases by a factor of 4. The observed increase in void number density qualitatively corresponds with the degree of intragranular precipitation, which was very slight in Type 304, heavy in Type 316, and even heavier in Type 347. Void nucleation is probably enhanced by the presence of small precipitate particles. However, it appears that the overall vacancy supersaturation is not affected to the same extent since the fewer voids in Type 304 grew much larger than those in the Type 347, and produced almost as much swelling.

The data obtained from pin F2V confirm a trend obtained from the examination of Type 304 stainless steel irradiated in the MT3 capsule.¹⁶ Those data showed that void nucleation essentially ceased at high fluences. The F2V and F2H data (Figures 11-8 and 11-10, and Table 11-3 show that little nucleation has occurred between 3.1 and $7.3 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$ (void number density doubled), whereas the mean void diameter has increased from 365 to 640 Å, resulting in an increase in swelling from 0.7 to 8.7% $\Delta V/V$. Volume change due to the presence of voids is linear with number density and cubic with void diameter.

11.3.3 MT-3 Chemistry

A chemical analysis has been completed on an irradiated section of the stainless steel strip from the MT-3 capsule. The results (0.054% C, 69.9% Fe, 18.7% Cr, 8.95% Ni, 1.5% Mn, 0.5% Si, 0.056% N) conformed to an ASTM Type-304 stainless steel. The alloy content was determined by a combination of wet chemistry, spectrographic analysis, and combustion-gas chromatography (for carbon). The spectrographic analysis also gave several trace element (Cu, Ti, Co, Mo, Pb, V, Sn, Al) concentrations.

Examinations of the above material by transmission electron microscopy revealed very large (>2000 Å) voids in the material. The chemical analysis was performed to confirm that the material was in fact Type-304 stainless steel, and thus the fast neutron induced void formation observed in MT-3 was typical of a Type-304 stainless steel.

11.4 CLADDING ALLOY MECHANICAL PROPERTIES

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

11.4.1 Biaxial Creep Program

The biaxial creep facility previously described¹⁷ was designed and fabricated for creep testing highly irradiated tubing materials at elevated temperatures. The test program is designed to obtain creep, stress rupture, and ductility data on irradiated tubular specimens.

11.4.2 Creep Test Results

Specimens being tested have been described previously.¹ These specimens were fabricated from Type 304 capsule tubing irradiated in EBR-II at temperatures of 371 to 482°C (700 to 900°F) to fluences of $\sim 3 \times 10^{22}$, $> 0.1 \text{ MeV-GE}$.

After 2275 hours, 12 biaxial creep specimens have failed. Failure of these constant stress specimens at relatively low pressure levels was evidenced by a slow loss of pressure. Failure is also indicated when the specimen diameter ceases to increase after a given period of time under test. A confirmation of failure is obtained by puncturing and pressure-testing the specimens. Failure of eight specimens has been confirmed in this manner. All the failed specimens showed less than 1% total strain and had failed in less than 1486 hours.

It was necessary to assess the extent of swelling annealing at the test temperatures and the potential strain measurement error introduced by swelling annealing. After the puncturing operation, density measurements were made on sections from two specimens tested at 704°C (1300°F) and one specimen tested at 482°C (900°F). The post-test densities were then compared with the pre-test densities. The results are shown in Table 11-4. Both specimens tested at 704°C (1300°F) showed a density increase. The density of the 482°C (900°F) specimen was unchanged. This group of creep specimens had relatively small amounts of swelling ($< 0.5\%$ $\Delta V/V$ for the L2 specimens and $< 2\%$ $\Delta V/V$ for the F2 specimens) and the density measurements indicate that only about 50% of the swelling is recovered; hence, the measurement error introduced by swelling annealing in the highest temperature group of specimens is small—a maximum error of 0.0007 in. This problem may well be acute in attempting any form of creep tests on higher fluence

Figure 11-7. Type-304 Stainless Steel from MT-3. Specimen irradiated at 555°C to $3.7 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. Mean void diameter = 1000 Å, void number density = $2.1 \times 10^{13} / \text{cm}^3$, swelling = 1.8% $\Delta V/V$.

60,000X

Figure 11-8. Type-316 Stainless Steel from Pin F2H. Specimen irradiated at 545°C to 3.1×10^{22} n/cm², $E > 0.1$ MeV-GE. Mean void diameter = 365 Å, void number density = 2.1×10^{14} /cm³, swelling = 0.7% $\Delta V/V$.

150,000X

Figure 11-9. Type 347 Stainless Steel from Pin F2T. Specimen irradiated at 540°C to $3.9 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. Mean void diameter = 250 Å, void number density = $15 \times 10^{15} / \text{cm}^3$, swelling = 2.3% $\Delta V/V$

150,000X

Figure 11-10 Type-316 Stainless Steel from Pin F2V. Specimen irradiated at 525°C to $7.1 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. Mean void diameter = 640 Å, void number density = $4.4 \times 10^{14} / \text{cm}^3$, swelling = 8.7% $\Delta V/V$.

