MASTER

SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

**THIRTY-NINTH QUARTERLY REPORT
MAY-JULY 1971**

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

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Thirty-Ninth Quarterly Report

May — July 1971

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LIST OF ABBREVIATIONS, SYMBOLS, AND ACRONYMS

Å	=	angstroms
AEC-RDT	=	Atomic Energy Commission Reactor Development and Technology
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
ELC	=	Extra Low Carbon
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
FORTRAN	=	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

GETR	=	General Electric Test Reactor
gm	=	grams
GMUG-2	=	Multigroup data file based on modified ENDF/B-II data
i.d.	=	inside diameter
in./in./h	=	inches per inch per hour
I.P.	=	integrated power
Kcal/mole	=	kilocalories per mole
k_{eff}	=	effective neutron multiplication factor
keV	=	kilo electron volts
kW/ft	=	kilowatts per foot
ℓ	=	angular momentum states
Lb_f	=	pounds force
Lb_m	=	pounds mass
LMFBR	=	Liquid Metal Fast Breeder Reactor
MARS	=	Computer code for core disassembly calculation
MeV	=	million electron volts
MeV-GE	=	General Electric adjusted fluence for EBR-II ⁽¹⁾
mg	=	milligram
msec	=	millisecond
mole	=	gram molecular weight
MWd/Te	=	megawatt-day per ton (metric, tonne, 2205 lb)
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)
R	=	Roentgen, unit of radiation dosage
S/A	=	subassembly
SEFOR	=	Southwest Experimental Fast Oxide Reactor
SS	=	stainless steel
(ST)	=	solution treated
SUR	=	strong unbroken ring
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 model
TREAT	=	Transient Reactor Test Facility
TRIAL	=	Computer code for testing accuracy of Three-Dimensional (TRI) (A)xial (L)inking
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
W/cc	=	watts per cubic centimeter
wt %	=	weight percent
watts/cc	=	watts per cubic centimeter
XPORT	=	a computer routine to prepare code for shipment to outside users

ZEBRA-3	=	critical assembly 3 in ZEBRA facility
ZPPR-2	=	critical assembly 2 in Zero Power Plutonium Reactor facility
ZPPR-3	=	critical assembly 3 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
μm	=	micrometers
ν	=	number of neutrons emitted per fission
σ_c	=	capture cross section
σ_f	=	fission cross section
$\bar{\xi}$	=	average logarithmic energy decrement

HIGHLIGHTS

- Intergranular attack at the cladding inside surface was less than 1 mil in three fuel pins irradiated in EBR-II at $\sim 1200^{\circ}\text{F}$ (maximum) cladding temperature to $\sim 55,000$ MWd/Te peak burnup.

Slight intergranular attack (~ 0.2 mil) occurred uniformly over the hotter region of the cladding inside surface in the annealed type 321 stainless steel cladding of pin F5M. A localized penetration of ~ 1 mil was found at the fuel-upper blanket interface.

- Destructive examination of the first in-pile experiment simulating loss of flow confirmed earlier indications that the consequences of complete sodium flow blockage are less severe than expected.

Visual observations of the C10X fuel specimen show virtually complete melting and gross slumping of the high power (~ 20 kW/ft) pellet, and melting past midradius in the adjacent (~ 18 kW/ft) pellets. The fuel pellets showed no signs of fragmentation or dispersion in spite of the presence of sodium, and post-irradiation diameter measurements of the sodium containment showed no significant deformation attributable to pressure loading.

- The second high-burnup EBR-II specimen apparently did not fail during a TREAT transient more severe than that which failed the earlier companion specimen.

The fission-gas puncture of the C4G capsule indicates that the C4G fuel specimen (previously irradiated to $\sim 50,000$ MWd/Te in EBR-II) did not fail during TREAT irradiation that resulted in an estimated $\geq 50\%$ of the fuel volume exceeding solidus temperature. This is in contrast to the earlier C4H test in which failure occurred during a less severe transient.

- The rate of center void formation in a fuel pin operating at 2500°C center temperature appears to be sufficiently rapid to permit fuel center temperature calculations on new fuel to be based on equilibrium fuel structures.

In-pile experiments to measure the rate of center void formation have indicated that center void formation occurs in less than 26 minutes for fuel with a pellet density of 88%.

- Over 100 tensile specimens (types 304, 316L, 321, 347, and Incoloy-800) were tested at 480° , 590° , and 700°C (900° , 1100° and 1300°F) after exposure in EBR-II to 4 to 7×10^{22} n/cm².

Greater elongation was measured at strain rates of 2 min^{-1} than at $2 \times 10^{-4} \text{ min}^{-1}$. Ultimate and tensile strengths were also dependent upon strain rates. The major mechanical property changes resulting from irradiation hardening at temperatures in the region of 480°C (900°F) apparently occur at fluences less than 3×10^{22} n/cm² ($E > 0.1$ MeV). The additional property changes produced by exposure to fluences double this value are comparatively small.

- An analysis of nineteen EBR II irradiated fuel pins indicates a swelling rate 1.7 times that previously incorporated in the BEHAVE-I model.

Assuming that steady-state fuel-cladding mechanical interaction had occurred in those pins having the minimum indicated available void at the end of life, a swelling rate of 0.38% per 10^{20} fissions/cc is calculated.

- Two-dimensional, diffusion theory calculations on the ZPPR-2 demonstration plant benchmark critical assembly given k_{eff} values within 0.3% of measured using modified ENDF/B-II data.

- A pulsed eddy current (PEC) cladding analyzer has been built and used successfully to locate discontinuities on the inner surface cladding of irradiated fuel pins.

The ~ 1 mil localized intergranular penetration observed at the cladding inside surface of pin F5M was detected by the PEC cladding analyzer during the nondestructive phase of the examination.

- An unencapsulated pin subassembly which reached 7.5 at. % peak burnup in EBR-II appears to be in excellent condition.

Visual examination of the 37 unencapsulated wire-wrapped pins of the F9C experiment revealed no indications of unusual behavior. Profilometer traces from five pins showed maximum diameter increases of 1.1 to 1.6 mils. All the F9C pins are clad with annealed 316SS.

- Void formation in irradiated types 316 and 347 annealed stainless steel was observed to virtually cease at a fluence of 4×10^{22} n/cm².

Transmission electron microscopy (TEM) examinations of additional cladding samples from the F2 series of fuel pin irradiations have confirmed that void formation in types 316 and 347 stainless steels irradiated at 525° to 545°C (978° to 1013°F) virtually ceases after an exposure of $\sim 4 \times 10^{22}$ n/cm², $E > 0.1$ MeV. The continued increase in swelling is due entirely to void growth.

- The BEHAVE-II fuel element performance code is now operational.

This model was used in the analysis of the HEDL-P-19 linear-heat-rating-to-melting test and is being employed in the analysis of the transient overpower TREAT irradiations.

- The major subroutine packages for the FREADM-2 fast reactor accident analysis program are ready for incorporation. These include improved versions of TOFF (Transient Overpower Fuel Failure), VOIDUM (Primary Loop Fluid Dynamics Model), and BUBBLE (Sodium Boiling and Channel Voiding Model).
- Fabrication of forced convection capsule B9D continued and is on schedule for a November 1971 irradiation.

Nine testing and operating procedures for capsule B9D were prepared, reviewed, and approved. A total of 30 procedures have been prepared and approved to date.

ABSTRACT

This report is the thirty-ninth 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 May through July 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, 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 are the:

- 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 May, June, and July, 1971, on the currently active tasks of this program is reported.

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

2. SUMMARY

2.1 TASK B—DEFECTED FUEL BEHAVIOR

Fabrication and assembly of capsule B9D continued on schedule. This capsule will investigate the performance of mixed-oxide fueled pins defected in high temperature (1050°F) flowing sodium after several days of full power operation.

A reaudit of forced convection capsule fabrication was completed by an AEC-RDT audit team. In general, the auditors' findings were favorable.

Work continued on the development of a heat pipe to remove heat from a forced convection capsule in case of a loss of pump power or shutdown of the pump after the capsule is removed from the reactor.

Fabrication of capsules for scoping tests to develop a self-actuating fuel pin defect mechanism for defected fuel tests in EBR-II was completed. The first test was initiated.

Design of capsule B9B which is to investigate defected fuel behavior at high burnup was continued.

Preparations for disposal of capsule B9C were completed.

Work on the design of capsule B10A was discontinued by agreement with AEC-RDT.

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

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

Based on the activity level of the capsule cover gas it appears that the C4G fuel specimen did not fail during TREAT irradiation slightly more severe than that of the failed C4H fuel pin (see subsection 4.2.2). The C4H fuel pin failed in two places in the upper quarter of the fuel column. Both failures were in the form of cracked cladding, with no apparent fuel loss. These preliminary C4G and C4H results indicate that the mode, seriousness, and threshold level of failure are less severe than anticipated on the basis of prior test results.

HEDL has supplied two fuel pins (PNL-2-2 and PNL-2-7) for use in the Series 4 test program (tests have been designated C4M and C4N, respectively).

The topical report covering the C6A-B tests (defected or sodium-bonded fuel) has completed review and is being printed.

A preliminary design basis specification for the C8A Mark-II loop has been completed.

The BEHAVE-2 fuel performance code (developed in Task P) has been modified to study the potential for fuel-cladding interference during the early portion of a typical TREAT test, prior to fuel melting in the test specimen. The objective is to determine if the typical initial TREAT test condition of zero power can result in cladding deformation during early fuel heatup, and thereby bias the remaining portion of the test.

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 and irradiation of the C10X loss-of-flow simulation test were completed and examination initiated. The C10X irradiation proceeded as planned with the cladding and a significant portion of the fuel reaching melt in the presence of sodium. Fuel slumping did occur but very little fragmentation was noted. The consequences of any fuel-sodium interaction which may have occurred were relatively mild.

The high heat flux heaters for the Mockup 1B out-of-pile loss-of-flow test have been received and are undergoing acceptance inspection.

The Design Basis Specification, detailed drawings, and a preliminary Product Quality Plan for the C10A Capsule were completed.

Work continued on improved multichannel heat transfer and fluid flow modeling capabilities required for analysis of transient undercooling experiments with the THTE program.

A meeting on the subject of high heat flux heaters was held with Oak Ridge National Laboratory (ORNL) personnel.

2.4 TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

Capsule E2R, designed for investigating rate of center void formation, initiated irradiation for time periods of 1, 10, 15, and 34 minutes during this quarter. The first evidence of center void formation was observed after a total of 26 minutes of irradiation at a calculated center temperature of 2500°C. This void was observed in 88% dense fuel pellets. No void was observed in the 94% dense fuel even after a total of 1 hour of irradiation. The results of this

Table 2-1
TASK B-IRRADIATIONS

Design Parameters													
		Fuel							Operating Conditions				Status
Capsule	Objective	Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in.-diam)	Composition			Cladding Material	Irradiation Facility	Cladding Surface Temp (°F)	Peak Rating (kW/ft)	Peak Burnup (MWd/Te)	
					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	800	14	14,500	Report in process
		0.216	2	0.010	83	25	1.93	347	GETR				
B8C (2 pins)	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel- sodium-clad compatibility.	0.144	2.2	0.006	86	25	1.97	316	V-Raft	1100	21	113,500	Examination completed
		0.204	2.25	0.017*	84	25	1.97	316	GETR			57,000	
B8D	Fuel-sodium compatibility versus density, stoichiometry and Pu content	0.214	1.75	0.0065	80-88	40	1.97	316	V-Raft	950	17.5	41,000	Examination completed
		0.214	1.75	0.0065	80-88	25	1.97		GETR			43,500	
		0.214	1.75	0.0065	80-88	25	1.94					41,400	
B9A (2 pins axially aligned)	Defected fuel pin. Fuel swelling versus stoichiometry in flowing sodium	0.216	5	0.005	89	25	1.99	316	Special	1019	23.4	8,600	Report in process
		0.216	5	0.005	89	25	1.95		V-Raft GETR	971	23.4	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	1015	20.0	~7300**	Test eliminated
		0.216	4.25	0.004	89.6	25	1.97	20% CW 316 Annealed	V-Raft GETR	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 in process

* Large gap required to accommodate sampling device
** Planned irradiation.

experiment indicate that the center void forms early in the life of a fuel element operating at temperatures of 2500°C.

Analyses of fuel, cladding and fission gases for tritium indicate that less than 1% of the produced tritium is retained within the fuel pin after irradiation.

Cs₂O was added to fueled capsules of Types 316, 321, and 347 stainless steel operating for 100 hours in a 550° to 1100°C thermal gradient. These experiments, involving (U-25 wt % Pu)O_{2,00}, indicated that Type 321 had no reaction while both Types 316 and 347 developed intergranular attack. The intergranular attack of Type 347 suggests that a chemical characteristic of titanium may account for the immunity of Type 321 and that carbide stabilization is not effective in resisting cesia attack.

2.5 TASK F—FAST FLUX TESTS AND SUPPORT

Data packages for the F11A experiment and for reconstitution of the 37-pin F9C experiment into the first of two 19-pin Mark E subassemblies for the run-to-failure program have been forwarded to the EBR-II Project. Analyses of the perturbations of the experiment expected if F9A is reconstituted into the Mark E subassemblies have been forwarded to the AEC (RDT). A report describing the damage to pin F9C-13 in the EBR-II hot cell was received from the EBR-II Project. Fabrication of the subassembly channel with grid spacers for experiment F9E continued. Approval-in-principle has been received for the F10C experiment, and redesign of the experiment is underway to focus on the new objective, which is to determine the extent of fuel-cladding chemical interaction. Preparation of the fuel powder for the F20 experiment is complete.

The calculated burnup using EBR-II methods of the five unfailed F3B pins being examined is up to 19% higher than the 7.7 to 8.7 at. % burnup measured by chemical analysis for neodymium. The fission gas release from the fuel in these pins was ~100%, based upon the measured burnup values but 88% based on the calculated values.

No evidence of significant intergranular attack has been found to date in either the annealed SS316 cladding of pin F5F or the 50% cold-reduced cladding of pin F5Q that operated at a calculated peak temperature of 1200°F. The pins have an unidentified zone between the fuel and cladding in the form of a rim of material (~1 mil thick) extending around the circumference. A slight intergranular attack of ~0.2-mil penetration was observed occurring uniformly over the hotter region of the cladding inside surface of the annealed SS321 cladding of pin F5M. A localized penetration of ~1 mil was found at the fuel-upper blanket interface in this pin.

The test results of the strength of the beam welds of the grid spacer for the F9E experiment indicate a wide margin over the < 5-pound maximum load expected during irradiation.

The 37 pins of the F9C experiment which reached 7.5 at. % peak burnup appear to be in excellent condition, according to the nondestructive interim examination at EBR-II.

A compilation of the fission-gas release data from all the Task F fast flux irradiations shows the trend of increased release with increasing burnup, with near 100% release observed at as low as 4 at. % burnup.

A pulsed eddy current cladding analyzer has been built and applied successfully for irradiated fuel pins as well as incoming tube inspection. The ~1 mil localized intergranular penetration observed later in the metallographic sections at the cladding inside surface of pin F5M was detected during the nondestructive phase of the examination.

The status of the fast flux and thermal flux experiments under this task is summarized in Tables 2-3 and 2-4.

2.6 TASK G-6—CORE DESIGN ANALYSIS

Detailed design drawings of the current reference conceptual FFTF grid-spaced driver fuel assembly have been completed.

Mechanical tests in support of the EBR-II LMFBR diagnostic irradiation test F9E were conducted, including a grid insertion and inspection test, a grid spacer push test, and a shipping load and vibration test.

A grid spaced 37-pin sodium flow test was designed and is now being fabricated.

The conceptual design of the FFTF diagnostic grid assembly test F9F planned for EBR-II irradiation resulted in the tentative selection of parameters on which to base the final design.

A final report on fuel element failure, coolant boiling, and operating safety limits received final review.

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	1.973	316SS
C4J		powder	13.45	powder	83.5	20.2	0.89	1.973	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.								
C4N	Comparison to C4M								
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

Table 2-2 (Continued)
TASK C-1-EXPERIMENTAL TESTS

Status								
Capsule	Fabrication	<u>Pre-irradiation</u>	TREAT		Examinations [†]			Report
			Calibration	Test	Nondestructive		Destructive	
					Before Transient	After Transient		
C4A C4B C4C C4D C4I C4J								GEAP-13562
C4E C4F	C C	C* C*	P P	P P	C C	P P	P P	
C4G C4H	C C	C* C*	P C	P C	C C	I C	P I	
C4K C4L	C C	C* C*	P P	P P	C C	P P	P P	
C4M C4N	C C	C C	P P	P P	P P	P P	P P	
C4X	C	NP	C	C	C	C	C	
C5A C5B								GEAP-13543
C6A-1 C6A-2 C6A-3 C6B-1 C6B-2 C6B-3	C C C C C C	NP NP NP NP NP NP	C C C C C C	C C C C C C	C C C C C C	C C C C C C	C C C C C C	In progress

Table 2-3
TASK F FAST FLUX TESTING PROGRAM IN EBR-II (a)

Group	Purpose and Major Parameters	Total Number of Pins in Group	Date In	Projected Date Out	Actual Date Out	Number of Pins in S/A	S/A	Position
FO	Low power swelling rate of high density fuel at various burnup levels	5	7/16/65		8/15/68	1	XG02	7A1
			7/16/65		8/ 8/70	2	XG03	7D1
			7/16/65		8/ 8/70	2	XG04	7B1
			9/14/70	4/13/72		2	XG03A	8D2
			9/14/70	3/12/73		2	XG04A	8A7
F1	Short time performance and proof test	6	5/ 6/65		5/23/65	6	XG01	4F2
F2	Medium power scoping irradi- ation. Major parameters include smear density, stoichiometry, cladding- coolant interactions and burnup	21	9/ 3/65		3/ 6/68	9	XG05	4C2
			9/ 3/65		2/20/67	12	XG06	4E2
			2/23/69		5/18/70	4	X050	4C2
F3A	Bare pin irradiation to evaluate fuel cladding and cladding-coolant interaction	16	8/15/68		6/30/69	16	X040	5B2
			10/ 4/69		5/18/70	16	X040A	5B2
F3B	Density and stoichiometry effects for encapsulated fuel pins	18	11/21/67		5/18/70	18	X027	4B3
F4	Low density fuel behavior at high heat ratings	9	5/ 9/66		6/28/67	9	X011	2F1
			8/15/68		12/24/68	7	X039	3C2
			1/10/70		8/ 8/70	7	X070	3E1
F5	Fuel performance at peak for cladding temperatures	19	5/28/69		8/ 8/70	19	X064	4F2
F6	Fuel-cladding interactions under accelerated swelling	4	3/24/66		4/24/69	4	X010	7F3
F7	Pre-irradiated axially restrained fuel prior to transient test. Main variable is fuel density	6	12/23/67		12/23/67	2	X015	4A2
			1/23/69		5/18/70	4	X050	4C2
F8A	Effect of density on fuel swelling rate	16	1/13/67		8/ 2/69	7	X019	6D2
			1/13/67		1/31/70	9	X020	6B5
			5/24/70	8/20/72		9	X081	7F6
F8B	Effect of void deployment on fuel cladding reaction	19	7/25/68		8/24/70	19	X036	7E1
			2/24/71	12/24/71		9	X117	8B2
			2/24/71	8/ 1/71		10	X118	8D7

Range of Average Pin Powers ^(b) kW/ft	Target Burnup at. % MWd/Te		Average Calculated Burnup ^(c) at. % MWd/Te		Peak Calculated Fluence ^(d) (nvt X 10 ⁻²³)		Status
					Total	E > 0.1 MeV	
7.76	3.91	35000	3.34	30199	0.276	0.199	Post exam complete
5.47— 5.92	8.95	80000	6.88	62247	0.593	0.432	Reconstituted into XG05A
5.44— 6.24	11.18	100000	7.62	68916	0.662	0.490	Reconstituted into XG04A
5.58— 6.23	8.95	80000	7.59	68681	0.659	0.479	In reactor
5.42— 6.19	11.18	100000	8.32	75249	0.729	0.537	In reactor
12.32—14.49	0.22	2000	8.21	1882	0.017	0.015	Reported in GEAP-5578
11.42—13.91	8.39	75000	6.69	60543	0.369	0.488	Post exam complete
8.48—14.83	5.59	50000	5.02	45451	0.422	0.362	Reported in GEAP-13549
10.73—12.86	11.19	100000	11.75	106358	1.034	0.887	Post exam in progress
12.87—14.65	5.62	50000	2.88	25959	0.236	0.200	Reconstituted into X040A
12.53—14.25	5.62	50000	5.17	46620	0.430	0.365	Post exam initiated
11.48—12.80	13.49	120000	8.86	79922	0.754	0.647	Interim exam
13.74—14.53	5.65	50000	3.64	32758	0.298	0.257	Reconstituted into X039
13.41—13.83	9.04	80000	4.85	43633	0.400	0.344	Reconstituted into X074
12.72—13.37	9.04	80000	8.25	74217	0.695	0.597	Interim exam
12.56—13.36	11.23	100000	5.71	51568	0.476	0.408	Removed from reactor
8.59— 9.34	5.59	50000	5.71	51645	0.480	0.391	Post exam complete
12.54—12.65	5.64	50000	5.40	48599	0.456	0.390	Terminated
11.91—12.25	5.63	50000	5.43	48914	0.449	0.384	Terminated
8.53— 9.15	5.62	50000	4.97	44777	0.412	0.347	Post exam complete
8.37— 9.11	11.25	100000	6.15	55398	0.524	0.438	Reconstituted into X061
6.78— 8.46	11.24	100000	8.07	72759	0.693	0.567	In reactor
4.99— 6.39	11.24	100000	4.73	42600	0.401	0.303	Removed from reactor
4.37— 5.58	6.62	50000	4.75	42781	0.408	0.299	In reactor
5.04— 6.28	5.62	50000	5.25	47307	0.447	0.336	In reactor

Table 2-3 (Continued)

Group	Purpose and Major Parameters	Total Number of Pins in Group	Date In	Projected Date Out	Actual Date Out	Number of Pins in S/A	S/A	Position
F9A	High power, high burnup, unencapsulated fuel pin irradiations	37	2/20/69 10/27/70	6/ 8/72	5/18/70	37 37	X043 X043A	4D2 6D3
F9B	Medium power, high burnup, unencapsulated fuel pin irradiations	37	5/23/69	11/ 6/72		37	X062	7F3
F9C	High power, high burnup, unencapsulated fuel pin irradiations	37	4/ 3/69 10/27/70	6/22/72	7/16/70	37 37	X056 X056A	5C2 6B2
F9D	Medium power, high burnup, unencapsulated fuel pin irradiations	37	4/24/69	7/22/73		37	X058	7B6
F9F	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 for cladding temperatures	37						
F20-1	Irradiation to probe operating limits	19						

- (a) All calculated power, burnup, and fluence numbers are based on the fission rates provided in the "Guide for Irradiation Experiments in EBR-II."
- (b) Values are the calculated axial average powers. Ratio of peak-to-average power is 1.125.
- (c) Burnup values are axial average values for the pin in the subassembly with the highest fuel burnup regardless of position in subassembly. All values are corrected for fuel depletion. Burnup values in MWd/t are based on a value of 192 MeV/Fission and calculated from data supplied by EBR-II.
- (d) Fluence values reported are calculated for the peak axial flux for the pin in the subassembly located closest to the center of the reactor core. Values are based on data supplied by EBR-II.

Range of Average Pin Powers ^(b) kW/ft	Target Burnup at. %	Burnup MWd/Te	Average Calculated Burnup ^(c)		Peak Calculated Fluence ^(d) (nvt X 10 ⁻²³)		Status
			at. %	MWd/Te	Total	E > 0.1 MeV	
12.06–13.36	11.25	100000	5.57	50189	0.464	0.398	Reconstituted into X043A
10.64–12.72	11.25	100000	6.62	59670	0.557	0.477	Out of reactor
7.99–10.48	11.24	100000	6.40	57673	0.538	0.451	In reactor
10.50–11.55	11.24	100000	5.67	51157	0.472	0.401	Reconstituted into X056A
10.20–11.71	11.24	100000	6.68	60207	0.561	0.476	Out of reactor
7.83– 9.63	11.24	100000	5.36	48375	0.451	0.378	In reactor

Fabrication in progress

Shipped to EBR-II

Shipped to EBR-II

Fabrication in progress

Fabrication complete

Approval-in-principle

Table 2-4
SUMMARY OF TASK F THERMAL FLUX IRRADIATIONS

Capsule Number		Fuel Design Parameters ^(a)					Operating Conditions ^(b)			Status
		Diameter (inch)	Length (inch)	Density (%TD)		O/M	Cladding Surface Temperature (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)	
				Pellet	Smeared					
F12A (E5F)	Linear fuel movement versus burnup	0.214	10.0	92.0	87.1	1.98	850	17	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	Examination Complete
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- and 15-mil cladding thickness)	0.219	2.0	92.0	91.0	1.98	1160	15	~100,000	Interim
F12Q (E2Q)		0.229	2.0	92.0	91.0	1.98	1160	15	~100,000	Examination

(a) In all capsules, the cladding is Type 316 stainless steel and the fuel is (U-0.25 Pu)O₂.

(b) The irradiation is conducted in the GETR.

2.7 TASK G-10—SAFETY METHODS AND ANALYSIS

The major subroutine packages for FREADM-2 are ready for incorporation. These include:

TCFF - Transient Overpower Fuel Failure Model

VOIDUM - Improved Primary Loop Fluid Dynamics Model

BUBBLE - Improved Sodium Boiling and Channel Voiding Model

In addition, disassembly calculation packages from other sources are operational and are being coupled to FREADM. These include:

MARS - Updated version of R-Z modified Bethe-Tait (already coupled)

VENUS - Lagrangian hydrodynamics in R-Z geometry

Exploratory analyses are in progress in the areas of core disassembly and transient undercooling.

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 has been assigned to work at ANL to provide improved liaison.

2.9 TASK L—CLADDING DEVELOPMENT

Transmission electron microscopy (TEM) examination of Type 347 cladding from the F-2 series of fuel pin irradiations has confirmed that the continued swelling of both Type 316 and Type 347 as the fluence increases from 4 to 7×10^{22} n/cm², $E > 0.1$ MeV-GE, is due primarily to the growth of existing voids rather than to further void nucleation. Comparison of TEM samples F2T-8 and F2G-H after exposures of 3.9 and 7.0×10^{22} n/cm², $E > 0.1$

MeV-GE, respectively, at comparable temperatures shows that the void number density was the same ($\sim 1.5 \times 10^{15}$ voids/cm³) while the mean void diameter increased from 250 to 470 Å.

The void morphology observations made on the F2 cladding materials suggest that the effect of an externally applied stress at 500°-550°C (932°-1022°F) in these claddings does not cause grain boundary cavitation (creep voids) or enhance growth but appears to increase the extent of intragranular precipitation and consequently to increase the void number density.

Tensile tests were performed on austenitic stainless steels (Types 304, 316L, 321, and 347) irradiated in EBR-II to a peak fluence of 6.8×10^{22} E>0.1 MeV-GE. Increases in tensile properties and decreases in ductility were observed. These changes are attributed to irradiation-induced hardening and embrittlement. At test temperatures of 593°C (1100°F) and 704°C (1300°F) and over the strain rate range of from 2 to 2×10^{-4} min⁻¹, all measured mechanical properties — 0.2% yield strength, ultimate tensile strength, total elongation, and uniform elongation — increase with increasing strain rate.

Fabrication of three L-16D capsule inserts has been initiated. These capsules will contain highly irradiated cladding alloy samples for reirradiation to higher fluences in EBR-II.

An evaluation of samples of Type 316M 20% CW FFTF fuel clad tubing is in progress. Chemical analysis of this tubing showed all elements were well within the ranges specified by GE specification 22A1793 except boron. The boron level was determined to be 0.0035 compared to the specified 0.003 maximum.

The current status of the Task L irradiation program is summarized in Table 2-5.

2.10 TASK N—CORE PHYSICS ANALYSIS

Two-dimensional, diffusion theory calculations on the ZPPR-2, demonstration plant benchmark critical assembly, give k_{eff} values only 0.3% high, using modified ENDF/B-II data.

A test computer code, called TRIAL, for evaluating three-dimensional flux linking, has been programmed and checkout has been started.

A 29-group generalized nuclear data file based on modified ENDF/B-II data has been generated for use with the TDOWN code. The modifications are similar, but not identical, to those being considered by CSEWG for ENDF/B-III.

Computations for defining a Doppler coefficient benchmark problem using SEFOR results indicate that a two-dimensional model will be required.

Data testing performed in conjunction with CSEWG indicate that the modified ENDF/B-II data produce better comparisons with experiment for several critical benchmark problems. The k_{eff} values are all within 1% of the measurement and a general improvement in computed fission ratios was noted.

Topical reports on *Implications of Nuclear Data Uncertainties to LMFBR Design* and on the SYN flux synthesis and fuel cycle code have been issued. The SYN code has been sent to the Argonne Code Center. A user's manual for the ENDRUN-2 code has been submitted to the AEC for review and the final draft of the TDOWN report is nearing completion.

A draft of specifications for a standard input card format and card sweeper routine was completed and sent to other members of the Card Format Subcommittee of the *de facto* Committee on Computer Code Coordination.

2.11 TASK P—FUEL ELEMENT DESIGN ANALYSIS

The advanced fuel element performance code, BEHAVE-II, is operational and is being used in the analysis of experimental tests.

Nineteen fuel pins were considered in an analysis of fuel swelling which indicates that the rate may be higher than had previously been assumed in BEHAVE-I, and that the low burnup data and the diametral increase for the high burnup fuel pin F2G cannot both be explained by this single swelling rate.

A comparison was made of recent long-term stress rupture test data with a theoretical expression for the diametral ductility penalty to be expected as the result of tubing wall defects which shows that safe penalties (over-estimates) can be analytically predicted. The analysis indicates that defects less than 10% of the wall thickness are not associated with a diametral ductility penalty.

Table 2-5
TASK L—CAPSULES IN EBR-II

Group	Objective	Total Pins	Number of Specimens		Date In	Date Out	Row S/A	Core Position	Target Fluence	S/A Average Fluences***		Reactor Exposure (MWd)	Status
			Tube	Flat					(n/cm ² X 10 ²¹) Total	(n/cm ² X 10 ²¹) 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	0.4	0.33	0.29	9,317	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	0.2	0.19	0.16	5,355	Reported in GEAP 10066
L-4	PI Mech Prop Eval of Type 304, 316, 321, 347, & Incoloy 800	5	25	50	07/17/66	04/10/67	2-X014	2D1	0.2	0.16	0.14	3,674	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	0.2	0.28	0.24	7,894	Reported in GEAP-10028-31
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	0.2	0.29	0.26	8,399	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-XJ35	4C2 3C2	0.75	0.54	0.46	12,632 2,532	Post Exam Complete
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	0.75	0.90	0.75	12,632 11,706	Post Exam Complete
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	1.0	0.64	0.55	19,280	Post Exam in Progress
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	2.0	0.78	0.67	10,687 2,334 9,851	Post Exam in Progress
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 6-X080	3C2 6E5	0.3	0.23	0.19	2,532 2,525	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	2.0	0.78	0.67	19,280 2,525	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	0.09	0.08		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††		Fabrication 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 defected fuel behavior and its consequences to reactor design and safety. Defective fuel elements can be expected in any reactor, no matter how elaborate the 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 fuel pins having breaches in the cladding.

The specific purpose of Task B is to (1) determine analytically and experimentally the consequences of failure of a fuel pin under normal or near normal LMFBR operating conditions on the operation of the reactor, (2) provide experimental and analytical data to permit selection of fuel pin design parameters and operating conditions which limit the consequences of fuel failure, (3) provide experimental data on failure kinetics to permit modeling and prediction of fuel failure behavior for the projected range of LMFBR operating conditions, and (4) 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 high burnups. The capsule will contain one fuel pin to be defected at power after an exposure of approximately 25,000 MWd/Te. Design work on capsule B9B was resumed (It had been interrupted in February 1971 to permit increased effort toward preparation and implementation of quality control and quality assurance procedures required for capsule B9D.)

3.2.2 Forced Convection Capsule B9C

Capsule B9C is designed to investigate the performance of defected substoichiometric (O/M=1.97) mixed-oxide fuel operating at typical LMFBR conditions in flowing sodium.

Work on disposition of the capsule was completed. Preparation and approval of disassembly and nuclear safety procedures required for possible salvage of the main capsule components (fuel pin, flowmeter, and pump helix) was completed. Power to the capsule heater jacket was shut off and the sodium in the capsule frozen. Prior to sectioning of the capsule for salvage, it was x-rayed and the fuel pin was found to be defected. Defection of the fuel pin negated any economic advantage to be obtained from salvage of capsule components since it was necessary to assume that all the parts within the capsule primary containment were alpha contaminated. It was therefore decided, with AEC-RDT concurrence, that the capsule be disposed of without salvage. A shipping container was fabricated for transporting the capsule to its burial site. The capsule was loaded in the container and prepared for shipment.

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. Effort was continued on the preparation, review, and approval of testing and operating procedures for capsule B9D. Nine additional procedures were approved to bring the current total to 30. Some 14 additional procedures are under review.

Fabrication of capsule B9D is presently on schedule for irradiation in November 1971. Procurement of all fabricated parts and materials is now complete. Work during this quarter consisted of fabrication of new subassemblies, and the disassembly, inspection, and reassembly of subassemblies fabricated prior to the suspension of fabrication in August 1970. All fabrication and inspection operations are being conducted in accordance with approved written procedures.

Figure 3-1 shows the subassembly sequence for the capsule subassemblies. Completed subassemblies are indicated by a C adjacent to their titles. The approximate status of subassemblies still under fabrication is indicated by the numbers adjacent to their titles, e.g., 50% means the subassembly is approximately one-half completed.

*Contributors are listed in Appendix B.

3.2.4 Forced Convection Capsule B10A

Capsule B10A was to be the first in-core, multiple-pin, forced convection capsule. The purpose of the capsule was to be to investigate pin-to-pin failure propagation for pins operating at typical LMFBR operating conditions. A feasibility and cost study has shown that the technical objectives of this subtask can be attained at lower cost by other routes, and hence work on this capsule was terminated with the full concurrence of AEC-RDT.

3.3 TECHNICAL SUPPORT FOR IN-PILE DEFECTED FUEL TESTING

3.3.1 Heat Pipe Development

Developmental testing was continued on a radial heat pipe for use as a "thermal switch" to prevent overheating of the fuel pin in a forced convection capsule in the event of failure or shutdown of the coolant pump when a capsule is removed from the reactor. Procurement and fabrication of parts for a test of a full-size heat pipe was completed. The test apparatus was assembled and is undergoing checkout.

3.3.2 Defect Development

Work continued on scoping tests for a self-actuating defect mechanism for defecting fuel pins. Pins with the automatic defecting mechanisms will be tested in EBR-II. In the scoping tests, stainless steel* capsules containing UO_2 and potentially corrosive chemicals (CsOH , CsMoO_4 , and AgF) will be heated to temperatures of 800° and 900°F . The capsules will be pressurized to approximately 400 psi (pressure at temperature) and examined periodically for evidence of failure. The tests will be run for a maximum of 500 hours. At the termination of the tests, the capsules will be examined and the most promising compound(s) selected for further testing. Fabrication of the test capsules, loading of the UO_2 and chemicals into capsules, and sealing of the capsules was completed. The 800°F test was initiated.

3.4 REAUDIT OF FORCED CONVECTION CAPSULE FABRICATION BY AEC-RDT TEAM

A reaudit of forced convection capsule fabrication was carried out by an AEC-RDT audit team on May 17-18, 1971. The reaudit was based on a previous audit in October 1970. The general consensus of the audit team was that significant improvements in the fabrication facilities, documentation of operating procedures, and quality control and quality assurance of the fabrication operations had been made since the October 1970 audit.

*Type 316, 20% cold worked.

Figure 3-1. Status of B9D Capsule Fabrication

4. TASK C-1—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, the 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 (PNL-2-2 and 2-7), which operated at a lower power level of ~10 kW/ft.

4.2.1 Capsule Fabrication, Assembly and TREAT Irradiation

The damaged heater circuit of the C4G capsule¹ was repaired and the capsule successfully completed TREAT irradiation.

G. Thomas visited the HEDL Laboratories to initiate the technical information exchange program centered on the GE TREAT irradiation of two HEDL FFTF low-power fuel specimens (previously irradiated in EBR-II). The initially selected specimens, PNL-2-14 and PNL-2-15, were examined, and it was found that PNL-2-15 has significant axial misalignment (permanent offset deformation) in the region of the fuel column. The form of the misalignment is an S-shaped bend in the fuel region, with a maximum deviation of ~50-60 mils off fuel centerline. The misalignment is of such magnitude that the pin is unacceptable for use in the Task C-1 Series 4 capsule design. Ensuing discussions with HEDL have led to the substitution of two 0.250-inch diameter fuel specimens in place of the previous 0.238-inch diameter specimens.