165,000X

Table 11-3

**TRANSMISSION ELECTRON MICROSCOPY RESULTS FOR VARIOUS
AUSTENITIC STAINLESS STEELS IRRADIATED AT
525°-550°C (978°-1032°F)**

Alloy	Sample Source	EBR-II Fluence E > 0.1 MeV n/cm ² X 10 ⁻²² , GE	Calculated Temperature During Irradiation		Void Number Density (cm ⁻³)	Void Mean Diameter (Å)	Swelling ΔV/V (%)	Extent of Intragranular Precipitation
			(°C)	(°F)				
304	Materials Capsule MT3	3.7	555	1032	2.1 X 10 ¹³	1000	1.8*	Light
316	Fuel Pin F2H	3.1	545	1013	2.1 X 10 ¹⁴	365	0.7**	Heavy
	Fuel Pin F2V	7.1	525	978	4.4 X 10 ¹⁴	640	8.7**	Heavy
347	Fuel Pin F2T	3.9	540	1004	1.5 X 10 ¹⁵	250	2.3**	Very heavy

*Void volume fraction from TEM examination

**Swelling from immersion density measurement.

specimens with the potential for greater swelling annealing. In view of the preceding conclusions, it was considered necessary to evaluate the extent of swelling annealing on the high fluence ($\sim 7 \times 10^{22}$ n/cm² > 0.1 MeV) specimens before creep tests could be initiated.

Two unstressed, 1/2-in.-long sections of high fluence F2 capsule tubing have been annealed at 482°C (900°F) for 500 hours to determine the extent of swelling annealing (if any) at this temperature. Length and diameter measurements have been made and no change in dimensions has been detected within the measurement capabilities of ± 0.0003 in. It would appear that swelling annealing is not significant at 482°C (900°F) for the F2 materials, and creep tests at this temperature have been initiated on the highly exposed samples. Swelling annealing is expected to be significant at 704°C (1300°F), and annealing of samples at this temperature has been initiated. It is uncertain whether the swelling annealing is time and temperature dependent or solely a function of temperature. If the latter case is true, it is feasible to attempt strain measurements; however, if swelling annealing is a function of both temperature and time, it will be difficult to obtain reliable creep strain measurements at the higher test temperatures.

11.4.3 Tensile Testing

Tensile specimens from the L2 materials capsule irradiated in EBR-II S/A X050 to a peak fluence of 6.8×10^{22} , E > 0.1 MeV-GE are now being tested.

Four types of stainless steel are being evaluated: Types 304, 316, 321, and 347, along with one group of Incoloy-800 specimens.

Testing is being carried out at four temperatures: 482°C (900°F), 593°C (1100°F), 704°C (1300°F), and 815°C (1500°F); and three strain rates: 0.0002 in./in./min, 0.02 in./in./min, and 2 in./in./min. Five specimens have been tested at 704°C (1300°F) thus far, and results indicate ductility values of 0.1 to 1% for the stainless steels. At the conclusion of this series of tests, tensile tests will be initiated on the MT-3 high nickel alloy specimens irradiated in EBR-II S/A XA08 to a peak fluence of 6.5×10^{22} n/cm², E > 0.1 MeV-GE. The three alloys are: Hastelloy-X, Inconel-625, and Incoloy-800.

11.5 MATERIALS MANAGEMENT

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

11.5.1 Stress-Rupture Behavior of Defected Tubing

One of the foremost objectives of current studies on the behavior of cold-worked materials is the selection of a cold-work level most acceptable for use in a fuel cladding tube. In such a selection, establishment of the behavior of a material to a purely thermal environment is an essential prerequisite. As this was accomplished on relatively perfect material, the logical second phase of the program was the evaluation of the influence of imperfections. Particularly important in this regard is the tube surface defect, a local wall thinning that constitutes one of the most common production defects and one of the easiest defects to escape detection.

To permit a controlled study of the effect of such defects, tube sections selected from well characterized and acceptable stock were impressed with intentionally fabricated defects of given sizes and depths. The test program was restricted to tubes with 10 and 20% residual cold work since it was these materials that exhibited the most acceptable long-term properties of the four types tested.

Defects produced by electrical discharge machining were impressed on both the inner and outer surfaces of the tubes. The defects were oriented axially and ranged in depth from 1.0 to 3.5 mils. This represents seven to 23% local thinning of the tube wall. All defects were nominally 0.125 in. long. The tubes were tested at conditions identical to those used in the stress rupture testing of as-received tubing, discussed previously.⁴

Although the test program at 704°C (1300°F) has not been completed yet, the number of tube failures that have occurred is sufficient to establish the following trends. The 20% cold-worked tubing, shown in the earlier work to be the stronger, is the more resistant of the two cold-worked types to surface defects. The 20% cold-worked material also shows less notch sensitivity than the 10% cold-worked tubing in the defect range currently established as the minimum allowable, i.e., 10% of the tube wall in depth. This was evidenced by the frequency at which failures occurred at the defect. The data from which these observations were inferred are shown in Table 11-5. Also shown in this table is the effect of outside diameter surface defects on the stress-rupture life of cold-worked tubes. The results indicate that although the 20% cold-worked tubes are superior at all levels of stress, neither the 10% nor the 20% cold-worked tubes with 10% of the wall-deep defects shows degradation in rupture lifetimes at hoop stresses as high as 13,000 psi. This stress level is substantially higher than pressures generated during in-reactor operation are expected to impose.