These two HEDL-supplied fuel pins, PNL-2-2 and PNL-2-7 (redesignated C4M and C4N, respectively), have arrived at Vallecitos Nuclear Center, and have been visually inspected. The PNL-2-2 pin (C4M) contains solid mixed-oxide fuel and appears to have no notable bow. The PNL-2-7 pin (C4N) contains annular mixed-oxide fuel and has an ~1/2-inch bow over its length. Such a bow should not cause capsule assembly problems and both pins will be acceptable replacements for the original 0.238-inch diameter pins.

The basic C4 capsule design has been modified to adapt to the HEDL FFTF fuel specimens and to take into account experience gained in the assembly, testing, and examination of the C4X, G, and H capsules. The primary features of the redesign include: (1) elimination of the dependency upon potting epoxies for pressure vessel head integrity through use of high-pressure electrical connectors; (2) elimination of the capsule remote closure weld through use of a metal-to-metal seal high-pressure threaded fitting; and (3) replacement of the existing pressure transducer system with a system less sensitive to lead-line capacitance and temperature.

4.2.2 Post-Irradiation Analysis

Fuel specimen C4X has been profilometered, sectioned, and selected pieces mounted for detailed metallographic study. Measurements from polished transverse sections indicate a molten fuel volume of ~70%. One of the more interesting features noted during the examination was the extrusion of molten UO_2 into the ~8-mil diametral gap of the two lowest pellets of the upper UO_2 blanket, another example of the mobility of molten fuel (see Figure 4-1).

Profilometry of the cladding in the region of the C4X fuel column indicated ~3-mil average diametral deformation with an additional ~1-mil fuel-pellet interface ridging overlay.

*Contributors are listed in Appendix B

Figure 4-1. Task C-1 Zero-Burnup Specimen C4X, Molten Fuel Movement into Pellet Cracks and Pellet-Cladding Gap in the Blanket Region (Upper End of Fuel Column)

Fuel specimen C4H was removed from its capsule and visually inspected. Two locally adjacent areas of breached cladding were discovered, both located in the upper quarter of the fuel column. The breaches were both in the form of localized cracks $\gtrsim 20$ mils wide by $\gtrsim 1$ inch in length (Figure 4-2), and there was no apparent associated fuel loss. The C4H fuel specimen had been previously irradiated to $\sim 50,000$ MWd/Te burnup in EBR-II prior to TREAT irradiation, and the current best estimate of the TREAT-induced molten fuel volume is $\lesssim 40\%$ (defined as the aerial fraction exceeding solidus temperature). This figure will be verified or revised during destructive examination.

A fission gas puncture of the C4G capsule indicates that the C4G fuel specimen (also previously irradiated to $\sim 50,000$ MWd/Te in EBR-II) did not fail during TREAT irradiation that resulted in an estimated $\gtrsim 50\%$ molten fuel volume. Absolute confirmation of fuel melt volume and failure/nonfailure will have to await deencapsulation of the specimen and subsequent destructive examination; but it should be noted that the first positive indication of failure of the C4H fuel specimen (identical companion pin) was high activity in the capsule cover gas detected during the fission gas puncture of the C4H capsule.

Even if the melt volumes are less than 40%–50% for C4H and C4G, the mode, seriousness, and threshold level of the failure are less severe than was anticipated on the basis of previous test results.

A dial-gage profilometer has been fabricated for use on failed C4 fuel pins being examined in the large α -cell at Vallecitos. In addition, a Neutrograph holder has been designed and fabricated that will allow these failed pins to be removed from the α -cell and Neutrographed.

Both systems will be used for the first time on the C4H fuel specimen.

Based on the results of the destructive examination of the C4X fuel specimen (measured $\sim 70\%$ molten fuel volume vs. original prediction of $\sim 85\%$), the thermal analysis model of the C4 inner capsule has been modified to more properly account for γ -heating of the capsule. The first results using this revised model indicate that γ -heating will account for 5%–10% of the total indicated power of the capsule.

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 $\text{UO}_2\text{-PuO}_2$ during identical transient overpower conditions.

A topical report summarizing the results of the initial Series 6 tests (Capsules C6A and C6B - defected or sodium-bonded fuel) has completed internal review and is being prepared for publication.

4.4 MARK II LOOP TESTS

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

A preliminary Design Basis Specification for the test section assembly for the initial Mark II loop test in TREAT has been prepared and is in the process of internal review.

4.5 MODEL DEVELOPMENT

The BEHAVE-2 fuel performance code, under development by Task P, is being modified under Task C-1 to accept inputs characteristic of transient overpower conditions. BEHAVE-2 calculates stresses and strains in fuel and cladding in (r, z) coordinates as a function of time. Effects considered include fuel cracking, axial flow of molten or highly plastic fuel, and fuel-cladding slippage or locking. Post-failure effects are not considered. Modifications to the code have included provision for transient heat transfer either by a one-dimensional calculation or by employing results calculated by the THTD code.²

The initial use of the modified BEHAVE-2 code has been to model the TREAT irradiated pin C4A.^{3,4,5} In problems run to date, the fuel diameter and density have been assumed to be constant and equivalent to the average C4A values. The axial power profile is believed to have been nearly flat so calculations for this configuration yield answers that are independent of location along this axis.

Figure 4-3 shows the calculated closure of the fuel-cladding gap as a function of time after the initiation of the transient. BEHAVE-2 considers that the fuel will crack radially into a number of pie-shaped segments at the first application of power. The actual positions of the segments within the cladding is indeterminant when there is radial clearance between the fuel segments. BEHAVE-2 assumes that the fuel segments reside in contact with cladding inside surface i.d., but results may be effectively interpreted in the more conventional manner of gap closure. The ability of the code to predict fuel-cladding gap will be utilized in predicting a time-dependent gap conductance for heat-transfer calculations.

Figure 4-2. Task C-1 Pre-Irradiated Fuel Specimen C4H, Exterior Details of Breached Cladding Regions

Figure 4-3. Task C-1 Fuel Specimen C4A, Diametral Fuel-Cladding Gap as a Function of Time from Initiation of Transient

Figure 4-4 shows the increase in length of the fuel column and cladding as a function of time after the transient initiation. Locking of the fuel to the cladding at the instant of gap closure precludes further relative movement of the fuel and cladding and henceforth the fuel expands axially at the rate of the cladding. The initial modeling assumption of constant fuel diameter is somewhat at odds with the actual case in which a relatively uniform diameter lower and middle fuel column was topped by a larger diameter pellet capped by two particularly small pellets. The result was a large molten-fuel filled axial gap located directly above the larger pellet.^{3,4,5} This situation will be simulated in future runs by a 3-axial region calculation which should produce somewhat less fuel column extension than is indicated in Figure 4-4.

The calculation of cladding diameter increase is highly sensitive to the magnitude chosen for the "inviscosity temperature" or temperature at which the fuel loses significant resistance to shear deformation. Estimation of this parameter relies upon extrapolation of data obtained at much lower strain rates and lower temperatures. Because of this, calculations performed to date have assumed an inviscidity temperature equal to the melt temperature. This assumption yields $\Delta D/D$ predictions of 0.7% compared to measurements of 0.4%. Peak plastic strain is predicted to occur before fuel melting and before the time of failure estimated from thermocouple data. However, the microstructure of C4A indicates partial hot-pressing of fuel at temperatures lower than melting (in the region immediately outside of the molten fuel boundary), indicating a lower inviscidity temperature than assumed. Use of a lower temperature would yield results in closer agreement with data and cases employing a best estimate value of this parameter will be run in the future.

Figure 4-4. Task C-1 Fuel Specimen C4A, Change in Fuel Column and Cladding Length Relative to Unirradiated 20°C Condition as a Function of Time from Initiation of Transient

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 LMFBF 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 malfunction to neighboring pins, and the time characteristics of signals (such as temperature and flow) are of 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 in the presence of sodium to verify these models and assess any secondary events

The purpose of this task is to conduct a program of such tests utilizing both stagnant and flowing coolant capsules in in-pile as well as electrically heated out-of-pile sodium loops.

5.2 MOCKUP 1B

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

Seven high heat flux heaters to be used as fuel rod simulators have been received and the receiving/acceptance inspection has been initiated. Three of the seven heaters will be selected for use in the Mockup 1B test.

The Design Basis Specification for Mockup 1B has been approved and issued. The sodium fill system has been checked out and operated for a short period of time in preparation for sodium filling of the test capsule. Drawings for modifications to the fill system capsule holder to permit mounting and operating the capsule pump during cleanup are being reviewed. This modification will enhance the cleanup operation and reduce the possibility of residual local accumulations of sodium oxide.

An analysis of the dynamic void behavior for the Mockup 1B loss-of-flow test has been initiated using the BUBBLE code.

5.3 CAPSULE C10X

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

Fabrication of the C10X capsule was completed and irradiation in GETR was initiated on April 30, 1971. The loss-of-flow simulation transient was completed on May 5, 1971. The total full-power operating time for the test fuel pin was approximately 60 hours. The 2-inch long, 1/4-inch diameter UO_2 fuel pin operated at a nominal 18 kW/ft with a single fully enriched pellet located near the lower end of the fuel column operating at approximately 20 kW/ft. The high-power pellet so located resulted in a hot spot on the cladding which was intended to establish the point of boiling initiation. The transient portion of the test followed this period of steady-state operation and consisted of interrupting the radial heat transfer from the fuel pin and its associated sodium and allowing the fuel pin to operate at full power for a period of 4.5 seconds before terminating the test with a reactor scram. The predicted temperature response during the transient is shown in Figure 5-1. From this, it can be seen that sodium boiling was expected to occur after ~ 1 second, cladding melting after ~ 2 seconds and fuel melting after ~ 4 seconds.

*Contributors are listed in Appendix B

Figure 5-1. Capsule C10X Transient Radial Temperature Profiles

It was estimated that at the time of reactor scram an excess of 40% of the fuel would have reached the melting temperature. Following this irradiation, the capsule was removed from the GETR Trail Cable facility and neutron radiographs were taken. These neutron radiographs (Figure 5-2) revealed that the fuel pin cladding had melted over the entire length of the active fuel but that the fuel column was still intact. In addition, no significant indications of structural deformation resulting from pressure loading could be found.

The destructive examination included examination of each of the containment tubes to determine if a noticeable deformation had taken place which could be attributed to internal pressure loading. The sodium containment tube surrounding the fuel pin and its associated sodium was the only containment which was damaged during the transient. This containment was breached in two places, one by contact with molten fuel and the other by contact with molten cladding. The extent of the damage is shown in Figure 5-3. Pre- and post-related diameter measurements did not reveal any noticeable pressure-related deformations even though thermocouple measurements indicated temperatures in excess of 2250°F were reached on the outside surface of this nominally 7/16-inch o.d. by 0.012-inch wall tube. The maximum pressures which could be contained within a tube of these dimensions and not exceed yield during the transient is shown in Figure 5-4. In this case, the temperature indicated by the thermocouple is taken as the average temperature of the tube and the existence of the hole is neglected. It is not intended that this represent a rigorous evaluation of the pressures associated with possible sodium-fuel interaction but it does provide a datum point from which subsequent capsule designs can be referenced.

As was indicated previously, the 1/4-inch o.d. by 0.015-inch wall 316SS fuel pin cladding material was melted over the length of the active fuel. The appearance of the fuel and cladding is shown in Figures 5-5a and 5-5b both before and after cleaning with acetic acid. This figure shows the relationship between the fuel, fuel pin cladding and sodium containment. Following removal of the fuel pellets, the molten cladding material was found to have resolidified on the inner surface of the sodium containment as shown in Figure 5-6. The fuel column showing the degree of melting and slumping associated with the high-power pellet is shown in Figures 5-7a and 5-7b. Finally, the degree of melting in both the high-power and adjacent pellet was found to agree well with the predicted behavior as interpreted from the metallographic sections of the fuel shown in Figures 5-8a and 5-8b.

At this point in the investigation the fuel does not appear to exhibit a strong tendency toward fragmentation under the conditions of this test. Furthermore, the consequences of any molten fuel-sodium interaction which may have occurred seems to be relatively mild. Work is continuing in an attempt to provide more quantitative information on the results of this first simulated loss-of-flow experiment.

5.4 CAPSULE C10A

Capsule C10A was the first in a series of planned in-pile loss-of-flow tests utilizing the forced convection capsule (FCC). The purpose of this series of tests was to investigate the complete loss-of-flow accident from initial flow reduction, through voiding, and into the area of secondary events from sodium-molten fuel interactions. In these tests, every effort was to have been made to include the influence of fuel, blanket, and plenum, as well as flow restrictions and boundary conditions. Work on this series of tests was terminated as of July 1, 1971, by AEC-RDT directive.

The Design Basis Specification, detailed drawings and a preliminary Product Quality Plan have been generated. These documents will be archived pending a final decision as to their disposition.

5.5 MODEL DEVELOPMENT

The THTE program is being utilized to provide the multichannel heat transfer and fluid flow modeling capabilities required for the analyses of single as well as multipin transient undercooling experiments. The THTE program is the fifth generation program in the THTE 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 continued on the incorporation of the binary gas and thermal-expansion options into the THTE program. During this period, 20 subroutines were modified with another 10 required to complete the project. The first round data processing functions have been completed; the routines required for second round data processing have been modified and are being tested. The program is now able to read all the new data types, stack the data, and print the input edit. Testing of the data checking routines is in progress.

5.6 LIAISON

Plans and results of this program are interchanged with those of other organizations concerned with safety. A meeting was held at GE on July 8, 1971, with ORNL personnel to discuss the ORNL heater development program. Information was also exchanged on the examination, handling, and operation of high heat flux heaters of the type planned for Mockup 1B.

Figure 5-2. Capsule C10X Neutron Radiograph

Figure 5-3. Capsule C10X Sodium Containment Showing Damage Resulting from Contact with Molten Fuel

Figure 5-4. Internal Pressure Required for Yielding C10X Sodium Containment

Figure 5-5a. Capsule C10X Fuel Column Showing Sodium Residue Before Cleaning

Figure 5-5b. Capsule C10X Fuel Column After Cleaning

Figure 5-6. Capsule C10X Sodium Containment Following Removal of Fuel Showing Both Intact and Melted Cladding

Figure 5-7a. Capsule C10X Fuel Column Following Removal from Sodium Containment (Grossly Melted Fuel is from High-Power Pellet)

Figure 5-7b. Capsule C10X Fuel Column Showing High-Power Pellet and Molten Core of Adjacent Pellet

Figure 5-8a. Capsule C10X Fuel Section (Average Pellet)

Figure 5-8b. Capsule C10X Fuel Section (High-Power Pellet)

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 reactions is being studied in irradiated samples.

6.2 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR

6.2.1 Central-Hole Formation at Reactor Startup — Capsule E2R

This capsule, containing 88% and 94%TD mixed oxide pellets, is intended to investigate fuel restructuring (time for center-hole formation) at reactor startup rates of 1 kW/ft/min and center temperatures of 2500°C. The capsule has been irradiated for periods of 1, 10, 15, and 34 minutes and neutrographs have been taken after each exposure. Central holes were observed in four of the 88%TD pellets of the lower tier after the third exposure (a total of 26 minutes). These ranged in diameter from 20 to 30 mils. No holes were detected in the 94%TD pellets in the fuel rod center region or in the 88%TD pellets at the top of the fuel column. The capsule's fourth irradiation cycle brought its total exposure to 60 minutes. The neutrograph taken after 60 minutes of irradiation (Figure 6-1) showed a central void in five of the six lower pellets of 88% theoretical density and in four of the seven pellets in the upper tier of the same density. No void was discernible in any of the 20 pellets of 94% theoretical density.

6.3 ELECTRON MICROPROBE EXAMINATION

6.3.1 Microprobe Examination of Sample 682 from Pin F3B-3 (~5.5 at. % Burnup)

A longitudinal specimen from this pin, which exhibited substantial intergranular fission product attack was described in Reference 1. Additional microprobe analysis of this sample indicated that oxygen and molybdenum penetrated into the cladding grain boundaries to a depth comparable to the depth of the intergranular attack. This reacted grain boundary was observed to be devoid of iron, chromium, and nickel, and the cesium and tellurium were confined to the fuel-cladding interface.

An additional area of pin F3B-3 showing intergranular penetration was examined. This same depth of penetration was brought out in the negative specimen current images and indicates an altered grain boundary chemical composition to this depth. This is shown in Figure 6-2. The iron, chromium, and nickel x-ray images of this same area are shown in Figure 6-3. The attacked grain boundaries of the iron image are most clearly delineated. They indicate the absence of this element in the reacted area. The figure also shows that the boundaries are depleted in chromium. A dark spot within a grain boundary of the iron x-ray image noted in Figure 6-3 was observed. This area is also obvious in the chromium and nickel views as well as in the photomicrograph of Figure 6-2. These observations indicate that this area exhibited less than expected levels of iron, chromium and nickel. In Figure 6-4, however, which illustrates the distribution of molybdenum, cesium, oxygen and tellurium, in this region the spot appears intense, indicating concentrations of cesium and molybdenum. Cesium was also observed in the grain boundaries. Oxygen penetrated to a lesser extent while the tellurium extended inward only as far as the open grain boundaries (~2 grains in depth).

Another region of this specimen was examined. It showed more matrix attack and shallower intergranular attack than the region described above. The matrix reaction product consisted of uniformly distributed cesium, molybdenum, tellurium, iron, and oxygen, with nickel concentrated closer to the fuel. In contrast to the observations of Figure 6-3, no chromium was present. Intergranular penetration in this specimen was limited to ~2 grains in depth. In contrast to previous findings, chromium was concentrated within the grain boundaries to the depth of attack. This is illustrated in Figure 6-5. In addition, the attacked boundaries were depleted in iron and nickel. Cesium, oxygen, and tellurium were present within the grain boundaries to the depth of penetration. No molybdenum was found within the reacted boundaries. This is the first observation, under this program, where chromium was not observed in the matrix reaction phase and was observed in the grain boundaries.

*Contributors are listed in Appendix B

Figure 6-1. Neutron Radiograph of E2R Fuel Pin After a Total of 60 Minutes of Irradiation

PHOTOMICROGRAPH OF INTERGRANULAR
PENETRATION OF CLADDING
A682-15, 16, 17 (500X)

CLADDING

REACTION
ZONE

FUEL

NEGATIVE SPECIMEN CURRENT
IMAGE
IMA52B, 50A, 51A, 51J (500X)

Figure 6-2. Photomicrograph and Negative Specimen Current Image of Intergranular Penetration of the Cladding of F3B-3

Figure 6-3. Iron, Chromium, and Nickel X-Ray Images of Region of Intergranular Fission-Product Penetration Shown in Figure 6-2.