In terms of their effect on stress-rupture life, defects on the inner surface cannot be distinguished from similar size and orientation defects occurring on the outer surface. This applies to both cold-worked grades tested. In terms of ductility, however, the results to date show that the reduction in maximum diametral ductility of tubes with defects on the inner surface is a somewhat more critical function of depth than are tubes with outside diameter surface defects. For example, at defect depths of nominally 20% of the tube wall, 10% of the cold-worked tubes with inside diameter surface defects show, on the average, a 50% reduction in maximum diametral ductility compared with tubes having similar size defects on the outside diameter surface. This effect is not readily apparent in the higher cold-worked type.

Table 11-4

EFFECT OF POST IRRADIATION TEST TEMPERATURE ON VOLUME SWELLING IN CREEP SPECIMENS

Specimen Number	Density After Irradiation Prior to Creep Test	$\Delta V/V$ (%)	Creep Test Temperature	Density After 2275 Hours Creep Testing	$\Delta V/V$ (%)
F2CB (A3)	7.7685	1.69	704°C 1300°F	7.8247	0.98
F2HB	7.763	1.76	704°C 1300°F	7.8409	0.77
F2CA	7.8261	0.96	482°C 900°F	7.8192	1.05

Table 11-5

**INFLUENCE OF SURFACE DEFECTS (BOTH INSIDE AND OUTSIDE) ON
STRESS-RUPTURE PROPERTIES OF COLD WORKED TUBING OF
TYPE 316H STAINLESS STEEL TESTED AT 1300°F (704°C)**

	1.5 ± 0.5 Mil Deep Defect (7 - 13% of Wall)		3.0 ± 0.5 Mil Deep Defect (17 - 23% of Wall)	
	10%CW Tubing	20%CW Tubing	10%CW Tubing	20%CW Tubing
Failure Location (% failing at defect)	70%	20%	100%	100%
Maximum Reduction in Diametral Ductility	51%	17%	61%	50%
Reduction in Life* at 11,000 psi, (Nominal Defect Depth)	0%	0%	5%	< 5%
Reduction in Life* at 13,000 psi, (Nominal Defect Depth)	0%	0%	37%	30%
Reduction in Life* at 15,000 psi, (Nominal Defect Depth)	13%	< 5%	59%	34%
Reduction in Life* at 17,000 psi, (Nominal Defect Depth)	23%	Not Tested	61%	Not Tested

* Reduction with respect to undefected control material. (These data refer to tubes with o.d. surface defects only.)

12. TASK N—CORE PHYSICS ANALYSIS

Task Leader: B. A. Hutchins*

12.1 GENERAL

The purpose of Task N is to contribute to the development of improved physics computational tools for the LMFBF program. To accomplish this work, computational methods and nuclear data used in the design of LMFBF's are evaluated and the need for improved methods and data is determined. Acquisition of the improved methods and data is accomplished by development efforts within Task N and by cooperative activities in the national LMFBF program. Results of the evaluation studies, together with results from associated reactor design and development activities, also provide direction for establishing current LMFBF design and safety criteria.

12.2 COMPILATION AND DISTRIBUTION OF EVALUATED DATA

A task force meeting on ENDF/B, Version II, data was held at Brookhaven National Laboratory on February 4-5, 1971, to evaluate the results of the testing program of the Cross Section Evaluation Working Group, and to establish guidelines for modifying the ENDF/B data files. All of the results reported showed ENDF/B II giving low k_{eff} values for the benchmark critical experiments. Preliminary results of modifications to σ_f for Pu-239 and U-238 that improve the computed k_{eff} values were reported by General Electric at this meeting, and were factored into the task force decision to evaluate alternate data sets by adjusting group constants in the benchmark calculations. The principal data changes to be considered are a 3 to 10% increase in the Pu-239 fission cross section between 10 keV and 1 MeV, a 5% reduction in the U-238 capture cross section between 100 keV and 1 MeV, an 8% increase in the U-238 fission cross section and a gradual increase—up to 10%—in the fission spectrum temperatures at higher incident neutron energies. Subsequent to the meeting, the ZPR3-48 and ZEBRA-3 results were recomputed with this modified data and the eigenvalues were found to change as follows:

	Version II Data	Modified Version II Data
ZPR3-48	0.9688	0.9992
ZEBRA-3	0.9480	0.9860

The results indicate that the modified data considerably improve the agreement between calculated and measured eigenvalues.

Concurrently with the CSEWG testing program, a more detailed evaluation of data modifications that improve the computed parameters for the benchmark assemblies was carried out. The effects of important cross section variations on the reactivities of the critical assemblies were determined from perturbation calculations and were used as a basis for modifying the ENDF/B II data. Variations in the cross sections were limited to the uncertainty ranges specified by Greebler, et al.¹⁸

Cross section adjustments were determined that minimize the sum of the squares of the differences between the calculated and measured eigenvalues. These results are given in Table 12-1. The first modification in Table 12-1 includes the increases to the Pu-239 and U-235 fission cross sections that improve the agreement between calculated and measured eigenvalues for those assemblies with a low U-238 concentration (i.e., VERA-11A), and an increase in the U-238 fission cross section that is within the limits of the uncertainty range for that reaction. These changes result in a considerable improvement in the calculated eigenvalues for most of the assemblies, but are still not completely satisfactory for those assemblies with the highest concentration of U-238. The second modification in Table 12-1 includes reductions in the U-238 capture and inelastic cross sections, and gives eigenvalues that are within about 0.5% of the measured values for all assemblies. These results were presented at the Third Conference on Neutron Cross Sections and Technology which was held on March 15-17, 1971, at Knoxville, Tennessee.¹⁹

A similar set of modifications to ENDF/B-II was also presented by T. Pitterle, et al.,²⁰ at the Knoxville conference. Pitterle's recommended modifications, along with the results of the GE evaluation in Table 12-1 have been used as a basis for generating a 29-group cross-section file which should produce reliable LMFBF design parameters.

* Refer to Appendix B for listing of contributors.

Table 12-1

**EFFECTS OF RECOMMENDED MODIFICATIONS TO ENDF/B,
VERSION II DATA ON CALCULATED EIGENVALUES**

Plutonium-Fueled Assemblies	ENDF/B Version II Data	Calculated Eigenvalues	
		Modification 1 [†]	Modification 2 ^{††}
JEZEBEL	0.9892	1.0010	1.0010
VERA-11A	0.9686	0.9924	0.9944
SEFOR	0.9630	0.9916	0.9995
ZPR3-48	0.9650	0.9950	1.0056
ZEBRA-3	0.9480	0.9801	0.9986
Uranium-Fueled Assemblies			
VERA 1B	0.9894	0.9957	0.9978
GODIVA	0.9979	0.9993	1.0001
ZPR3-6F	0.9918	0.9991	1.0009
ZPR3-12	0.9813	0.9960	1.0045
ZEBRA-2	0.9620	0.9834	0.9972
ZPR3 11	0.9650	0.9786	0.9984

[†] MODIFICATION 1

- a) Five percent increase in the U-235 fission cross section between 1-100 keV
- b) Five percent increase in the Pu-239 fission cross section between 100-600 keV
- c) Ten percent increase in the Pu-239 fission cross section between 10-100 keV
- d) Eight percent increase in the U-238 fission cross section

^{††} MODIFICATION 2

Those cross section adjustments in modification 1 plus

- a) Three percent reduction in the U-238 capture cross section between 1 keV and 1 MeV
- b) Fifteen percent reduction in the U-238 inelastic cross section above 800 keV

The cross section changes from ENDF/B II are summarized as follows

- The U 238 fission cross section was increased by 6%
- The ν values for U 238 were increased by 0.5%
- The U 238 inelastic cross section was reduced by 10% above 800 keV
- The U 238 capture cross section was reduced by 3% between 4 keV and 100 keV, and reduced by 10% between 100 keV and 1 MeV
- The Pu 239 fission cross section was increased by 2% between 1 keV and 10 keV, increased by 10% between 10 keV and 100 keV, and increased by 2.5% from 100 keV to 1 MeV
- The Pu 239 alpha value was increased by 8% from 500 eV to 10 keV, increased by 15% from 10 keV to 25 keV, and increased by 7% from 25 keV to 10 MeV

These modifications differ from Modification 2 in Table 12.1 principally in a smaller reduction in U 238 inelastic, a larger reduction in U 238 capture, a smaller increase in Pu 239 fission and an increase in the Pu 239 alpha value. They are in essential agreement with the modifications recommended by T. Pitterle, et al.,²⁰ with the exceptions of a decrease in U 238 inelastic (rather than an increase in the nuclear temperature), a slight decrease in U 238 capture between 4 and 100 keV, and a larger increase in Pu 239 fission between 10 and 100 keV. The decrease in U 238 inelastic was necessary to bring assemblies with high U 238 concentrations into line.

Calculations with the modified 29 group cross section file are presently being run for several of the benchmark assemblies. First results for ZPR III 48 give the following

	Calculation	Measurement
k_{eff}	1.002	1.000
$\sigma_f^{238}/\sigma_f^{239}$	0.0345	0.0314
$\sigma_c^{238}/\sigma_f^{239}$	0.139	0.141
$\sigma_f^{240}/\sigma_f^{239}$	0.250	0.249

The indicated measured fission ratios are from back to back counters. Foil measurements of $\sigma_f^{238}/\sigma_f^{235}$ show some sensitivity to cell position which may indicate that the counter value for $\sigma_f^{238}/\sigma_f^{239}$ is low.

The analyses performed for the evaluation of ENDF/B II included computations for the SEFOR reactor. As with the benchmark assemblies, ENDF/B II results in a low k_{eff} which is improved by the data modifications studied. The SEFOR results are included in Table 12.1.

12.3 SUPPLEMENTARY ANALYTICAL INTERPRETATION OF INTEGRAL DATA

Meetings were held on February 23, and March 4, 1971, with personnel from Argonne National Laboratory, to discuss the experimental programs in the ZPPR and ZPR 6 facilities and to present General Electric's needs and priorities for these programs. For the analysis of uncertainties in physics design quantities, it was recommended that the pin measurements in ZPPR, particularly the within pin foil activations, be expanded to include more spatial information. The need was also expressed for capture rate measurements in plutonium isotopes to evaluate breeding ratios. The possibility of performing reactivity/reaction rate measurements for these reactions will be investigated.