MOLYBDENUM

IMA52E, 50E, 51E

CESIUM

IMA52C, 50H, 51P

TELLURIUM

IMA52D, 50F

PHOTOS REDUCED TO
85% FOR REPRODUCTION

OXYGEN

IMA52F, 51F

500X

CLADDING INSIDE SURFACE

Figure 6-4. Molybdenum, Cesium, Tellurium, and Oxygen X-Ray Images of Intergranular Penetration Area in F3B-3 (Section 682, Area 15-17)

NEGATIVE SPECIMEN CURRENT IMAGE

IMA59A

Cr X-RAY IMAGE

IMA59D

O X-RAY IMAGE

IMA59K

Cs X-RAY IMAGE

IMA59
800X

Figure 6-5. Negative Specimen Current Image and Chromium, Oxygen, and Cesium-X-Ray Pulse Images of F3B-3

Metallographic examination of the interface between the fuel and insulator indicated the presence of a two-phase structure. Microprobe analysis of this phase indicated uranium, cesium, and oxygen. Areas of UO_2 were also observed within the second phase. The matrix phase was UO_2 .

6.3.2 Microprobe Examination of Unirradiated Specimen

A sample of extra low carbon type 304 stainless steel heat treated in the Cs_2O -loaded thermal gradient capsules described in subsection 6.4.2 was examined. This sample showed chromium buildup in the affected grain boundaries (Figure 6-6), and iron depletion similar to that noted in the sample of F3B-3, (Figure 6-5). This out-of-pile test operated at $\cong 690^\circ\text{C}$ for 100 hours.

6.3.3 Calibration of Oxygen Standards for Quantitative O/M Determination of Mixed-Oxide Fuels

Although the variation in oxygen between $(\text{U-25 wt \% Pu})\text{O}_{1.93}$ and $(\text{U-25 wt \% Pu})\text{O}_{2.00}$ only amounts to 0.4 wt % of the total mass, the microprobe has been used to delineate values within this range. Four samples with O/M ratios ranging from 1.933 to 2.00 have been analyzed for oxygen. Precision varied from ± 0.01 O/M units for samples with O/M ratios of about 1.97-2.00 to ± 0.017 O/M units for the lower value specimens.

These data are corrected for dead time and background but not for:

1. Absorption of the $\text{O}_{K\alpha}$ (23.3 Å) line in the mixed-oxide matrix.
2. Atomic number differences ($Z_{\text{O}} = 8$, $Z_{\text{M}} = \sim 92.5$ where M is the average atomic number of uranium and plutonium).
3. Fluorescence of the $\text{O}_{K\alpha}$ (0.5 keV) by the higher energy uranium-plutonium lines.

The high specimen currents required to obtain this data perturb the fluorescing oxygen $K\alpha$ signal intensity which in turn seriously affects the count rate. This detrimental effect is magnified in samples with lower O/M values in which the electrical resistivity of the oxide is higher. A fusible alloy mounting compound is being investigated as a means of improving the analytical precision beyond that noted above.

6.4 FUEL-FISSION PRODUCT-CLADDING REACTIONS

6.4.1 Fission Product Cladding Reactions

The modes of combination of Cs, Te, and I, as well as the effects of the Cs/Te and Cs/I ratios on intergranular attack, are being investigated. These fission products were loaded into fueled type 316 stainless steel capsules that were heat treated for 100 hours in a $550^\circ\text{-}1100^\circ\text{C}$ thermal gradient. After heat treatment the capsules were gamma scanned to locate the fission products. The capsules containing iodine were then isothermally annealed at 200°C and rescanned to establish whether the iodine had migrated further. These results are being analyzed and will be reported in the next quarterly report.

6.4.2 Cladding Modification Testing

A total of 12 capsules, containing cesia and mixed-oxide fuel, that were exposed in a $550^\circ\text{-}1100^\circ\text{C}$ thermal gradient for 100 hours have been metallographically examined. Six additional capsules have been treated and are being prepared for examination. These capsules, along with the next scheduled tests, are described in Table 6-1. Intergranular attack to a depth as much as two mils was observed in most samples. Principal findings of these tests are:

1. Type 321 tubing is immune to Cs_2O attack while type 316 reacts at $\geq 575^\circ\text{-}600^\circ\text{C}$, and Type 347 reacts above 750°C . These reactions are shown in Figure 6-7.
2. The degree of cladding cold work in 304 and 316 tubing did not affect the reaction. As Table 6-1 shows, specimens ranged from 0-50% cold work.
3. Grain size within the range of ASTM 5 1/2 - 10 did not affect the degree of reaction.
4. The extra low carbon type 304 ELC (80 ppm C) behaved in a similar manner to type 304 with 600 ppm C.

These results suggest that stabilization in itself does not retard fuel-cladding reaction. It does, however, indicate that titanium has a beneficial role beyond that of gettering carbon. The temperatures at which attack was observed are listed in Table 6-2.

Table 6-1
DESCRIPTION OF CLADDING MODIFICATION CAPSULES^(a)

Capsule	Cladding		ASTM Grain Size	Other Variables	Status
	Alloy	Condition ^(b)			
I-1	316	Ann	6.5	Cs ₂ O at cold end 316 Discs ^(c)	Examined
I-2	316	Ann	6.5		
I-3	321	Ann	8		
I-4	347	Ann	8.5		
I-5	316	Ann	5.5		
I-6	347	Ann	10		
II-1	304	12% CW			
II-2	304	20% CW			
II-3	316	10% CW			
II-4	316	30% CW			
II-5	316	50% CW			
II-6	316	Ann	6 to 7.5	304 ELC (80 ppm C) Discs ^(c)	
III-1	304	12% CW + GT	6 to 7		Exposed; metallographic specimens in preparation
III-2	304	20% CW + GT	7		
III-3	316	Ann	6 to 7.5	Low Si (0.17%) IN 744 Discs ^(c)	
III-4	316	Ann	6 to 7.5		
III-5	I 800	Ann	8		
III-6	I 800	30% CW			
IV-1	321	Ann -- V	8	(316 + Ti) Discs ^(c)	Planned
IV-2	321	Stabilized	8		
IV-3	321	Ann	8	200 mg Cs ₂ O	
IV-4	316	Ann	6 to 7.5	High O/M Fuel, No Cs ₂ O	
IV-5	304	12% CW + GTV	6 to 7		Planned, heat treatment of cladding complete
IV-6	304	20% CW + GTV	7		

(a) Operated 100 hours in 1100 to 550°C thermal gradient; 50 mg Cs₂O added to hot end unless noted.

(b) Ann = annealed (1950°F, WQ), CW = cold worked, GT = Garafalo treated (900°F, 24 hours + 1300°F, 216 hours),
Ann-V = annealed at VNC, GTV = Garafalo treated at VNC.

(c) Discs of specific alloys interspersed between fuel pellets.

Table 6-2
METALLOGRAPHIC EXAMINATION OF CLADDING MODIFICATION ON CAPSULES

Alloy	Approximate Temperatures At Which Interaction Between Cladding and Contents Was Observed
304	> 900°C
304 ELC ^(a)	> 900°C
316	> 900°C
316 ^(b)	575–600°C ^(b)
347	> 750°C
321	No Attack Observed ^(c)

(a) This sample of Type 304 ELC contained 80 ppm carbon.

(b) This capsule had Cs₂O added at cold end, all other capsules had Cs₂O added at hot end.

(c) Maximum temperature examined for 321 was ~ 1000°C.

NEGATIVE SPECIMEN CURRENT IMAGE

IMA80E
1000X

NEGATIVE SPECIMEN CURRENT IMAGE

IMA80H
2500X

Cr X-RAY IMAGE

IMA80F
2500X

Fe X-RAY IMAGE

IMA80G
2500X

Figure 6-6. Specimen Current and X-Ray Images Showing Chromium Enrichment and Iron Depletion in Grain Boundaries. Sample is Unirradiated Type 304 ELC (0.008%C) at 690°C.

Figure 6-7. Cs_2O Attack of Austenitic Stainless Steels in Thermal Gradient Capsules

6.5 OXYGEN CONCENTRATION OF METALLIC FISSION PRODUCTS

Two synthetic metallic fission-product ingots were heat treated with stoichiometric mixed-oxide fuel at 1850°C for 5 to 20 hours. Analysis of samples from these ingots showed 260, 370, and 420 ppm oxygen. These measurements indicate that the oxygen content of fission-product nuggets is small and need not be considered in calculations of oxygen distribution and combination in irradiated fuel.

6.6 TRITIUM CONCENTRATION

Residual tritium distribution within irradiated fuel elements is of interest because of its 12.3-year half-life and possible release during fuel reprocessing. Tritium has been analyzed in the fission gas, cladding, and fuel of four fast reactor pins that attained burnups of 12 at. %. The results are reported in Table 6-3. These show that virtually no tritium is present in the fuel or cladding, that only a small amount (<40 ppb) is present in released fission gases and that the overwhelming proportion of the generated tritium passed into the sodium. Some, or most, of the tritium attributed to the fission gas may actually be Kr⁸⁵. Kr⁸⁵ is trapped with the tritium during the latter's conversion to water and subsequent analysis, and is inseparable in the scintillation counting. Of the small amount of tritium retained in the fuel elements ~95% is present within the fuel.

Table 6-3
TRITIUM CONCENTRATIONS FROM FUEL PINS IRRADIATED IN EBR-II

Fuel Element	Region	Sample Location (inches from fuel bottom)	Tritium Concentration (ppb)	Percent Retention (a)
E1H-29	Fuel	6.0	0.09	0.04
F3B-10	Fuel	6.9	1.19	0.75
E1H-29 ^(b)	Cladding	6.0	~0.0045	----
E1H-29	Cladding	8.5	0.005	----
F3B-10 ^(b)	Cladding	6.9	~0.06	----
F2V	Cladding	1.5	0.04	----
F2V	Cladding	7.2	0.01	----
F2V	Cladding	12.5	0.05	----
F2V	Cladding	Plenum	< 0.001	----
E1H-29	Gas	----	< 40	----
F3B-1	Gas	----	< 4	----

(a) Based on calculations of the amounts of tritium expected from the calculated burnups.

(b) Based on combined analysis of fuel and clad versus analysis of fuel alone where ~95% of the tritium found was located in the fuel.

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, and fuel thermal performance, 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). More recently, model analysis (under Task P) has been applied quantitatively to interpretation of the results, and to aid in the selection of ranges of parameters for further tests. The results of prior fast flux irradiation tests are interpreted as quantitatively as possible to guide the design of future tests. The selection of test parameters is also influenced and guided by the work under Task B on defected fuel testing; Task C, which focuses on the preferred range of parameters for safety considerations of LMFBR designs; and Task G, 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, diametral gap, axial gap, and external restraint. Fast flux tests are performed with both encapsulated and unencapsulated pins.

Selected irradiation tests in thermal test reactors are being performed in support of the fast flux irradiations in EBR-II. 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 2-3. 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 2-3 were calculated using the fission rates provided in the latest revision of the "Experimenters' Guide for Irradiations in EBR-II" as a basis.

Capsule irradiation tests performed in thermal spectrum test reactors which are being performed under Task F and capsule tests which were transferred from Task E to Task F, are listed in Table 2-4.

7.2 SCOPING AND STATISTICAL CONFIDENCE IRRADIATIONS

7.2.1 F3B Experiment (Fuel Void Deployment and Cladding Studies)

The destructive examination of six of the eighteen F3B encapsulated fuel pins is complete. These pins operated at ~ 15.5 kW/ft and $\sim 1150^\circ\text{F}$ cladding inside surface temperature calculated (peak) to $\sim 88,000$ MWd/Te peak calculated burnup. The results of chemical burnup determinations are shown in Table 7-1. The calculated values, using the reference method recommended by the EBR-II Project, are substantially higher than the values determined by chemical analysis for neodymium. A comparison of calculated peak burnup with measurements from fuel samples from the peak flux region of the pins shows the calculated burnup to be 11% to 19% higher than measured. This is consistent with past GE experience.

The discrepancy in burnup has a direct effect on the analysis of the fission-gas release from the fuel. The results, based on gas chromatography measurements of the gas collected from the pins, are shown in Table 7.2. For the measured burnup results, approximately 100% release is indicated.

7.2.2 F12P, Q Experiment (Effect of Cladding Thickness in a Thermal Flux Irradiation to High Burnup)

The F12P and F12Q fuel pins with 15-mil and 10-mil 316SS cladding respectively, have reached 100,000 MWd/Te in the GETR and have been removed for interim examination. The neutrographs indicate that both pins are intact.

*Contributors are listed in Appendix B

Table 7-1
COMPARISON OF CALCULATED AND
MEASURED BURNUP FOR THE F3B EXPERIMENT

Pin Number	Nd Determination			Calculated ^(b) Peak Burnup (atom %)	% Calculated Above Measured
	Sample Burnup (atom %)	Percent of Peak	Peak Burnup (atom %)		
E1H-21	8.05	98.9	8.14	9.43	15.8
E1H-23	8.36	98.9	8.45	9.86	16.7
E1H-26	8.56	98.9	8.66	9.61	11.0
E1H-29	7.61	98.9	7.69	9.18	19.4
E1H-30	7.75	98.9	7.84	9.28	18.4
E1H-33 ^(a)	N.A.	N.A.	N.A.	9.14	N.A.

(a) Pin Failed

(b) Using the reference method recommended by the EBR-II Project.

Table 7-2
PERCENT GAS RELEASE FOR F3B EXPERIMENT

Pin Number	Based on Calculated Burnup (%)	Based on Measured Burnup (%)
E1H-21	91	~ 100
E1H-23	85	96.7
E1H-29	87	~ 100
E1H-30	89	~ 100

7.3 ELEVATED CLADDING TEMPERATURE IRRADIATIONS

7.3.1 F5 Experiment (Fuel Pin Performance at High Cladding Temperature)

Post-irradiation examination of three of the nineteen encapsulated F5 fuel pins at ~55,000 MWd/Te is continuing. The calculated peak operating conditions for these pins were ~1200°F cladding inside surface temperature at ~16 kW/ft. The fuel pin design and operating parameters for F5F, F5M, and F5Q are listed in Table 7-3.

The fuel pins have been removed from their capsules, visually examined, photographed, and profiled for diameter changes. The F5F pin had a heavy white deposit on the outside cladding surface extending from the bottom of the pin to a region above the top of the fuel column. All three pins had a light deposit on the cladding surface in the cover gas region of the capsules. The deposits were easily removed by washing in a mild acetic acid solution, which resulted in a bright, shiny cladding surface. The maximum diameter increase on all three pins was 0.6% $\Delta D/D$, with no gross discontinuities.

The fuel pins were punctured for fission-gas collection and total gas and void volume determinations. The punctures were made 1.5 to 2.0 inches above the bottom of the fuel column. The gas from F5F and F5M was released in less than 1 minute. The gas was released from F5Q within 10 minutes. Gas chromatography analysis of the fission gas is in progress.

The fuel pins have been metallographically sectioned in the peak flux (fuel midplane) and maximum cladding temperature (fuel-upper blanket interface) regions. The F5F pin, with solution annealed Type 316SS cladding, showed no evidence of intergranular cladding attack in either region. There was an unidentified zone at the fuel-cladding interface, a denuded zone with a heavily precipitated matrix structure behind the denuded zone at the inside surface, and grain boundary precipitates across the cladding thickness, as shown in Figure 7-1.

Table 7-3
F5 EXPERIMENT DESIGN AND IRRADIATION PARAMETERS

Fuel Pin	F5F	F5M	F5Q
Cladding Type	316SS	321SS	316SS
Cladding Treatment ^(a)	Solution Annealed	Solution Annealed	50% Cold Reduced ^(b)
Fuel Form	Solid Pellet	Solid Pellet	Solid Pellet
Fuel Type	Co-ppt	Co-ppt	Co-ppt
Average Stoichiometry, O/M	1.97	1.97	1.97
Average Fuel Density, % T.D.	93.98	94.11	94.28
Smeared Density, %	90.34	89.86	90.70
Fuel Column Length, inches	13.529	13.468	13.514
Cold Gap Diameter, mils	3.2 to 5.0	3.4 to 6.7	3.6 to 6.9

(a) Nominally 0.250-inch o.d. x 0.015-inch wall seamless tubing.

(b) Based on reduction of cross-sectional area in the final pass.

*Figure 7-1. Typical Structure of Solution-Annealed Type 316SS (F5F Pin)
After 55,000 MWd/Te at ~1200°F (Maximum)*

Figure 7-2. Localized Intergranular Attack at the Fuel Upper Blanket Interface of Solution-Annealed Type 321SS (F5M Pin) After 55,000 MWd/Te at ~1200°F (Maximum)

Figure 7-3. Typical Structure of Solution-Annealed Type 321SS (F5M Pin) After 55,000 MWd/Te at ~1200°F (Maximum)

Figure 7-4. Unidentified Rim of Material in the Peak Flux Region of 50% Cold-Reduced Type 316SS (F5Q Pin) After 55,000 MWd/Te at $\sim 1200^{\circ}\text{F}$ (Maximum)

Figure 7-5. Typical Structure of 50% Cold-Reduced Type 316SS (F5Q Pin) After 55,000 MWd/Te at $\sim 1200^{\circ}\text{F}$ (Maximum)

The F5M pin, with solution annealed Type 321SS cladding, showed slight uniform intergranular attack over all the cladding inside surface, to a depth of ~ 0.2 mil in the peak cladding temperature region. There was no attack noted in the peak flux region. Localized intergranular attack to a depth of ~ 1 mil maximum was observed at the fuel upper blanket interface, as shown in Figure 7-2. The presence of the 1-mil attack was predicted by pulsed eddy current scanning prior to sectioning. There was a denuded zone around the intergranular attack, indicating no precipitation in the zone around the grain boundaries. The extent of cladding attack was limited to the fine grain area adjacent to the inside surface of the cladding. The zone of fine grain structure was present in the as-received tubing. There was an unidentified zone at the fuel-cladding interface, and grain boundary precipitate across the clad thickness (Figure 7-3).

The F5Q pin, with 50% cold reduced Type 316SS cladding, showed no evidence of intergranular cladding attack in either the peak flux or maximum cladding temperature regions. There was an unidentified zone at the fuel-cladding interface in the peak cladding temperature region, and a wide unidentified zone in the peak flux region in the form of a detached rim of material between the fuel and cladding (see Figure 7-4). The cladding had a heavily precipitated matrix structure, with precipitation along the grain boundaries and slip lines, as shown in Figure 7-5.

There was no evidence of sigma phase in any of the cladding structures, indicating that these pins did not operate for significant periods of time above 1200°F .

Metallographic sectioning at other locations, including the plenum region, is in progress.

7.4 GRID SPACER IRRADIATIONS

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

Fabrication of the fuel pins and the hex channels is complete. Testing of a grid-spaced test channel under static and dynamic load conditions simulating the loads expected during shipping, fuel pin insertion, and in-pile operation is under way.

Test results indicate that the strength of the beam welds of the grid spacer offers a wide margin over the <5 -pound maximum load expected during irradiation. As shown in Table 7-4, the mean joint breaking strength within each test in tests at room temperature, at 900°F , and after cycling 50 times between $80^{\circ} - 1200^{\circ}\text{F}$ was >30 -pounds in all cases.

7.5 IRRADIATIONS TO EVALUATE OPERATING LIMITS

7.5.1 F20 Experiment (Short-Term Irradiation to Determine Power to Melting Parameters)

Precipitation of the mixed-oxide powder for F20 is complete, and analysis and blending of the powder is underway. As suggested by RDT, annular pellet fuel will be used in two pins. Methods and materials for flux and temperature monitoring within the subassembly are being reviewed, with the intent of increasing the reliability of the values of the operating parameters, which will be critical to the evaluation of this experiment.

7.6 UNENCAPSULATED FUEL PIN IRRADIATIONS

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

Revised data on the presence of high worth driver subassemblies adjacent to the F3A subassembly have been received from the EBR-II Project. Ten runs are involved, rather than the two originally reported in the reactor run reports. The accumulated perturbed exposure was 3692 MWd, which is $\sim 30\%$ of the total 12,884 MWd irradiation. These data are being factored into the calculations for cladding temperature. A proper estimate of the cladding temperature is necessary to interpret the intergranular attack observed in this experiment.

The unperturbed calculated peak linear power of the F3A pins is shown in Table 7-5, along with the peak power including the perturbation. The unperturbed peak power occurred at the start of life in EBR-II run 30A. The perturbed peak power occurred at the start of run 30B.

As discussed in the last quarterly report, the measured burnup of F3A pins, determined by chemical analysis for neodymium, is substantially lower than the calculated values, using the reference method recommended by the EBR-II Project. Table 7-5 also shows the peak perturbed power corrected for this burnup discrepancy. Calculation of the peak cladding temperatures based upon these revised estimates of the pin power is in progress.

Table 7-4
F9E GRID SPACER BEAM WELD STRENGTH

Conditions	Joint-Breaking Strength (lb) ^(a)	
	Full Wave Beam	Alternate Wave Beam
1. Room Temperature	58	61
2. Room Temperature		49
3. Room Temperature After Flexing Grid		Center Area Joints 63 Edge Joints 37
4. 900°F	31	31
5. After Cycling 50 Times Between 80°F and 1200°F		Center Area Joints 31 Edge Joints 32

(a) Mean value within each test of a number of randomly selected spacer beam joints.

Table 7-5
F3A PEAK LINEAR POWER

Pin Number	Peak Linear Power (kW/ft)		
	Calculated ^(a) Unperturbed	Calculated ^(a) Perturbed	Perturbed Corrected For Burnup Discrepancy
F3B-1	15.9	16.3	14.2
F3B-2	16.3	17.9	15.6
F3B-3	17.0	19.4	16.9
F3B-4	16.3	17.3	15.1
F3B-5	15.0	15.9	13.9
F3B-6	14.9	15.3	13.4
F3B-7	16.5	16.9	14.8
F3B-8	16.6	17.6	15.4
F3B-9	16.6	18.3	16.0
F3B-10	16.8	19.6	17.2
F3B-11	15.6	18.0	15.7
F3B-12	16.1	17.9	15.7
F3B-13	16.4	18.4	16.1
F3B-14	16.8	20.1	17.6
F3B-15	16.5	18.7	16.3
F3B-16	15.0	16.3	14.3

(a) Calculated using the reference method recommended by the EBR-II Project.

7.6.2 F9C Experiment (37 Unencapsulated Pin Wire-Spaced Subassembly Committed to the EBR-II Mark E Run-to-Failure Program)

The interim examination of the F9C pins at ~ 7.5 at. % peak burnup, prior to reconstitution into Mark E subassemblies, was completed. All the pins are clad with annealed 316SS. Fuel parameters include pellets, both coprecipitated and physically blended, and powder which was physically blended. Smeared density is 84% to 88% in the pellet fuel, with fuel-cladding diametral gaps of 3 to 8 mils. The calculated peak linear power was ~ 13 kW/ft, with a corresponding maximum cladding temperature of $\sim 1020^\circ\text{F}$.

The hex can was easily removed from the subassembly X056A with a pull of only 100 pounds. Cleaning was very effective as evidenced by the fact that no residual sodium oxide was observed. Visual examination of individual fuel pins revealed that the pins were in excellent condition with surface appearances similar to a previous examination at 5 at. % burnup. Areas which were exposed to a silicone rubber contaminant (GE-RTV-30) at EBR-II during the previous examination were examined carefully. No indications of the location of this material or detrimental effects from it were observed.

Profilometer traces on five typical pins showed maximum diameter increases of 0.44% to 0.68%. The calculated peak fluence was $\sim 5 \times 10^{22}$ n/cm², $E > 0.1$ MeV. The diameter measurements were made using a continuous recording profilometer which followed a spiral trace to avoid interference with the spacer wire on each pin. Total pin length increases of 0.010 to 0.030 inch were measured.

Neutron radiographs indicated that no significant change had occurred in the fuel since the previous examination.

A report describing the damage to pin F9C-13 in the EBR-II hot cell and the probable sequence of events which led to bending the pin has been received from the EBR-II Project.

The data package for reconstituting the first F9C Mark E subassembly, which will operate at conditions similar to those of the original 37-pin experiment, was forwarded to the EBR-II Project.

7.6.3 F10C Experiment (19 Unencapsulated Pins Irradiated in Fast Flux High Cladding Temperatures)

Redesign of this experiment to focus on the redefined objective, which is to determine the extent of fuel-cladding chemical interactions, is underway. Scoping calculations are being performed to investigate the fabrication parameters that would be required for a plutonia-rich fuel pin. Irradiation of a plutonia-rich pin would provide data on cladding attack in an environment (oxygen potential) more prototypical of the LMFBR than can be obtained with U-235 enriched pins.

7.6.4 F11A Experiment (37 Unencapsulated Pins Irradiated in Fast Flux High Cladding Temperatures)

The data package for F11A experiment has been forwarded to the EBR-II Project for review.

7.7 IRRADIATION EVALUATION SUPPORT

7.7.1 Fission-Gas Analysis

A compilation of fission-gas release data from all Task F fast flux irradiations to date is summarized in Figure 7-6. The trend is towards increased release with increasing burnup, with near 100% release observed at as low as ~ 4 at. %. The percent gas release values were based upon the measured burnup, assuming a gas generation rate of 0.246 gas atoms/fission. Five pins show a calculated release $> 100\%$, to a maximum of 108%. A review of the calculation parameters indicates that the burnup and the gas generation rate contain the largest uncertainties, which are estimated to be $\pm 3\%$ and $\pm 8\%$, respectively. The total uncertainty of the gas release calculation is approximately 9% (standard deviation). The calculated values showing $> 100\%$ release are therefore within experimental error.

7.7.2 Cladding Stress in Fuel Pins F2Vm F2X, F2H

Calculations were performed to bracket the cladding stress levels in the 316SS clad F2 pins, as part of the investigation of the increase in pin diameter that was observed at high burnup ($> 100,000$ MWd/Te).

A continuous loading mechanism is required to sustain significant stress levels for long periods of time, since stresses relax rapidly at the experiment temperatures. A steady-state creep model was chosen to estimate the maximum time-average stress levels. The $\Delta D/D$ values determined from the profilometry measurements were reduced by the calculated interference strain at startup and by the immersion-density-determined cladding swelling values. The remaining strain, called the "excess strain," was assumed to be caused by the stress-dependent mechanisms of irradiation⁶ and thermal⁷ creep. The average creep rate was assumed to operate over the entire cladding life. It was further assumed for simplicity of calculation that a uniaxial state of stress existed in the hoop direction. If the actual

Figure 7-6. Fission-Gas Release versus Measured Burnup (Based on 0.246 Gas Atoms/Fission)

state of stress is biaxial, the result of the uniaxial assumption will be an underprediction of the true stress level. However, because of the strain insensitivity of the calculation, relatively small corrections in stress would result. The calculated stress and component strains are plotted as a function of length along the pin for F2V in Figure 7-7. If sustained-loading and consequent creep were the mechanism which produced the excess strain, then the stress would have had to have been >30,000 psi.

If stress enhancement of cladding swelling occurred in these pins, the immersion density values should be higher than a prediction based upon the stress-free values, or should be on the high side of the values calculated using the metal swelling correlations. As shown in Figure 7-7, F2V sustained significantly more swelling than calculated. Pins F2X and F2H, however, did not. The cladding swelling was calculated using a representative correlation.³ The swelling discrepancy (immersion density minus calculated) is plotted as a function of the stress level determined above in Figure 7-8 for all three pins. The stress dependency is weak, indicating that the cladding swelling was not enhanced by the high stress that would have been necessary to produce enough steady-state creep to explain the excess strain observed. The uncertainties are large, however, and additional data are needed before a definite answer can be reached.

7.7.3 Pulsed Eddy Current Cladding Analyzer

Nondestructive examination using pulsed eddy current (PEC) techniques to identify defects in tubing has been applied to the evaluation of irradiated fuel pins. Development and application of the PEC technique has been carried out at Argonne National Laboratory. The technique can be applied to the practical problem of nondestructively inspecting irradiated fuel rods. The basic principle involved in this technique is the monitoring of a mixed spectrum of frequencies transmitted from a field coil inducing localized eddy currents on the tubular specimen. The field generated by the eddy currents is in turn sensed by a pickup coil. Any discontinuities in the tubing will cause a distortion in the eddy current pulses, and a corresponding signal is produced from the pickup coil and recorded.

A PEC cladding analyzer has been built, checked out on the bench and in-cell, and used successfully to locate likely regions of discontinuities at the cladding inside surface of three F5 fuel pins (F5F, F5M, and F5Q) prior to destructive examination. The pulse monitored by the pickup coil is approximately 5 microseconds in duration. Sample points may be selected at any point in time on the waveform for analysis. Two sample points were used simultaneously to monitor the outside and inside surfaces thereby providing maximum sensitivity for locating discontinuities. The two

Figure 7-7. Stress and Strain Components for Pin F2V

Figure 7-8. Stress versus Swelling Strain Discrepancy

outputs were recorded on a two-pen, fast-response recorder. Although the same waveform is used for both the outside and inside surfaces different portions of the waveform are monitored for each surface. A discontinuity on one surface will appear on the outputs of both surfaces, but the largest signal will appear on the surface where the discontinuity exists.

The transducer (containing the field and pickup coils) and a preamplifier were mounted on a traversing system. The fuel rods were held in place and the transducer moved axially along the fuel rod. An indexing device was incorporated to allow rotation of the tube in 60-degree increments (the tube does not rotate during inspection). Consequently, a minimum of six inspection passes was made on each tube. Recordings were made of each rod during the forward and return traverses of the carriage at each 60-degree increment.

The transducer drive system pulsed an event marker on the recorder providing 0.2-inch indications on the chart recordings. The markers, along with pre- and post-irradiation neutrographs, were used to locate the indications of defects. The fuel rod spring and the spacers on either side of the spring gave distinct signals on the eddy current trace, which were also used as reference points.

The data indicated that discontinuities at the cladding inside surface (intergranular attack, metallic reaction zones, etc.) exist in each of the F5 fuel pins tested. The indexing of the fuel pins during inspection allows discontinuities to be located at a specific angular orientation. Verification of the indications is being obtained by metallographic examination.

A typical trace is shown in Figure 7-9, for fuel pin F5M, clad with solution annealed Type 321 stainless steel. The peak observed in the figure on the inside surface near the fuel upper blanket interface has been verified by metallographic examination to be a zone of local intergranular attack. A portion of this zone is shown in Figure 7-2.

7.7.4 Neutron Radiography Standards

Fabrication of four mixed-oxide neutron radiography standards is complete. The standards GE-FE1 and GE-FE2 have been shipped to EBR-II to be included in the neutron radiographs of Task F fuel pins. The other two standards will be used at Vallecitos Nuclear Center.

TYPE 321 SS, ANNEALED STANDARD

TYPE 321 SS, ANNEALED F5M

Figure 7-9. Pulsed Eddy Current Cladding Analyzer Trace of Fuel Pin F5M

8. TASK G-6—CORE DESIGN ANALYSIS

Task Leader: P. R. Pluta*

8.1 GENERAL

The main purpose of this task is to develop a grid-spaced fuel assembly for the fast breeder reactor. 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 bundle 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 basis for the selection of the grid-spacer support system as the preferred system for the LMFBR.

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.

8.2 FFTF GRID-SPACED DRIVER FUEL ASSEMBLY

The purpose of this subtask is to assess the performance of various 217-pin 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.

Detailed design drawings of the current reference conceptual FFTF grid-spaced driver fuel assembly have been completed.

A structural analysis of the effect of the notches on the FFTF channel wall is continuing.

It is planned to omit grid spacers from the pad region above the FFTF core in order to preclude the possibility of disturbing the channel or pad material in the region of the clamping loads. The resulting 10-inch grid-to-grid axial pitch is not expected to result in a design problem because the pad is located outside of the core region of the fuel assembly. The vibration characteristics of this design will be evaluated in a hydraulic flow test.

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 inch o.d. fuel pins at a P/D = 1.28, high swelling stainless steel).

During this period, mechanical tests and evaluation in support of the design were conducted as the fabrication of F9E continued. Dimensional modifications in the F9E design were made as a result of this work.

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 conceptual design of the F9F assembly, based on a fuel pin outside diameter of 0.250 inch, was completed. The following design parameters were incorporated in the F9F design:

- Pin-to-channel edge spacing to achieve prototypical rod bowing and grid-to-channel differential swelling
- Pin-to-grid contact loading at beginning of life
- Various grid beam thicknesses
- Modified end tab design
- Grid-spacer axial pitch above the core
- Backup grid-spacer design

Design changes resulting from a reduction of the pin outside diameter to 0.230 inch are being evaluated.

*Contributors are listed in Appendix B

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 the interaction with fuel pins and the channel wall.

8.5.1 Grid Insertion and Inspection Tests

Wave beam grids are positioned axially within the channel by locking tabs on the end of beams which are engaged into notches in the channel wall. Although the beam tabs are positioned in the channel notches by the fuel pins when they are inserted, the grid tabs must be properly seated during assembly and remain in position before the pins are inserted. The purpose of this test was to verify that wave beam grid tabs are properly seated in the channel notches during assembly.

The first test was concerned with tab seating of 37-pin grids similar to those being used in the EBR-II grid-spacer test assembly F9E. Grids were inserted into a prenotched section of an EBR-II channel using the grid insertion tool. Inspection of the inserted grids was done using a boroscope. As a result of this test the grid insertion and inspection procedure for F9E was modified.

8.5.2 Grid-Spacer Push Tests

In order to determine the uniformly applied force required to axially dislodge a grid spacer within a fuel channel, grid-spacer push tests were conducted on a 37-pin EBR-II grid assembly mockup.

The 37 pins inserted in the grids provided prototypical grid spacer alignment and tab engagement. Results of the test indicate that approximately 75 pounds are required to axially dislodge an EBR-II size grid spacer of the design being used in the F9E.

8.5.3 Shipping Load and Vibration Tests

The test assembly F9E, which will be irradiated in EBR-II, will be shipped to the reactor with grid spacers inserted into the channel. Fuel pins will be inserted into the assembly after it has been received at EBR-II. In order to ensure that the grid spacers will not be dislodged or damaged during shipment, a shipping case drop test and vibration test were conducted.

The shipping case drop test was performed with a modeled crate and the actual shipping instrument package. Its two purposes were to verify that the grid spacers would not disengage when the crates were subjected to high shock loads, and to obtain a calibration for the accelerometer readings whereby a set of readings on the accelerometers could be interpreted in terms of a known event. These events were drop tests on a rigid surface from heights ranging from 3 to 10 feet and at varying orientations. No grid dislodgement or damage was observed.

A 21-inch section of a fuel assembly containing typical F9E grid spacers was mounted to a shaker table and subjected to a range of vibration loads and frequencies. Examination of the assembly had not been completed at the end of this quarter.

8.6 SODIUM FLOW TEST

The vibration and wear characteristics of various grid-spacer designs will be investigated in a 37-pin sodium flow test. The design of the test was completed and is now being fabricated. This test will be conducted in 1000° F sodium at the maximum velocities expected in FFTF driver fuel assemblies. The FFTF shield-orifice region has been simulated in order to achieve prototypical coolant turbulence.

8.7 HYDRAULIC FLOW TEST

The vibration characteristics of the FFTF grid-spaced driver fuel design will be investigated in a 217-pin hydraulic flow test which is now being designed. This hydraulic test will be operated at flow rates up to 500 gpm. The instrumentation which will be used for this experiment will consist of accelerometers which are mounted inside dummy fuel pins as well as proximity sensors fixed to the channel wall.

8.8 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, the final GEAP report summarizing this work received final internal review.

9. TASK G10—SAFETY METHODS AND ANALYSES

Task Leader: D. L. Brown*

9.1 GENERAL

The purpose of Task G-10 is to develop and use analytical 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 (TOFF⁹) has been extended from a single layer model to a multilayer model to treat the full fuel pin. The improved model has been coupled to the THTE transient heat transfer program for checkout purposes. The specifications have been prepared for coupling the TOFF model to the FREADM¹⁰ program.

The improved model allows for:

1. Axial fuel movement in the central molten region
2. A pressure gradient in the solid fuel
3. Computation of the axial failure location
4. User control of the fission gas distribution within the fuel
5. User-specified cladding thickness as a function of axial location

The distribution of the fission gas retained in the fuel at the transient initiation has a major influence on the fuel failure threshold. The flexibility of the model will allow the sensitivity of the failure threshold to be examined by specifying the fission-gas retention in the columnar grain region, equiaxed grain region and the as-fabricated region.

To accommodate the cladding thickness variation with fluence or burnup, a user-specified thickness has been programmed so that a different value may be specified for each axial node.

Upward fuel movement will take place if the fission gas-molten fuel pressure overcomes the weight of solid fuel above it, the downward force of the fission-gas pressure in the plenum, and the frictional force between the cladding and solid fuel. Calculating this last quantity requires a knowledge of the friction coefficient. Currently a value of 0.5 is used as derived by Ross and Stoute¹¹ in their contact heat transfer coefficient studies. This value has some experimental support. The solid fuel area over which the molten fuel fission gas acts to lift the solid fuel above it is taken to be the minimum central void area, or if there is no central void, the area generated by the molten fuel radius.

9.2.2 Loop Hydraulics and Channel Voiding

9.2.2.1 Loop Hydraulics

The purpose of this subtask is to develop computer models to describe the transient flow throughout the primary coolant system under hypothetical LMFBR accident conditions. The VOIDUM routines are a set of subroutines to perform these computations within the FREADM program.

The routines were originally written for a tank-type coolant system. They have been revised¹ to allow the dynamic response of a piped coolant system to be simulated. Two types of piped systems are included, with the pumps located in either the hot or cold leg of the system.