The SEFOR reactor has provided the first measurement of the Doppler coefficient of fuel in the full core of an LMFBR. Analyses were performed to provide a direct comparison of the calculated isothermal Doppler coefficient, using a documented data file with the experimental result. The ENDF/B II data file was used with the two dimensional synthesis diffusion theory code, BISYN, to compute an isothermal Doppler coefficient in SEFOR of $-T(dk/dT) = 0.0093$. This is considerably larger than the measured value of 0.0081 ± 0.0014 ²¹ but within the experimental error, and 9% higher than the value computed with the SEFOR data file²¹. It is believed that the high computed value is partially due to the low k_{eff} obtained with ENDF/B II (see Table 12.1), and the modified 29 group file described above will result in a smaller computed Doppler.

Since this is a comparison of analytical isothermal Doppler coefficients, several possible sources of error in obtaining equivalent isothermal values from the SEFOR measurements are being investigated. The effect that spectral hardening due to the Doppler effect has on power distribution in SEFOR has been evaluated. Radial diffusion theory problems at two power levels were completed, one at an isothermal temperature of 650°K (corresponding to low power) and the other using a core radial fuel temperature distribution for SEFOR at 20 MW. It was found that the core

radial power distribution was not significantly different between the low power and 20 MW cases, but that the prompt neutron lifetime decreased by 3.4%. It is not felt that this will produce a significant change in the measured value for the isothermal Doppler coefficient.

Another possible source of error in the isothermal value is the assumption of a constant fuel temperature across the fuel pin. Because of the large diameter of the SEFOR fuel, the radial temperature variation across the pin may cause a significant variation in the effective self-shielding of the resonances with radial position in the pin. Calculations to evaluate this effect are underway.

12.4 IMPROVEMENT OF FUEL CYCLE AND MANAGEMENT SCHEMES

The fuel cycle code FUMBLE^{2,2} was sent to the Argonne Code Center, along with descriptive documentation. The XPORT routine was used to put FUMBLE into a format that can be more easily read and operated on other computers.

The routine that performs a stepwise interpolation between precomputed beginning of cycle and end of cycle parameters (e.g., regional fluxes, cross sections, volumes, etc.), has been checked out. This routine will permit a more accurate calculation of the change in fuel isotopic concentrations during the operating cycle.

The topical report describing the FUMBLE code^{2,2} was issued.

12.5 IMPROVEMENTS OF EXISTING METHODS FOR GENERATING GROUP CONSTANTS

A draft of the Users Manual for Version 2 of the ENDRUN code has been completed, and a Programmers Manual is being written. A procedural error was found and corrected in the TDOWN 2 code which reduced the number of iterations required for the elastic removal correction. TDOWN 2 has undergone a procedural checkout and now awaits documentation before being sent to the Argonne Code Center.

12.6 COMPUTERS, COMPUTER LANGUAGE, AND CODE CENTER

The computer code, XPORT, which prepares code tapes for shipment to the Argonne Code Center and other outside users, is being used on the SYN and TDOWN 2 codes. TDOWN 2 has been successfully run through the code, but SYN, due to its rather large bulk of cards, has required further modifications in XPORT in order to compact the card data on tape. XPORT generates a seven track, BCD mode tape at 556 BPI in the standard IBM Code 5 character set.

The input card format subcommittee, of the *de facto* Committee on Computer Code Coordination, met in Chicago on April 6-7. This subcommittee agreed to adopt a completely free style type of input data card format as a standard. This format has all the ingredients which the *de facto* committee had specified as being desirable. The rules governing the use of this format have essentially been specified.

In the process of specifying the input card format, it was necessary for the subcommittee to consider the general approach to input handling. The approach that is consistent with the card format tends to relegate the "card sweeper" routine subordinate to a following processor routine or series of processors. The argument list connecting the "card sweeper" to the processor was also discussed. A report covering the general approach and the definition of the input data card format will be sent to the members of the *de facto* committee for their edification and comments before being finalized.

13. TASK P—FUEL ELEMENT DESIGN ANALYSIS

Task Leader: F. E. Gelhaus*

13.1 GENERAL

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

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

13.2 FUEL ELEMENT MODELING

The BEHAVE I computer code was used in the analysis of the fuel pins in the AEC RDT sponsored Fuel Modeling Round Robin. With a fuel swelling rate of 0.22% per 10^{20} fissions/cc, only the cladding strain of the GE fuel pin F2H was predicted to have occurred by mechanisms operating in addition to metal swelling. The strain at first startup and that resulting from a mid-life power increase were included.

Fuel pin FOE was used to calibrate the power increase effect. After 217 days of EBR II operation in grid position 7A1, a maximum linear power increase from 5.8 to 10 kW/ft occurred due to the introduction of a stainless steel reflector. A second mid-life power change occurred upon the return to a reactor power of 45 MW after operation at 30 MW. The modeling of this mechanism is discussed in the Thirty-Seventh Quarterly Report.¹ The calculated and measured axial diameter profiles for FOE are compared in Figure 13.1; it is seen that with the power increase effect, the observed post-irradiation diameter increase can be accounted for. The inclusion of the power increase effect in the modeling of fuel pin F2H is not sufficient in itself, however, to effect agreement with the measured diameter profile.

13.3 FUEL BEHAVIOR STUDIES

13.3.1 Fuel Axial Flow Resistance

The GE BEHAVE codes are unique in their emphasis on the importance of properly accounting for the axial flow of fuel down the hot plastic core of the fuel. Other models make the plain strain assumption that there is no axial fuel flow. However, detailed data analysis has indicated that local variations in fabrication parameters are not, in general, manifest in local effects on cladding diameter.^{2,3} Profilometry data indicate the ability of fuel to flow over considerable distances, yielding effects that are "global" in nature rather than local. Because of the importance of axial flow effects, the mean pin diametral increase is best used in correlations with fabrication parameters and operating history rather than local values.

A recent analysis evaluates the distances over which fuel may flow axially under characteristic pressure differences and steady state fuel swelling rates.

Figure 13-2 illustrates the relative ease with which fuel can move axially, and considers that the volumetric expansion of the swelling fuel can be relieved only by extrusion into an unoccupied region some distance away along the fuel pin centerline. This is a limiting case and corresponds to no available local porosity. In this analysis, it is assumed that no radial displacement can occur, that the stress in the cladding is kept reasonably low, and that the entire volumetric expansion is relieved through the axial flow effect.

As the center temperature is increased due to an increase in linear power, the resistance to axial flow decreases because of a decrease in fuel viscosity. Figure 13.2 indicates the length through which a local hydrostatic pressure of 2000 psi will push fuel that is swelling at a rate of $\Delta V/V = 5 \times 10^{-6}$ per hour. It can be seen that a 1900°C center temperature is required to produce flow the length of a 0.25 in. fuel pellet, thus negating the dependence of diametral increases on local pellet properties. Applying the same analysis to center temperatures close to melting requires a great deal of extrapolation of the creep data, but results indicate that flow could occur over distances approaching the length of the fuel pin. If central melting should be present, a step decrease in the viscosity of the central fuel would occur and the axial flow resistance of the fuel would be virtually eliminated. At the much higher strain rates characteristic of power increase effects, this analysis indicates that little axial fuel flow is to be expected where the plastic core is full of solid fuel. However, where the central core of fuel consists of molten fuel, the axial flow resistance will be negligible due to the relatively low viscosity of the material in the molten state.

* Refer to Appendix B for listing of contributors

13.4 FAILURE MODE STUDIES

13.4.1 STATUS: Local Deformation and Lifetime code

The purpose of STATUS is to calculate local cladding strain distribution and residual cladding lifetime as dependent upon fuel pin operating history. The method of calculating the strain distribution for the loading mechanisms of differential thermal and swelling expansion of the cladding was reported in the Thirty Fourth Quarterly Report^{1,2}. The Thirty Sixth Quarterly Report⁴ contains a review of the effects considered within STATUS, and their degree of implementation. The method of separating the total strain into components according to the cause of strain and the failure criteria were also described. In the current quarter, steady state loading has been included by superimposing the creep strain calculated by BEHAVE on the creep components calculated by relaxation in the transient step model presented previously.

13.4.2 Effects of Stress Biaxiality on Failure Criterion

An evaluation of the biaxial treatment and failure criterion employed in STATUS was made by comparison with experimental data. The cladding stress was assumed to be biaxial with radial stress neglected. STATUS is encoded to calculate the effective strain from the principal strain components at each cladding node by the method of von Mises. Effective strain can be compared directly with failure strains obtained in uniaxial material properties tests, while values of individual components of strain at failure are functions of the stress "biaxiality." Figure 13.3 shows the theoretical normalized values of longitudinal and circumferential strains as functions of the stress biaxiality ratio (longitudinal stress / circumferential stress). On this diagram, a longitudinal uniaxial stress state (biaxiality 1.0) occurs at the left, while the circumferential uniaxial state (biaxiality 0.1) is at the right. Also indicated for reference are the biaxiality states for pressurized closed end tubes (1.2) and spherical vessels (1.1). It is necessary to consider the full spectrum of biaxiality and not just the pressurized tube case because, on a local basis, any of these states might exist due to nonuniform loading such as superimposed differential thermal and metal swelling. For illustration, if these curves are interpreted at failure for a pressurized tube (biaxiality ratio, 1.2), the tube would be expected to fail when the circumferential strain reached approximately 85% of the ductility in a uniaxial tensile test.

If the material properties are isotropic, as is assumed in the theoretical development, the maximum strain ratios in the two uniaxial stress states (1.0, 0.1) should be identical, as shown by the calculated curves. The biaxial experimental data^{2,5} shown in Figure 13.3 do not conform to theory. This strain data was normalized by using the longitudinal strain under uniaxial longitudinal load. The considerable theoretical/experimental discrepancy at the uniaxial hoop stress

Figure 13.1 Axial Fuel Pin Diameter Profile, Pin FOE

Figure 13-2. Axial Flow of Fuel Under Thermal Creep

condition (experimental less than half the theoretical value) suggests that the test material ductility was not isotropic. This is not unexpected, considering the drawing processes in tube manufacturing. These results point out that better knowledge of the anisotropic nature of ductility is needed in order to permit the proper adjustment of the theoretical approach.