The modified routines are being incorporated into the FREADM program. A series of demonstration runs will be made to serve as final checkout of the modified model.

9.2.2.2 Coolant Boiling and Channel Voiding

The purpose of this subtask is to develop computer models to describe the boiling process and void distribution within an LMFBR core during hypothetical accident conditions.

*Contributors are listed in Appendix B

The BUBBLE routines have been revised and tested and are currently being incorporated into the FREADM program. These routines model the effect of axial condensation on the voiding process.

A preliminary study of radial condensation effects in LMFBR core voiding is in progress. The basic physical mechanisms for radial condensation have been summarized, and the conditions under which radial condensation might be important have been evaluated.

A void bubble is initiated when the local surface temperature exceeds the required voiding superheat. The bubble, being initially of a spherical shape, expands into the surrounding superheated liquid laterally as well as axially. It passes over a liquid film which remains on the fuel pin surface and whose evaporation provides most of the vapor within the bubble and causes its continued growth.

During the further expansion of the bubble, its interface can leave the region of superheated liquid and enter a low temperature region which will result in partial condensation of the vapor. Therefore, if the temperature difference between the hottest fuel pin surface and the hex channel is greater than the boiling superheat, radial condensation will take place.

Boiling in an LMFBR fuel assembly will probably be due to either local blockages within the pin bundle (large inlet blockages are extremely improbable due to constructional precautions) or loss of flow. The order of magnitude of radial temperature gradients across the fuel bundle in each case can be estimated:

- a. Local blockages - Since the bulk fluid temperature remains at $\sim 1050^{\circ}\text{F}$ and sodium boiling occurs at $1800^{\circ}\text{--}1850^{\circ}\text{F}$, the temperature differences can be as high as 800°F .
- b. Loss-of-Flow Transients - For very fast flow loss transients, McWethy^{1,2} has estimated a temperature difference of $\sim 600^{\circ}\text{F}$ between the bundle center and the hex wall. If the transient is very slow, the temperature difference approaches the steady-state value of 30° to 70°F . Thus, radial temperature differences between 40°F and 600°F can be encountered in loss-of-flow accidents depending on the time-scale of the transient.

Sodium superheat values typically range between 30°F and 200°F . Thus, it can be concluded that radial condensation will be extremely important in determining voiding patterns for the case of a local blockage and of significance for many loss-of-flow transients.

9.2.3 Probabilistic Modeling

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

Studies of reactor safety with a fuel failure probability tree continued. Beginning with a random cladding failure, the probability tree shows the possible consequences of failure propagation including deterioration to a whole core incident. Four cases were considered, differing with respect to the behavior of fission gas. The cases included vented and sealed plenum fuel designs, each with either a gas tight or open central void. Cladding stress due to fission gas trapped in the central void is a potential contributor to early cladding failure and increased failure propagation probability. A 1-month continuous power run was assumed, which would result in about a 20,000 psi cladding stress if the central fuel region were gas-tight.

The presence of high central void pressure is assumed to increase the frequency of cladding failures (at equal burnup), and to increase the susceptibility to failure propagation associated with mechanisms such as fission-gas blanketing, excessive spacer contact forces, ejection of fuel debris on failure, and voiding due to fission-gas release.

Table 9-1 shows the results of the probabilistic analysis for the four cases, assuming no control system action. The results suggest that vented fuel can potentially improve core economic and safety performance, particularly if the central void remains open. The possibility should be further explored that fission gas trapped in the central void will exert a significant pressure on the cladding.

In this trial calculation the frequency of voiding due to fission gas release appears to strongly influence the frequency of experiencing a significant power excursion.

9.2.4 Core and Reactivity Modeling

The purpose of this subtask is to improve the modeling methods for hypothetical accidents involving the whole core of an LMFBR.

Test cases are being run using the VENUS^{1,3} Lagrangian hydrodynamics program for checkout purposes and for comparison with the modified Bethe-Tait calculations. Comparative computations are being performed using two different equations of state for fuel.

The VENUS program is being coupled to the FREADM program.

Table 9-1
EFFECT OF FISSION GAS-ACCOMMODATION ON INHERENT REACTOR SAFETY

Case	1	2	3	4
Description of fission-gas accommodation	Vented, open void	Vented, gas tight void	Plenum, open void	Plenum, gas tight void
Assumed initial fuel failure frequencies (per year)	100	400	300	500
Mean-annual-cost ^(a) (10 ³ \$)	240	2040	2070	5040
Frequency of a significant power excursion (per year) initiated by fuel failure propagation assuming no control system action is taken	0.00035	0.00240	0.01800	0.04200

(a) Based on the cost data in GEAP-13639, "Core Design Development Needs In Relation To Fuel Failure Propagation, Sodium Boiling and Clad/Fuel-Sodium Thermal Interaction."
 Compiled by L. M. McWethy, October 1970.

9.2.5 FREADM Update

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

The following models are in the process of being incorporated or are ready for incorporation:

1. VOIDUM extended loop dynamics model
2. TOFF improved fuel-failure model
3. BUBBLE improved channel boiling model
4. VENUS Lagrangian hydrodynamics program

9.2.6 Parametric Analysis

The purpose of this task is to perform exploratory analyses in areas of potential safety significance and to provide analytical backup for model development and experiment design analysis.

9.2.6.1 Exploratory Hydrodynamics Analysis

A users manual for the PISCES IDL computer code has been obtained from Physics International, San Leandro, California. This code is one of a family of codes developed by Physics International to obtain solutions to the general hydrodynamics problem including stress wave generation and propagation. The code is being applied to problems associated with core disassembly analysis. Initial applications are primarily for investigating code operation and the cost of problem solving. The major input information required for the code is the equation of state for the material used. An equation of state subroutine for metals,⁷ which is compatible with PISCES (it gives pressure as a function of specific volume and energy), has been obtained and is being checked out for sodium.

The equation of state code must be input as a subroutine to the PISCES-IDL and there are 20 variables, such as molecular weight, critical temperature, heat of vaporization, etc., which are used to identify the particular metal. The vapor-liquid phase is based on the principle of corresponding states and employs a number of empirical formulations. The equation of state subroutine has been programmed so that it can be run separately from the PISCES code. Adjustments are being made to the input variables so that the code output for sodium will agree reasonably well with the Golden and Tokar⁸ data. To date this comparison has not been favorable, but a number of inconsistencies in input have been identified and the effort is continuing.

9.2.6.2 Transient Bundle Thermal-Hydraulics

The purpose of this exploratory analysis is to examine the potential for radial condensation resulting from thermal gradients near the channel wall during loss-of-flow accidents. Results of this work will provide further information on the importance of radial condensation in sodium boiling resulting from loss of flow.

A model has been constructed for the THTE program representing a region near the channel wall in a 127-pin bundle. The region considered in the analysis is shown on Figure 9-1. Initially the first 3 1/2 rows of pins are being considered, and the effects of circumferential conduction in the fuel and cladding are included as well as interchannel mixing, by the eddy diffusivity model of Rogers and Todreas¹⁶ (with enhanced coefficients to represent wire-wrap effects).

The model consists of 10 axial planes of nodes composed of ~100 nodes per plane. The flow distribution has been determined assuming no radial pressure gradients exist at any axial plane. The data for a single axial plane have been run and checked for accuracy. The model representing 10 axial planes has been constructed and is being checked prior to initiating calculations.

127 PIN BUNDLE

Figure 9-1. Typical Radial Section Near the Channel Wall

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 late in 1970.

10.2 LIAISON

The review of current programs and definition of needs, as described previously¹ have been initiated and P.M. Tschamper from BRD has begun a 1-year assignment at ANL, with office at Reactor and Safety Analysis Division, to promote liaison and coordination between related GE and ANL programs.

*Contributors are listed in Appendix B.

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, and especially on the survival of the cladding at temperatures and specific fuel rod 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 which influence and control it in LMFBR primary alloys and those which 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 is shown in Table 2-5.

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

The three L-6 capsules (Mark-B7 type) have been discharged and were shipped from EBR-II to the Vallecitos Nuclear Center. The L-6 capsules 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 2.3×10^{22} n/cm²-GE and 7.8×10^{22} n/cm²-GE, respectively.

11.2.2 Series L-16 Materials Capsules

A 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 including types 304, 316, and 321 as a function of temperature during irradiation, metallurgical condition, and fluence.

11.2.2.1 L-16A Capsules

The 19 L-16A capsules are designed to accommodate various temperature monitoring devices and flux wires to measure the temperature during irradiation and neutron flux in a typical L-16 type Mark-A19 subassembly. Assembly of the capsule inserts is complete and encapsulation of these inserts is scheduled for the next quarter at EBR-II FEF.

11.2.2.2 L-16B Capsules

The 19 L-16B capsules are in the EBR-II core row 4 (S/A X098). The irradiation exposure for this S/A is scheduled for a total fluence of 3×10^{22} n/cm²-GE. The total fluence accumulated to date is 0.9×10^{22} n/cm²-GE.

11.2.2.3 L-16D Capsules

These three capsules were designed to accommodate highly irradiated ring specimens and unirradiated controls 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 complete and fabrication of the capsule insert parts has been initiated. Approval-in-principle has been received for irradiation of these capsules in EBR-II.

11.3 CLADDING ALLOY SWELLING

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

* Contributors are listed in Appendix B.

11.3.1 Cladding Swelling Studies

Transmission electron microscopy (TEM) examination of Types 316 and 347 stainless steel cladding from the F2 series of EBR-II fuel pin irradiations continued. Initial results were previously reported.¹

Cladding samples examined were irradiated to fluences from 3.1 to 7.3×10^{22} n/cm², $E > 0.1$ MeV-GE at calculated temperatures over the range of 470° – 545° C (880° – 1013° F). Mean void size, void number density, calculated swelling, and immersion density swelling data are summarized in Table 11-1.

TEM examination of the high fluence Type 347 specimens from pin F2G confirmed the conclusion (initially based on the Type 316 data) that at these temperatures, 525° C to 540° C (975° – 1000° F), the increased swelling occurring between 4 and 7×10^{22} n/cm², $E > 0.1$ MeV-GE is due primarily to the growth of existing voids rather than further void nucleation. Comparison of data from samples F2T-8 and F2G-H irradiated to 3.9 and 7.0×10^{22} n/cm², $E > 0.1$ MeV-GE, respectively, at comparable temperatures shows that the void number density has remained essentially the same ($\sim 1.5 \times 10^{15}$ voids/cm³) while the mean void diameter increased from 250 to 470 Å with an associated increase in the swelling (measured by immersion density) from 2.25% to 10.02% $\Delta V/V$.

The detailed investigation of the void morphology in the F2 series cladding may provide insight regarding the possible effects of an externally applied stress on void formation and growth. The F2 fuel pin profilometry and cladding density measurements show that the three high burnup pins had undergone varying amounts of plastic deformation.^{1,7} The maximum plastic deformation values were 1.5% , 0.8% , and 0.1% $\Delta D/D$ for fuel pins F2G, F2V, and F2X, respectively. United Kingdom data from solution annealed Type 316 fuel pin cladding¹⁸ show greater swelling at

Table 11-1
SUMMARY OF IRRADIATION-INDUCED VOID FORMATION
DATA FROM F2 CLADDING

Specimen Designation	Cladding Type	Fluence (n/cm ² $\times 10^{22}$ $E > 0.1$ MeV-GE)	Temperature During Irradiation		Swelling ^(a) $\Delta V/V\%$	TEM Results		
			$^\circ$ C	$^\circ$ F		Mean Void Diameter Å	Void Number Density (cm ⁻³ $\times 10^{14}$)	Calculated ^(b) Swelling $\Delta V/V\%$
F2H-4	316 ↓	3.85	512	954	1.41			
F2H-7B		3.93	536	997	1.47			
F2H-7A		3.77	542	1007	1.38			
F2H-8C		3.10	546	1015	0.68	365	2.1	0.64
F2V-C	316 ↓	6.15	471	881	3.95	285	13.3	2.0
F2V-E		7.03	495	923	7.48			
F2V-F		7.32	508	946	7.76	454	12.6	7.05
F2V-J		7.32	523	975	8.68	640	4.8	7.85
F2V-M		6.40	534	994	5.98			
F2V-P		5.28	537	998	2.98	535	3.2	3.07
F2X-D	316 ↓	5.58	460	861	2.86			
F2X-J		6.56	505	941	5.86	464	8.9	5.5
F2X-M		5.60	516	960	3.37			
F2T-8	347 ↓	3.92	538	1000	2.25	251	15.3	1.89
F2T-9		3.76	543	1009	1.99			
F2T-11		3.27	546	1014	1.35			
F2G-D	347 ↓	6.40	487	909	5.80	270	30	4.3
F2G-H		7.00	526	980	10.02	470	13	9.3
F2G-M		5.50	541	1007	5.49			

(a) Measured by immersion density.

(b) Calculated from void size and number density data.

lower fluences but similar temperatures than was measured for the F2 pins. It has been frequently conjectured that the extensive swelling of these DFR fuel pins was enhanced by the biaxial stress system existing in the cladding from fuel-cladding mechanical interactions and internal pressure due to released fission gases.

A stress system could affect void formation and growth at high cladding temperatures in at least two ways: (a) by the formation of grain boundary creep cavities in the manner described by Hull and Rimmer¹⁹ which would be a minor effect, or (b) by causing an increase in void growth kinetics as described by Gulden and Kaae²⁰ which could have a major influence on the extent of swelling. Examination of the grain boundaries in the F2G, F2V, and F2X specimens showed no evidence of grain boundary creep cavities. Conversely, zones denuded of voids were apparent at the grain boundaries. Figure 11-1 shows a typical grain boundary area of a specimen (F2G-H) which had experienced 1.5% $\Delta D/D$ plastic deformation during irradiation.

The question of a stress effect on void size is more difficult to resolve experimentally. The void size is acutely dependent on irradiation temperature as shown for type 304.^{21,22} Thus any comparison of void size must be made at the same temperature. Knowledge of sample temperature during irradiation in EBR-II is subject to uncertainties, especially between differing specimen geometries. Void size is also dependent on neutron fluence so that size comparisons must be made at similar fluence levels. Unfortunately, no void size data are available from unstressed solution annealed Type 316 for fluences as high as those of pins F2V and F2X. Recently reported data on swelling of unstressed Type 316²³ are at fluences comparable to that obtained by F2H. These specimens (HEDL) irradiated at 1000°–1100°F (538°–593°C) to 2.63 and 4.47 $\times 10^{22}$ n/cm², $E > 0.1$ MeV showed average void diameters of 274 and 385 Å, respectively, with a void number density of 5 $\times 10^{13}$ to 10¹⁴ voids/cm³. Specimen F2H-8C (see Table 11-1) irradiated in the same temperature range to an intermediate fluence level showed an average void diameter of 365 Å and a number density of $\sim 2 \times 10^{14}$ voids/cm³. Considering the temperature and fluence uncertainties, the void sizes in the stressed and unstressed specimens are equivalent. Void number density is difficult to determine, a factor of 2 can be within the experimental error. However, the swelling in the HEDL specimens was low ($\sim 0.1\%$ $\Delta V/V$) compared to 0.68% $\Delta V/V$ for F2H-8C; thus if the void sizes are similar, then the higher void concentration in F2H-8C is real.

A comparison of type 316 specimens F2X-J and F2V-F (irradiated at 500°–510°C (932°–950°F)) is shown in Figure 11-2. Specimen F2V-F underwent 0.7% $\Delta D/D$ plastic deformation compared to 0.1% $\Delta D/D$ for F2X-J. It can be assumed that F2V-F was subjected to higher sustained stresses while being irradiated at the same temperature as F2X-J. The measured mean void diameters are almost exactly equal but the void number density is higher in F2V-F. As in the previous comparison the measured swelling also was greater in the specimen with the higher void number density. Since the void sizes are equivalent, the difference in void concentration is considered real and leads to the conclusion that the stressed specimens of solution annealed type 316 stainless steel irradiated at $\sim 500^\circ\text{C}$ (932°F) contain more voids than the unstressed specimens. Inspection of the intragranular precipitation in F2V-F and F2X-J indicates that the extent of matrix precipitation is also greater in the stressed specimens as can be seen in Figure 11-2. These data suggest that the effect of an external stress system on void formation in this temperature range is not to generate creep cavities or enlarge existing voids but rather to increase the degree of intragranular precipitation which in turn increases the void concentration and the overall swelling. This explanation appears to be consistent with the F2 series data described here, the HEDL unstressed data²³ and the DFR data¹⁸ in which the higher swelling of Type 316 compared to Type 316L was attributed to the more extensive matrix precipitate formation in Type 316. Measurements of the matrix precipitate concentrations of the F2 cladding will be made to further investigate this concept.

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 precut tensile specimens and tubular specimens from Task L and Task F cladding and capsule tubing. Various uniaxial and biaxial techniques are being used to perform the mechanical properties evaluation.

11.4.1 Tensile Tests on Stainless Steels (L-2)

Uniaxial tensile tests are complete on precut stainless steel strip specimens taken from the L2 materials capsule after irradiation in EBR-II S/A X050 to a peak fluence of 6.8 $\times 10^{22}$, $E > 0.1$ MeV-GE. Samples of four types of stainless steel; 316L, 321, 304, and 347, have been tested in air at three temperatures: 482°C (900°F), 593°C (1100°F), and 704°C (1300°F). At the higher temperatures, 593°C (1100°F) and 704°C (1300°F), the tests were performed at three strain rates: 2, 2×10^{-2} , and 2×10^{-4} min⁻¹. Control tests on unirradiated samples were carried out concurrently on material from the same heats. These materials were the same as irradiated previously in S/A XG06.²⁴

All tests were performed on an Instron TTC tensile machine with the temperature of the specimen monitored by two thermocouples positioned within 1/4 inch of the gauge length. Total elongation values and uniform elongation

TYPE 347 FROM PIN F2G. SPECIMEN IRRADIATED AT 525°C (980°F) TO $7.0 \times 10^{22} \text{N/CM}^2$, $E > 0.1 \text{ MeV-GE}$. MEAN VOID DIAMETER = 470 Å, VOID NUMBER DENSITY = $1.3 \times 10^{15} / \text{CM}^3$. SWELLING = 10%.