As presented, the theory is quite optimistic at the lower stress ratios (i.e., at a stress biaxiality of 1:2, 85% was predicted but only 30% achieved). Less optimistic results would be predicted if the experimental data were normalized to the smaller diametral ductility under uniaxial hoop stress (i.e., at 1:2 biaxiality, 70% would have been achieved instead of 30%). This adjustment would cause prediction of only about half the available ductility at the uniaxial-longitudinal stress state. This more conservative approach is suggested until more complete data for cladding are available.

13.4.3 Experimental Support of Failure Criterion

As reviewed in the Thirty-Sixth Quarterly Report,⁴ the STATUS failure model is based on the Cumulative Strain Damage Failure Criterion. This approach assumes that all increments of plastic strain (normalized to current failure strain) are additive. Failure is assumed to occur when the sum of these normalized strain increments, called the damage index, is equal to 1.0. The theory is expressed as:

$$\text{Damage Index} = \sum_i \left(\frac{\Delta \epsilon}{\epsilon_R} \right)_i$$

where

$\Delta \epsilon$ = strain increment during condition i ,

ϵ_R = rupture strain under condition i .

Figure 12-5 of the Thirty-Sixth Quarterly Report shows this relation graphically.

Data have been found that support the theory. WADCO^{2,6} conducted out-of-pile tests showing ductility results of early creep test termination, followed by a tensile test to failure. Data for Types-304 and -316 stainless steel at three test temperatures were normalized and replotted on the cumulative strain damage diagram in Figure 13-4. Good agreement exists between the data and the theoretical failure line. The uncertainty produced by variations in the normalizing values of creep and tensile ductility are indicated by the shaded region.

Similar data from Task L tests are shown in Figure 13-5. An insufficient number of normalizing values was available, and it was necessary to estimate the tensile limit. Only two creep ductility tests were conducted; these results were used to estimate uncertainty along the entire curve. These data also show reasonable agreement with the theory.

Using these two curves, a limiting value of the damage index could be chosen to preclude failure during similar tests. The agreement of data and theory indicates that the cumulative strain damage criterion will probably be a useful tool for life estimates of reactor structures, as soon as the needed ductility data are obtained.

Figure 13-3. Strain versus Biaxiality

Figure 13-4. Cumulative Strain-to-Failure Criterion Supported by Sequential Creep-Tensile Tests (Unirradiated) Data

Figure 13-5. Cumulative Strain Damage Diagram, Sequential Creep-Tensile Tests (Unirradiated)

APPENDIX A. PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Nos. 1 through 106, from July 1959 through March 1961.

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

GEAP-3888
First Quarterly Report
October–December 1961.

GEAP-3957
Second Quarterly Report
January–March 1962.

GEAP-3981
Third Quarterly Report
April–June 1962.

GEAP-4080
Fourth Quarterly Report
July–September 1962.

GEAP-4158
Fifth Quarterly Report
October–December 1962.

GEAP-4214
Sixth Quarterly Report
January–March 1963.

GEAP-4300
Seventh Quarterly Report
April–June 1963.

GEAP-4382
Eighth Quarterly Report
July–September 1963.

GEAP-4480
Ninth Quarterly Report
October–December 1963.

GEAP-4601
Tenth Quarterly Report
January–March 1964.

GEAP-4640
Eleventh Quarterly Report
April–June 1964.

GEAP-4723
Twelfth Quarterly Report
July–September 1964.

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

GEAP-4860
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February–April 1965.

GEAP-4931
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May–July 1965.

GEAP-4982
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August–October 1965.

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

GEAP-5158
Eighteenth Quarterly Report
February–April 1966.

GEAP-5198
Nineteenth Quarterly Report
May–July 1966.

GEAP-5292
Twentieth Quarterly Report
August–October 1966.

GEAP-5433
Twenty-First Quarterly Report
November 1966–January 1967.

GEAP-5491
Twenty-Second Quarterly Report
February–April 1967.

GEAP-5522
Twenty-Third Quarterly Report
May–July 1967.

GEAP-5541
Twenty-Fourth Quarterly Report
August–October 1967.

GEAP-5584
Twenty-Fifth Quarterly Report
November 1967–January 1968.

GEAP-5631
Twenty-Sixth Quarterly Report
February–April 1968.

GEAP-5677
Twenty-Seventh Quarterly Report
May–July 1968.

GEAP-5700
Twenty-Eighth Quarterly Report
August–October 1968.

GEAP-5753
Twenty-Ninth Quarterly Report
November 1968–January 1969.

GEAP-10028
Thirtieth Quarterly Report
February–April 1969.

GEAP-10028-31
Thirty-First Quarterly Report
May–July 1969.

GEAP-10028-32
Thirty-Second Quarterly Report
August–October 1969.

GEAP-10028-33
Thirty-Third Quarterly Report
November 1969–January 1970.

GEAP-10028-34
Thirty-Fourth Quarterly Report
February 1970–April 1970

GEAP-10028-35
Thirty-Fifth Quarterly
May 1970–July 1970

GEAP-10028-36
Thirty-Sixth Quarterly Report
August 1970–October 1970

GEAP-10028-37
Thirty-Seventh Quarterly Report
November–January 1971.

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

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

In addition, the following topical reports have been
issued:

GEAP-3287
Fast Oxide Breeder - Reactor Physics,
Part I - Parametric Study of 3000 MWe Reactor Core;
P. Greebler, P. Aline, and J. Sueoka
(November 10, 1959).

GEAP-3347
Fast Oxide Breeder - Stress Considerations
in Fuel Rod Design;
K. M. Horst
(March 28, 1960).

GEAP-3486
Fast Oxide Breeder Project -
Fuel Fabrication

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

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

GEAP-3487
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J. M. Cleveland and W. C. Cavanaugh
(August 15, 1960).

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P Greebler, B A Hutchins, and J R Sueoka
(March 9, 1961).

GEAP 3721

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

GEAP 3824

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G D Collins
(January 24, 1962)

GEAP 3833

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J M Gerhart
(November 1961)

GEAP 3856

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K P Cohen, M J McNelly and B Wolfe
(November 27, 1961)

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H. W Alter, G D Collins, and E L Zebroski
(February 1, 1962)

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

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G D Collins
(November 15, 1962)

GEAP 3885

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

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The RAPTURE Code,
J H Ferziger, P Greebler, M D Kelly,
and J Walton
(June 12, 1962)

GEAP 4028

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for Fast Ceramic Reactors,
H W Alter
(February 1, 1962)

GEAP 4058

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in Fast Reactor Fuels
R B Osborn and D B Sherer
(August 1962)

GEAP 4090

FORE A Computational Program
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P Greebler and D B Sherer
(October 1962)

GEAP 4092

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Large Fast Ceramic Reactors,
Effects of Improved Methods
and Recent Cross Section Information,
P Greebler and E Goldman
(December 1962)

GEAP 4130

Experimental Studies of Transient Effects
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Series I, UO_2 Irradiations,
J H Field
(November 15, 1962)

GEAP 4426

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of 565 MWe Fast Ceramic Reactor,
A G Silvestor
(April 1963)

GEAP-4271

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

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G. L. O'Neill, P. E. Novak, M. L. Johnson,
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(September 1963).

GEAP-4350

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B. C. Luci, D. R. Riley, and N. F. Friedman
(December 5, 1963).

GEAP-4410

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P. Greebler
(November 1963).

GEAP-4414

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K. P. Cohen
(November 1963).

GEAP-4418

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Pre-Irradiated Mixed Oxide (PuO_2 - UO_2),
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P. Greebler, B. A. Hutchins, and M. D. Kelley
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GEAP-4675

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G. L. O'Neill, J. Duffy, J. C. Gilbertson, F. W. Knight,
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(September 1965)

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J. Davies and P. Novak
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(January 1968)

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Implications of Recent Fast Critical Experiments
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GEAP-5744

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(March 1968).

GEAP 5748

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

GEAP 5761

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

GEAP 5570

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(February 1968)

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BISYN—PERT
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(April 1969)

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with Liquid Metal Coolant;
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GEAP-10054

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GEAP 13517

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J H Field
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B F Rubin
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C L Cowan
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D W Sandusky, A E Conti
(March 1971)

GEAP-13608

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D. Freeman, et al.
(September 1970)

GEAP 13620

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K E Gregoire, P E Novak, R E Murata
(March 1971)

GEAP 13637

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W G Meinhardt
(March 1971)

GEAP 13672

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BISYN, PERT, and BICYL Synthesis Codes,
M D Kelley
(April 1971)

APPENDIX B – LIST OF CONTRIBUTORS

TASK B—DEFECTED FUEL BEHAVIOR

Task Leader: C N Craig

Contributors:

F. L. Bamford	R E. Murata
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Task C-1—SAFETY EXPERIMENTATION (TRANSIENT OVERPOWER)

Task Leader: G R Thomas

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Task C-2—SAFETY EXPERIMENTATION (LOSS OF FLOW)

Task Leader: H C Pfefferlen

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Task E—FUEL PROPERTIES AND PERFORMANCE EVALUATION

Task Leader: B. F. Rubin

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Task F—FAST FLUX TESTS AND SUPPORT

Task Leader: J. D. Stephen

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Task G-6—CORE DESIGN ANALYSIS

Task Leader: P. R. Pluta

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Task G-10—SAFETY METHODS AND ANALYSIS

Task Leader: D. D. Freeman

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Task G-12—ANL SAFETY/CORE DESIGN GUIDANCE

Task Leader: D. B. Sherer

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Task L—CLADDING DEVELOPMENT

Task Leader: H. J. Busboom

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Task N—CORE PHYSICS ANALYSIS

Task Leader: B. A. Hutchins

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Task P—FUEL ELEMENT DESIGN ANALYSIS

Task Leader: F. E. Gelhaus

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