Figure 11-1. Transmission Electron Microscopy of Irradiated Type 347 Stainless Steel Showing Void Denuded Grain Boundary

(a) SECTION F2V-F IRRADIATED AT 510°C (950°F) TO $7.3 \times 10^{22} \text{ N/CM}^2$, $E > 0.1 \text{ MeV-GE}$.
 $1.3 \times 10^{15} \text{ VOIDS/CM}^3$, MEAN DIAMETER - 454 Å. SWELLING = 7.8%

(b) SECTION F2X-J IRRADIATED AT 510°C (950°F) TO $6.6 \times 10^{22} \text{ N/CM}^2$, $E > 0.1 \text{ MeV-GE}$.
 $9 \times 10^{14} \text{ VOIDS/CM}^3$, MEAN DIAMETER - 464 Å. SWELLING = 5.9%

Figure 11-2. Transmission Electron Microscopy of Irradiated Type 316 Annealed Stainless Steel Showing Increased Intragranular Precipitation

values were determined from load vs. crosshead deflection curves. The accuracy of the strain measurements taken from the load vs. deflection curves was confirmed by pre- and post-test measurements of 15 specimens selected at random. Metallography has not yet been performed on any of the specimens.

Figure 11-3 illustrates the mechanical property changes resulting from irradiation to high fluence levels as a function of test temperature (data all at 0.02 min^{-1} strain rate). These data indicate increases in yield stress and ultimate tensile strength with corresponding large reductions in elongation. The effect of strain rate, over the range 2 to $2 \times 10^{-4} \text{ min}^{-1}$, was determined at temperatures of 593°C (1100°F) and 704°C (1300°F). Over this strain rate range, the property values (0.2% yield stress, Y.S.; ultimate tensile stress, UTS; and total elongation, %TE) of the irradiated material increase with increasing strain rate, as shown in Figures 11-4, 11-5, and 11-6, and in Table 11-2. The effect of strain rate at 900°F was not determined (all tests at this temperature were at $2 \times 10^{-2} \text{ min}^{-1}$). Data for the unirradiated material tested at 704°C (1300°F) indicate lower total elongation values for the unirradiated material when tested at the fast (2 min^{-1}) strain rate at either 593°C (1100°F) or 704°C (1300°F) (Figure 11-6). This difference in the response of the irradiated and unirradiated material to a strain rate increase is attributed to the absence of necking in the irradiated specimens; the measured total elongation is almost entirely uniform. The unirradiated specimens undergo significant necking prior to failure. Over the strain rate range examined, the total elongation of the unirradiated samples remains relatively constant. However, the percentage of the total elongation attributable to necking increases with decreasing strain rate while the opposite is true for the uniform elongation. If the contribution of necking to the sample elongation is taken into consideration and the uniform elongations for irradiated and unirradiated material compared, both groups have similar trends as shown for the 704°C (1300°F) data in Figure 11-7.

Results in the present test series on material irradiated at 593°C (1100°F) or higher show a reduction in yield stress and ultimate strength values compared to samples irradiated at lower temperatures. This reduction is attributed to defect removal by thermal annealing occurring concurrently with defect generation. Elongation values for samples irradiated at the higher temperature were also slightly higher. The reduced elongation values for irradiated material are attributed to both helium embrittlement and matrix hardening due to dislocation interaction. At the higher irradiation temperature, the helium embrittlement is expected to remain while the matrix hardening should be reduced.

Table 11-2
EFFECT OF INCREASING STRAIN RATE ON THE MECHANICAL
PROPERTIES OF IRRADIATED AUSTENITIC STAINLESS STEELS
(Strain Rate Range 0.0002 to 2 min^{-1})

	Test Temperature 593°C (1100°F)		Test Temperature 704°C (1300°F)	
	Irradiated Material	Unirradiated Material	Irradiated Material	Unirradiated Material
0.2% Yield Stress	Increased	Slight increase	Property values higher at strain rate of $2 \times 10^{-2} \text{ min}^{-1}$ than at $2 \times 10^{-4} \text{ min}^{-1}$. No further increase at 2 min^{-1}	Slight increase
Ultimate Tensile Strength	Increased	No definite trend	Increased	Increased
Total Elongation	Increased	Little variation. Highest values at $2 \times 10^{-2} \text{ min}^{-1}$ strain rate	Increased	Little variation. Highest values at $2 \times 10^{-2} \text{ min}^{-1}$ strain rate
Uniform Elongation	Increased	Little variation. Highest values at $2 \times 10^{-2} \text{ min}^{-1}$ strain rate	Increased	Increased

ALLOY	SYMBOL		FLUENCE $\times 10^{22}$ E > 0.1 MeV-GE	IRRADIATION TEMPERATURE	
	UNIRRADIATED	IRRADIATED		°F	°C
304	□	■	5.9-6.7	825-1100	440-593
316L	△	▲	5.9-6.7	750-1100	399-593
321	○	●	3.5-4.5	825-1100	440-593
347	▽	▼	6.0-6.5	825-1100	440-593

Figure 11-3. Effect of Test Temperature on the Mechanical Properties of Austenitic Steels Irradiated in EBR-II (Strain Rate 0.02 Min^{-1})

ALLOY	SYMBOL		FLUENCE $\times 10^{22}$ E > 0.1 MeV-GE	IRRADIATION TEMPERATURE	
	UNIRRADIATED	IRRADIATED		°F	°C
304	□	■	5.9-6.7	825-1100	440-593
316L	△	▲	5.9-6.7	750-1100	399-593
321	○	●	3.5-4.5	825-1100	440-593
347	▽	▼	6.0-6.5	825-1100	440-593

Figure 11-4. Effect of Strain Rate on the 0.2% Yield Stress of Irradiated and Unirradiated Austenitic Stainless Steels

ALLOY	SYMBOL		FLUENCE $\times 10^{22}$ E > 0.1 MeV.-GE	IRRADIATION TEMPERATURE	
	UNIRRADIATED	IRRADIATED		°F	°C
304	□	■	5.9-6.7	825-1100	440-593
316L	△	▲	5.9-6.7	750-1100	399-593
321	○	●	3.5-4.5	825-1100	440-593
347	▽	▼	6.0-6.5	825-1100	440-593

Figure 11-5. Effect of Strain Rate on the Ultimate Tensile Strength of Irradiated and Unirradiated Austenitic Stainless Steels

ALLOY	SYMBOL		FLUENCE $\times 10^{22}$ E > 0.1 MeV-GE	IRRADIATION TEMPERATURE	
	UNIRRADIATED	IRRADIATED		°F	°C
304	□	■	5.9-6.7	825-1100	440-593
316L	△	▲	5.9-6.7	750-1100	399-593
321	○	●	3.5-4.5	825-1100	440-593
347	▽	▼	6.0-6.5	825-1100	440-593

Figure 11-6. Effect of Strain Rate on the Total Elongation of Irradiated and Unirradiated Austenitic Stainless Steels

ALLOY	SYMBOL		FLUENCE $\times 10^{22}$ E > 0.1 MeV-GE	IRRADIATION TEMPERATURE	
	UNIRRADIATED	IRRADIATED		°F	°C
304	□	■	5.9-6.7	825-1100	440-593
316L	△	▲	5.9-6.7	750-1100	399-593
321	○	●	3.5-4.5	825-1100	440-593
347	▽	▼	6.0-6.5	825-1100	440-593

Figure 11-7. Effect of Strain Rate on the Uniform Elongation of Irradiated and Unirradiated Austenitic Stainless Steels

A comparison was made of the data from the present series of tensile tests with results from a similar series of tests on lower fluence L2 materials capsule specimens.²⁴ The major property changes resulting from irradiation hardening occur at fluences less than $3 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV-GE}$). The additional property changes produced by exposure to fluences double this value are comparatively small. It should be noted that the preceding comment applies only to material irradiated at temperatures less than 593°C (1100°F). No conclusions can be made regarding a "saturation fluence" for damage for material irradiated at higher temperatures.

11.4.2 Tensile Tests on High Nickel Alloys (MT-3)

Tensile tests are also complete on the MT-3 high nickel alloy specimens irradiated in EBR-II S/A XA08 to a peak fluence of $6.5 \times 10^{22} \text{ n/cm}^2$ $E > 0.1 \text{ MeV-GE}$. These data are currently being evaluated.

11.4.3 Burst Tests (F3A)

Burst testing of irradiated F3A fuel pin cladding is planned. These unencapsulated pins clad with annealed type 316 have been irradiated in EBR-II Row 5 subassembly X040A for over 6000 hours to a fluence of $\sim 3.1 \times 10^{22} \text{ n/cm}^2$, $> 0.1 \text{ MeV-GE}$. The combined effects of the sodium exposure and irradiation on short-term mechanical properties are to be evaluated. The condition of the cladding as to defects or intergranular attack is being measured nondestructively prior to burst testing by using pulsed eddy current equipment.

A high temperature extensometer has been designed and will be used to measure diametral strain during testing. All necessary equipment including furnaces, nonstandard pressure fittings, and high temperature extensometer is on hand and is being assembled prior to insertion in the VNC hot cells.

11.5 MATERIALS MANAGEMENT

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

11.5.1 Quality Characterization of Tubing

An assessment of the overall quality of a number of lots of austenitic stainless steel tubing, produced as prospective fuel cladding material, is being performed. The producer is a domestic tube mill prominent in the production of heat exchanger tubing. Capability for producing "nuclear grade" cladding for fast reactor fuel has not heretofore been evaluated.

Production and dimensional evaluation of the first three lots of the six-lot order of Type 316H tubing (designated the K3633 tubing in conformance with previous vendor evaluations) was completed during the current quarter. These lots consisted of equal quantities of annealed, 10% and 20% cold-worked tubing, 0.250 inch in outer diameter with a 0.015-inch wall, fabricated from hot extruded billets. Results indicate that of the three conditions tested, the lot of annealed material exhibits the highest degree of conformity to specified dimensions. Poorest of the three in terms of dimensional control was the 10% cold-worked tubing of which 95% of the tubes tested yielded dimensions beyond those specified in the ordering documents. The principal basis for rejection of the latter material was a maximum eccentricity in excess of 1.0 mil, which accounted for a 72% reject level.

Results of individual measurements (based on a 40-tube sampling) are summarized below:

% Cold Worked	Percent Acceptability			
	i.d.	o.d.	Ovality	Eccentricity
0	98	100	100	83
10	90	32	100	28
20	70	100	100	65

On the basis of total acceptability, which assesses the number of tubes that conform to all dimension requirements, the following results were obtained:

% Cold Worked	Total Number of Tube Sections Acceptable (%)
0	80
10	5
20	50

Chemical analysis of the tubing produced from hot extruded billets, performed by an independent analytical laboratory, showed conformance in contents of all specified elements except that of boron. The boron level, restricted by the ordering specification GE-22A1793 to not greater than 0.003% by weight, measured 0.0046%.

Additional testing, including microstructural characterization and wall defect evaluation, is continuing.

Production of the remaining three-lot order of type 316H tubing—that manufactured from drilled bar hollows—has been completed and is undergoing final production testing.

11.5.2 Evaluation of FFTF Fuel Cladding Material

WADCO Corporation of Richland, Washington, provided on request 100 feet of tubing of the FFTF fuel cladding reference design. This Type 316 stainless steel tubing was fabricated by a prominent domestic tube mill to the rigid requirements of the USAEC standard RDT E 13-8. It was produced by the seamless process and was delivered in the as-drawn condition with 20% residual cold work, the numerical level having been established as the magnitude of reduction in cross-sectional area occurring in the final tube drawing sequence. The nominal dimensions of this lot of tubing are as follows:

o.d.: 0.230-inch

i.d.: 0.200-inch

This material was requested from WADCO to permit GE to evaluate the tubing and to compare it with that of material currently being used or produced for the experimental programs. (Material produced by this tube mill has not previously been evaluated by GE.)

The material was scheduled for the following examinations or analyses:

1. Chemical composition
2. Metallography
3. Dimensional measurements
 - Outer diameter/ovality
 - Inner diameter
 - Wall thickness/eccentricity
 - Straightness
4. Surface and wall condition

Results of tests completed to date are summarized below.

Chemical Composition

Chemical analyses of a sample representative of this lot of 20% cold-worked tubing of Type 316H stainless steel showed all elemental constituents, except boron, to be well within the ranges specified by the current issue of ASTM A-213, and the supplementary GE specification 22A1793. The boron level, specified to not exceed 0.003% by weight, was determined to be 0.0035%. When compared with the more restrictive requirements of RDT E 13-8, to which the FFTF tubing was supposedly ordered, the material failed to conform not only to the boron restriction (0.001 maximum) but to the minimum allowable chromium level (17.0 minimum versus 16.45 actual) and to the maximum allowable phosphorous (0.02 maximum versus 0.022 actual).

Dimensional Measurements

Continuous recordings of inner diameter and wall thickness and single measurements for bow have been completed on all tubes. The results indicate that the tubes are in full conformance with the GE specifications as regards inner diameter, eccentricity, and straightness. Moreover, variations in i.d. of all tubes and variations in wall thickness of all but four tubes were found to be less than 0.0005 mil.

Surface Condition

One of the outstanding features of tubing produced by this vendor is that the material is delivered in the as-drawn condition. Witness to this is an ASA average surface roughness of 4 on all tubes. This corresponds to an average peak-to-valley surface irregularity of 15-microinch. Although a surface of this quality is far superior to that called out in the GE specification, the distinct advantage offered by such tubing is readily apparent in the virtual elimination of surface noise that frequently confounds interpretation of pulse-echo ultrasonic signals.

Metallographic characterization, measurements of outer diameter-ovality and ultrasonic measurements for wall condition are currently underway.

11.5.3 Stress-Rupture Behavior of Defected Tubing

Failure of the last remaining specimens in the 704°C (1300°F) test program to assess the influence of defects on stress-rupture properties of cold-worked tubing occurred during the current report period. A specimen—tubular sample of 20% cold-worked type 316H containing a 1.7-mil-deep defect on the inside surface—had a stress-rupture lifetime of 5252 hours at a hoop stress of 9000 psi. Failure was observed to have occurred in the vicinity of the defect. Its companion specimen, containing a notch of similar magnitude on the outside surface, failed after 4430 hours, with the point of failure remote from the defect. Both specimens underwent a peak diametral expansion of 2.4% at the failure point. These results are consistent with those of previous failures in this test program¹ and support the earlier observation that the stress-rupture properties of 20% cold-worked tubing of type 316H stainless steel are virtually insensitive to both inside and outside surface defects as deep as 10% of the tube wall.

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

12.2 ANALYSIS OF INTEGRAL EXPERIMENTS

The program of analysis of integral experiments has had a change in emphasis from that of a general evaluation of nuclear design methods to concentration on the analysis of measurements associated with control, particularly as they relate to a demonstration plant. While this change in emphasis does not negate the preliminary nuclear data work which had been completed, some of this preliminary work has been repeated in order to make use of modified ENDF/B-II data.

Initial calculations have been performed on the ZPPR, Assembly 2, configuration 90. A two-dimensional diffusion theory calculation was performed, giving a k_{eff} value of 1.0041. However, the configuration used for this calculation has an estimated excess reactivity of 0.0007, which makes the computed k_{eff} larger than the measurement by about 0.0034. Preliminary calculations of radial distributions of fission ratios and reaction rates give the following comparison with experiment:

Pu-239 σ_f /U-235 σ_f —Calculations generally 6% low

U-238 σ_f /U-235 σ_f —Calculations 0 to 10% high

Pu-239 fission rate—Within 3% in core

U-238 fission rate—Within 3%, with largest discrepancy at inner core/outer core interface

The experimental data used for these comparisons are preliminary results obtained from Argonne National Laboratory.^{2,5}

The nuclear data used in this analysis are the modified ENDF/B-II set in the GMUG-2 file (see subsection 12.3, Nuclear Data Evaluation and Testing). The TDOWN-2 code was used to obtain self-shielded group constants, and spatial flux averaging across the reactor cells was performed. Spatial condensation from 29 to 6 groups was done prior to running the two-dimensional calculation.

Analysis of measurements with full and partial insertion of simulated control rods in ZPPR-3 will require a series of flux distribution calculations in three dimensions. In order to have the capability for obtaining three-dimensional fluxes, a simple procedure for linking two-dimensional flux sets has been specified and programmed. The input is being prepared and debugging will begin soon. This test code, called TRIAL, will provide only the linking capability, with no special input provisions and minimal output. The flux distributions resulting from this code will be compared with results from a direct three-dimensional flux solution to determine if more complete development of this approach is warranted.

The two-dimensional flux sets linked by TRIAL include those in a number of parallel (x,y) planes, for different axial levels, and a single set in cylindrical (r, z) geometry. In the linking process, the detailed flux distributions provided by the (x, y) sets are preserved in those planes, with their relative magnitudes normalized by taking an average of preselected axial distributions between the planes. These axial distributions are weighted flux distributions taken from the (r, z) flux set. The weighting factors are input, but will generally be fission cross sections. After the (x, y) sets are normalized, fluxes between these planes are interpolated with the aid of the (r, z) fluxes. This procedure is somewhat more elaborate than the linking one would try to perform by hand, but is simpler (and probably less accurate) than formal synthesis approaches. An evaluation of the approach in predicting reaction rate distributions in three dimensions is necessary before development of a production-level code is begun.

The effect of radial temperature distribution across the large SEFOR fuel rod on the effective Doppler coefficient was evaluated. It was found that using the average fuel temperature made less than 0.3% difference in the U-238 Doppler effect, compared to using a realistic radial temperature distribution.

*Contributors are listed in Appendix B.

12.3 NUCLEAR DATA EVALUATION AND TESTING

A 29-group generalized nuclear data file, based on modified ENDF/B-II data, was generated for use with the TDOWN-II code. The modifications have been described in the 38th Quarterly Report¹ and conform closely to those recommended by T. Pitterle, et al.,^{2,6} for U-238 and Pu-239, with the exceptions of a decrease in the U-238 inelastic cross section (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. This 29-group data file is called GMUG-2 and was used for the analysis of ZPPR-2, described in subsection 12.2, Analysis of Integral Experiments, and for the SEFOR benchmark calculations described below.

Computations aimed at defining a Doppler coefficient benchmark problem using static measurements from SEFOR²⁷ are under way. Results indicate that a two-dimensional model will be required for accurate benchmark evaluations. However, an equivalent one-dimensional, axial model, with perpendicular bucklings chosen to give the same Doppler coefficient as the two-dimensional model, will also be included. Corrections to the Doppler coefficient due to model inadequacies, such as heterogeneity effects, will be provided with the benchmark problem.

Preliminary results of calculations performed for the SEFOR benchmark indicate that k_{eff} is about 0.5% high and the calculated Doppler coefficient is lower than the measurement, using modified ENDF/B-II data in the GMUG-2 file. However, all correction factors for the Doppler coefficient have not been determined.

In association with the Cross Section Evaluation Working Group, benchmark calculations for ZPR3-Assembly 11, ZPR3-Assembly 12, ZPR6-Assembly 7, and ZPPR-Assembly 2²⁵ were carried out as part of the Phase II data testing program. The data used are ENDF/B-II with modifications to the principal fuel materials U-235, U-238, and Pu-239, as recommended by Pitterle, et al.^{2,6} These modified data were distributed by Brookhaven National Laboratory as materials 102, 103, and 104. Calculated eigenvalues and central reaction rate ratios are compared with measurements for the four assemblies in Table 12-1.

It can be seen that the eigenvalue calculation is improved for all assemblies by using the modified ENDF/B-II cross sections. The fission ratios in ZPR3-11 and ZPR3-12 are still low but they tend to be closer to the measured values when using modified ENDF/B-II.

Results of the benchmark calculations were presented at the CSEWG meeting of May 19-20, 1971 and, along with the results which were reported from other laboratories, were factored into the working group's decision to create a Version III, ENDF/B data file, which would be consistent with the modified ENDF/B, Version II data set. General Electric was assigned the responsibility for the Pu-239 evaluation and for the Phase I testing (check of the format, completeness, and accuracy of data) of U-238 and iron.

The topical report titled, "Implications of Nuclear Data Uncertainties to LMFBR Design" has been issued.²⁸ This report is an expansion of the paper presented at the Helsinki conference on Nuclear Data for Reactors, in June 1970.²⁹

12.4 METHODS FOR ANALYSIS OF FUEL CYCLES AND MANAGEMENT

The computer code SYN, which combines the two-dimensional flux synthesis code, BISYN, with the fuel management code, BICYCL, has been sent to the Argonne Code Center, and a topical report describing its operational aspects³⁰ has been issued.

Table 12-1
BENCHMARK ANALYSES^(a) RESULTS FOR ZPR3-11, ZPR3-12, ZPPR-2, ZPR6-7

Assembly	ZPR3-11			ZPR3-12			ZPPR-2		ZPR6-7	
	ENDF/B-II	Modified ENDF/B-II	Measure	ENDF/B-II	Modified ENDF/B-II	Measure	Modified ENDF/B-II	Measure	Modified ENDF/B-II	Measure
Reactivity	0.971	0.994	1.000	0.991	0.993	1.000	0.999	1.000	0.990	1.000
$\sigma_f(\text{Pu-239})/\sigma_f(\text{U-235})$	1.13	1.17	1.19	1.07	1.106	1.12	0.918	0.937	0.916	0.955
$\sigma_f(\text{Pu-240})/\sigma_f(\text{U-235})$	0.330	0.326	0.340	0.335	0.336	---	0.178	0.170	0.203	0.194
$\sigma_f(\text{U-238})/\sigma_f(\text{U-235})$	0.034	0.036	0.038	0.044	0.044	0.047	0.0216	0.0201	0.0247	0.0225
$\sigma_c(\text{U-238})/\sigma_f(\text{U-235})$	0.121	0.111	0.112	0.127	0.127	0.123	0.143	---	0.149	---

(a) The results for ZPR3-11 and ZPR3-12 were obtained with S4 calculations in spherical geometry.
The results for ZPR6-7 and ZPPR-2 were obtained with diffusion theory in spherical and cylindrical geometry, respectively.
The eigenvalue for ZPPR-2 includes a 0.0175 correction for heterogeneity.

12.5 METHODS FOR RAPID GENERATION OF GROUP CONSTANTS

The user's manual³¹ for the ENDRUN-2 code has been submitted to the AEC for review, and the final draft of the report on the TDOWN code is nearing completion. ENDRUN-2 processes data from the ENDF/B-II format and produces a GMUG-type multigroup file for rapid data adjusting. TDOWN generates a multigroup data file from a GMUG file which is for a specific reactor geometry and set of compositions.

12.6 COMPUTER CODE COORDINATION AND STANDARDS

In conjunction with the final activities of the *de facto* Committee on Computer Code Coordination, specifications for a standard input card format and an input card sweeper routine were drafted and sent to the other members of the Card Format Subcommittee for further comment. The input card format specified is completely free form with several optional capabilities for repeating, skipping, and overlaying of data. The proposed "Card Sweeper" interprets the data and stores it into a given buffer with appropriate error checking of the card format itself performed internal to the sweeper. It is left as a function of a card processor, which will follow the sweeper, to check the physical validity of the data.

13. TASK P—FUEL ELEMENT MODELING

Task Leader: F. E. Gelhaus*

13.1 GENERAL

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

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

13.2 PERFORMANCE MODEL DEVELOPMENT

The BEHAVE-II fuel element performance code is now operational. This advanced model treats in detail the effects of radial (r, θ plane) and transverse (r, z plane) cracks, including the generation of porosity upon the healing of mismatched crack surfaces. The model properly accounts for the orders-of-magnitude changes in fuel properties by accounting for the axial flow of densified, inviscid fuel above the transition temperature between elastic-plastic and inviscid behavior while simultaneously allowing the cooler, elastic-plastic fuel to slip or flow relative to the cladding. An axial restraint boundary condition may be employed, or the end of the fuel column may be considered unrestrained and free to slide within the cladding.

BEHAVE-II has been used in the analysis of the HEDL-P-19 linear-heat-rating-to-melting test³² and, with the incorporation of transient heat transfer capabilities, is being employed in the analysis of transient overpower TREAT irradiations (see discussion in subsection 4.5, Task C-1).

13.3 FUEL BEHAVIORAL STUDIES

13.3.1 Steady-State Fuel Swelling Model

On the basis of the results of the capsule testing strong radial restraint,³³ the rate of steady-state fuel swelling used in the STEADY subroutine of BEHAVE-I has been taken to be 0.22% per 10^{20} fissions/cc. This rate leads to a prediction of no fuel-cladding mechanical interaction at steady-state for pins irradiated under Task F to date. However, analysis of the data from 19 relatively high linear power pins indicates that a higher swelling rate may occur under the conditions experienced by these pins.

Profilometer data from 19 pins, ranging in average burnup from 5 to 11 at.%, were plotted to emphasize steady-state fuel swelling and to determine what rate is associated with this cladding-straining mechanism. Under the condition of steady-state swelling and external cladding restraint a certain fraction of the as-fabricated porosity of the fuel pellets will be effective in internally absorbing the volume change due to fission-product accumulation. After this porosity has been used up, the outside envelope of the fuel will enlarge to accommodate the additional fuel swelling. In Figure 13-1a the growth of the fuel volume is represented as a linear function of burnup. By definition, the line through the origin is the trajectory for the volume change of the fuel envelope under the assumption that there is no remaining available internal porosity. The slope of this line represents the volume averaged inexorable fuel swelling rate. Lines plotted above and parallel to the lower line correspond to fuel volumes in which there remains some internal available void. In our model available void is defined as all void within the envelope enclosed by the cladding with the exception of the as-fabricated porosity below 1300°C.

According to the model, the cladding diameter is independent of the degree of fuel swelling as long as there is available void remaining, because the low shear strength of the fuel permits the fuel to move and change shape. The trajectories described by the cladding i.d. have been plotted in Figure 13-1a for four hypothetical pins relative to the assumed fuel swelling volume. Pins A and D represent cases where there would be available void throughout the life of the pins, and the end-of-life data points lie above the lower dashed line. Fission-product accumulation in pins B and C, on the other hand, has consumed the available porosity. At the point where the available porosity goes to zero, the fuel loses the ability to deform internally, and the cladding trajectory must thereafter follow the fuel swelling line. Points B and C thus lie on the line passing through the origin while points A and D, since void is still available at the end-of-life, lie in the half-space above that line.

*Contributors are listed in Appendix B.

If data from a large number of fuel pins are plotted in this manner, it will be distributed within the half space above and including the zero percent available void line. No data point should lie below this minimum compacted volume line.

Data for 19 pins with end-of-life peak linear powers ranging between 12 and 16 kW/ft are plotted in Figure 13-1b. The vertical spread between points of identical burnup represents the variation in available void at the end of life plus experimental error. If each of the fuel pins which define the lower edge of the data-space (all of which have smear densities greater than 93% of theoretical) had no available void remaining, then the line described by these points defines the fuel swelling as a function of burnup. The straight line through the origin and these data corresponds to the new fuel swelling rate now used in BEHAVE-I. This result of 0.38% per 10^{20} fissions/cc is approximately 1.7 times the previous value (0.22%). The fact that point F2S lies below the line is assumed to be due to experimental error.

As can be seen from Figure 13-1b, the low burnup data and pin F2G cannot both be fit by the assumption that their deformation resulted from a single valued fuel-swelling rate. This indicates either that there was available void at the end of life and that the steady fuel swelling does not account for any of the observed diametral increase in accord with the observation of a sizable central void volume, or that the fuel-swelling rate is accelerating with burnup. The data are not well ordered enough to provide a clear choice between the two alternatives.

13.3.2 Comparison of BEHAVE-I Predictions with Experimental Data

The steady fuel-swelling rate of 0.38% per 10^{20} fission/cc deduced from the previously described analysis has been introduced into the BEHAVE-I fuel performance code; the mean fuel pin diameter changes have been calculated for 24 mixed-oxide pins irradiated in the EBR-II to burnups ranging from 0.2 to 11 at.%. The present BEHAVE-I model considers four mechanisms to be of importance in the determination of cladding diameter changes: startup, cladding swelling, steady fuel swelling, and power increase after restructuring. Figure 13-2 indicates the degree of correspondence between the predicted mean diameter increase and the observed value obtained from profilometry. The prediction for F2R and F2T is improved compared to earlier predictions made on the basis of a startup-plus-cladding-swelling model⁹ by the addition of a steady fuel-swelling effect; the result for pin FOE is in good agreement with experiment upon the addition of the power increase effect. In the four mechanism BEHAVE-I model, steady fuel swelling is necessary for the explanation of the diameter increase of E1H-30. However, a somewhat higher fuel swelling rate than assumed would be required to account for all of its observed diameter increase. The analytical results for pins F2V and F2G show poor agreement with the post-irradiation profilometry and imply that an important effect not presently being considered in BEHAVE-I is operative.

13.4 DIAMETRAL DUCTILITY PENALTY FOR CLADDING WALL DEFECTS

A theoretical study of the effects of variations of initial wall thickness on the expected diameter of a tube at failure was discussed in subsection 8.3.2 of the 27th Quarterly Progress Report³⁴ and a theoretical model was presented for predictive purposes. The governing equation for longitudinal defects (e.g., scratches) which are narrow with respect to the tube circumference was found to be:

$$\text{Diametral ductility penalty} = 1 - \left(1 - \frac{\Delta h}{h}\right)^q,$$

where

Δh = defect depth,

h = cladding thickness, and

q = creep exponent or reciprocal strain hardening exponent.

In the 32nd Quarterly Progress Report³⁵ data were discussed which indicated that the theoretical model provided a safe prediction (overestimate) of the penalty in diametral ductility experienced in high-temperature short-term tests of defected tubing.

Long-term stress rupture tests of defected tubing at elevated temperature, as described in the Task L subsection 11.5.1 of the 38th Quarterly Progress Report,¹ have been completed. Each specimen was approximately 4 inches long with a longitudinal scratch-like defect about 0.125 inch long, and was pressurized and tested at 1300°F. The key in Figure 13-3 summarizes the material conditions and stress levels used in these tests. Post-test diameters were typically measured at five locations axially spaced 0.5 inch apart. (Note that these data and results derive from and apply to unirradiated material.) The ductility values used in this analysis are the average of the three central measurements, excluding any taken at the point of the failure, and thus are assumed to represent data similar to the uniform

Figure 13-1a. Fuel-Swelling Volume Plus Available Void Volume Plotted versus Burnup for Four Hypothetical Pins

Figure 13-1b. Fuel-Swelling Volume Plus Available Void Volume Plotted versus Burnup for 19 Mixed-Oxide Fuel Pins Irradiated in EBR-II

Figure 13-2. Observed Mean Cladding Diameter Change for 24 Mixed-Oxide Pins Irradiated in a Fast Flux, versus Predicted Mean Cladding Strain on the Basis of a Four-Mechanism Model of BEHAVE-I

Figure 13-3. Diametral Ductility Penalty versus Depth for a Short Longitudinal Defect

elongation ductility in a tensile test. This is assumed to be the most meaningful measure of ductility since it represents the limits of stable geometry under pressurized loading with a gas. Most of the specimens with defects about 10% of the wall thickness failed at locations other than the defect, showing a very weak dependence of failure on defects of this size. Therefore, these data were averaged with the results of undefected tubing and were used in normalizing results of specimens that did fail at the defect.

The analysis of the effect of defects on these ductility results leads to the same conclusion reached for short-term testing: the theoretical model provides a safe prediction of the diametral ductility penalty. The calculated penalties are shown in the figure for both the long-term and short-term tests. The range-bar at the left margin represents the scatter observed in results of multiple tests of control specimens and the data point which shows a 40% improvement for defects is readily explained by this large variation. The weak dependence of failure location on defects of 10% of the wall thickness also shows in these results since these seven data points generally lie within the scatter range.

The conclusions from this analysis are: (1) safe penalties in diametral ductility can be predicted analytically for known defect dimensions, (2) for the material tested the cold work or stress level does not affect the expected ductility penalty, and (3) defects less than 10% of the wall thickness do not carry a diametral ductility penalty.

APPENDIX A. PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Nos. 1 through 108, from July 1959 through June 1971.

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

GEAP-3888

First Quarterly Report
October–December 1961

GEAP-3957

Second Quarterly Report
January–March 1962

GEAP-3981

Third Quarterly Report
April–June 1962

GEAP-4080

Fourth Quarterly Report
July–September 1962

GEAP-4158

Fifth Quarterly Report
October–December 1962

GEAP-4214

Sixth Quarterly Report
January–March 1963

GEAP-4300

Seventh Quarterly Report
April–June 1963

GEAP-4382

Eighth Quarterly Report
July–September 1963

GEAP-4480

Ninth Quarterly Report
October–December 1963

GEAP-4601

Tenth Quarterly Report
January–March 1964

GEAP-4640

Eleventh Quarterly Report
April–June 1964

GEAP-4723

Twelfth Quarterly Report
July–September 1964

GEAP-4795

Thirteenth Quarterly Report
October 1964–January 1965

GEAP-4860

Fourteenth Quarterly Report
February–April 1965

GEAP-4931

Fifteenth Quarterly Report
May–July 1965

GEAP-4982

Sixteenth Quarterly Report
August–October 1965

GEAP-5098

Seventeenth Quarterly Report
November 1965–January 1966

GEAP-5158

Eighteenth Quarterly Report
February–April 1966

GEAP-5198

Nineteenth Quarterly Report
May–July 1966

GEAP-5292

Twentieth Quarterly Report
August–October 1966

GEAP-5433

Twenty-First Quarterly Report
November 1966–January 1967

GEAP-5491

Twenty-Second Quarterly Report
February–April 1967

GEAP-5522

Twenty-Third Quarterly Report
May–July 1967

GEAP-5541

Twenty-Fourth Quarterly Report
August—October 1967

GEAP-5584

Twenty-Fifth Quarterly Report
November 1967—January 1968

GEAP-5631

Twenty-Sixth Quarterly Report
February—April 1968

GEAP-5677

Twenty-Seventh Quarterly Report
May—July 1968

GEAP-5700

Twenty-Eighth Quarterly Report
August—October 1968

GEAP-5753

Twenty-Ninth Quarterly Report
November 1968—January 1969

GEAP-10028

Thirtieth Quarterly Report
February—April 1969

GEAP-10028-31

Thirty-First Quarterly Report
May—July 1969

GEAP-10028-32

Thirty-Second Quarterly Report
August—October 1969

GEAP-10028-33

Thirty-Third Quarterly Report
November 1969—January 1970

GEAP-10028-34

Thirty-Fourth Quarterly Report
February 1970—April 1970

GEAP-10028-35

Thirty-Fifth Quarterly
May 1970—July 1970

GEAP-10028-36

Thirty Sixth Quarterly Report
August 1970—October 1970

GEAP-10028-37

Thirty-Seventh Quarterly Report
November—January 1971

GEAP-10028-38

Thirty-Eighth Quarterly Report
February—April 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

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

GEAP-3646

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

GEAP-3721

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

GEAP-3824

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

GEAP-3833

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

GEAP-3856

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

GEAP-3876

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

GEAP-3880

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

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

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

GEAP-3885

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

GEAP-3923

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

GEAP-4028

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

GEAP-4058

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

GEAP-4090

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

GEAP-4092

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

GEAP-4130

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

GEAP-4426

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

GEAP-4271

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

GEAP-4283

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

GEAP-4350

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

GEAP-4410

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

GEAP-4414

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

GEAP-4418

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

GEAP-4420

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

GEAP-4466

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

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Pre-Irradiated Mixed Oxide (PuO_2 - UO_2),
Irradiations Final Report,
Transient Irradiations;
J. E. Hanson, J. H. Field
(February 1968)

GEAP-4471

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Their Uncertainties in a 1000 MWe FCR;
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GEAP-4472

User Requirements for Cross Sections
in the Energy Range from 100 eV to 100 keV;
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GEAP-4580

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

GEAP-4630

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

GEAP-4675

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

GEAP-4770

A Technical and Economic Evaluation of
Vented Fuel for Sodium-Cooled Fast Ceramic Reactors;
G. L. O'Neill, J. Duffy, J. C. Gilbertson, F. W. Knight,
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GEAP-4787

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

GEAP-4804

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

GEAP-4809

Fast Reactor Meltdown Accidents
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R. Meyer and B. Wolfe
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GEAP-4819

Experimental Studies of
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Series III,
Pre-Irradiated Mixed Oxide ($\text{PuO}_2\text{-UO}_2$) Irradiations,
Interim Report, Steady State Irradiations;
J. E. Hanson and S. A. Rabin
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GEAP-4822

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for Fast Sodium Cooled Reactors;
C. L. Cowan
(November 1966)

GEAP-4870

Some Design and Irradiation Experience
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GEAP-4877

Diffusion of Plutonium and Uranium
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(August 1966)

GEAP-4878

The Solid-Liquid Phase Diagram
for $\text{UO}_2\text{-PuO}_2$ System;
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GEAP-4879

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GEAP-4922

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A Two Dimensional Synthesis Program;
M. D. Kelley
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GEAP-4939

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

GEAP-5271

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

GEAP-5272

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

GEAP-5273

FORE II, A Computational Program for
the Analysis of Steady-State and Transient
Reactor Performance;
B. E. Lawler, et al.
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GEAP-5340

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GEAP-5344

Electron Microanalysis of Radioactive Materials I,
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GEAP-5355A

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B. F. Rider
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GEAP-5411

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D. J. Bender, D. M. Switick, J. H. Field
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C. N. Craig, W. E. Baily
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GEAP-5583

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

GEAP-5633

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

GEAP-5635

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

GEAP-5683

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

GEAP-5744

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

GEAP-5748

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

GEAP-5761

The Irradiation of Mixed Oxide Fuel
to 125,000 MWd/Te in a Thermal Flux;
B. F. Rubin and K. J. Perry
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Fuel with Strong Radial Restraint;
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