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SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

FORTY-SECOND QUARTERLY REPORT

FEBRUARY-APRIL 1972

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

Forty-Second Quarterly Report

February—April 1972

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LIST OF ABBREVIATIONS, SYMBOLS, AND ACRONYMS

AEC-RDT	=	Atomic Energy Commission—Division of Reactor Development and Technology
ANL	=	Argonne National Laboratory
Ann.	=	annealed
Ann. V	=	annealed at VNC
ARC	=	Argonne Reactor Computation (System)
ASTM	=	American Society for Testing Materials
at. %	=	atom percent
bcd	=	binary coded decimal
BEHAVE	=	fuel modeling code
BOL	=	bulk outlet
BRD-QA	=	(GE) Breeder Reactor Division—Quality Assurance
BUBBLE	=	coolant boiling and channel voiding code, a subroutine of FREADM
co-ppt	=	coprecipitated
CSEWG	=	Cross Section Evaluation Working Group
C.W.	=	cold worked
DEMON	=	diffusion theory flux solution computer code
DFR	=	Dounreay Fast Reactor
DRL	=	Division of Reactor Licensing (AEC)
DTA	=	differential thermal analysis
EBR-II	=	Experimental Breeder Reactor II
EM	=	electromagnetic
ENDF/B-II	=	evaluated nuclear data file/B, version II
ENDRUN-2	=	computer code for cross section processing version 2
E.o.L.	=	end-of-life
ETOE-MC ²	=	computer codes for cross section processing
ETR-FEFPL	=	Experimental Test Reactor—Fuel Element Failure Propagation Loop
FCC	=	forced convection capsule
FCI	=	fuel coolant interaction
FFTF	=	fast flux test facility
FGAS	=	fission gas analysis system
FREADM	=	fast reactor excursion and accident dynamics model
FUMBLE	=	fuel management computer code for startup cycles
GETR	=	General Electric Test Reactor
GMUG-2	=	Multigroup data file based on modified ENDF/B-II data
GT	=	Garafolo treated
GTV	=	Garafolo treated at VNC
HCD	=	hypothetical core disassembly

HEDL	=	Hanford Engineering Development Laboratory
HFEF	=	hot fuel examination facility
h_{gap}	=	fuel cladding gap conductivity
i.d.	=	inside diameter
in./in./h	=	inches per inch per hour
I.P.	=	integrated power
kbar	=	thousands of atmospheres
k_{eff}	=	effective neutron multiplication factor
keV	=	kilo electron volts
ksi	=	thousands of pounds per square inch
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
mCi	=	millicurie
MeV	=	million electron volts
MeV-GE	=	General Electric adjusted fluence for EBR-II ⁽¹⁾
mg	=	milligram
Mod	=	modified
mole	=	gram molecular weight
msec	=	millisecond
MWd/Te	=	megawatt-day per ton (metric, tonne, 2205 lb)
ND	=	not determined
nvt	=	time integrated neutron flux
NOM	=	nominal
o.d.	=	outside diameter
O/M	=	oxygen-to-metal ratio
$\%\Delta k$	=	percent reactivity addition
$\%TD$	=	percent of theoretical density of fuel
PEC	=	pulsed eddy current (analyzer)
P.I.	=	postirradiation
PISCES	=	Physics International multispatial code for engineering and science
P_L	=	linear power
P_{max}	=	peak power
PNL	=	Pacific Northwest Laboratory
Phys Mix	=	physically mixed
ppm	=	parts per million
PPS	=	plant protection system

psi	=	pounds per square inch
psia	=	pounds per square inch (absolute)
psig	=	pounds per square inch (gauge)
PUNT	=	service code for maintaining GMUG data files
R	=	roentgen, unit of radiation dosage
RABBLE	=	computer code for special resonance self-shielding homogeneity
SA or S/A	=	subassembly
S Ann.	=	solution annealed
SEFOR	=	Southwest Experimental Fast Oxide Reactor
S.o.L.	=	start of life
SS	=	stainless steel
(ST)	=	solution treated
Stab.	=	stabilized
STP	=	standard temperature and pressure
SUR	=	strong unbroken ring
S _x	=	standard deviation
SYN	=	combined synthesis-fuel cycle code system
TD or T.D.	=	theoretical density
TDOWN-2	=	computer code for cross-section processing, version 2
TEM	=	transmission electron microscopy
Theo	=	theoretical
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
UTS	=	ultimate tensile strength
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
ZPR-2	=	critical assembly 2 in Zero Power Plutonium Reactor facility
ZPR-3	=	critical assembly 3 in Zero Power Plutonium Reactor facility
ZPR-6	=	zero power reactor facility 6
μm	=	micrometers

ABSTRACT

This report is the forty-second in a series of quarterly progress reports published under the Fast Ceramic Reactor Development Program. The major objectives are the development of the essential (critical) portions of a liquid metal cooled Fast Breeder Reactor for reliable, safe operation and the demonstration of low fuel cost potential for such reactors.

Progress during the period February through April 1972 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.

HIGHLIGHTS

- Post-irradiation neutron radiographs and gamma scans of capsule B9D indicate no gross fuel swelling or relocation of fuel or fission products within the sodium loop. The test results represent the first U.S. generated data on the operation of failed mixed oxide fuel elements exposed to clean ($< 5 \text{ ppm O}_2$) flowing sodium.
- Fabrication of the fuel rods for the second in the GE series of grid-spaced fuel subassemblies (F9F) for irradiation in EBR-II is complete. The Fabrication Certification Report has been forwarded to EBR-II for review. Fabrication of the mockup grid-spaced channel and fuel rods is also complete. This mockup of the Mark S-1 subassembly designed specifically for grid-spacer tests will be used for flow testing at GE and for practice assembly—disassembly at EBR-II.
- Two 37-rod grid assemblies continued operation at 1000°F in the Breeder Test Facility for a total of approximately 4400 hours at a maximum sodium velocity of 22.3 ft/second. This test will be completed by the end of April 1972. The post-test cleaning and examination will be initiated in May.
- The single rod hydraulic tests designed to measure the amplitude of vibration of grid spaced fuel rods have been completed and the fabrication of the 217 rod instrumented hydraulic flow test assembly is near completion.
- Stress-rupture tests have been initiated on sections cut from F3A fuel cladding to determine the effect of intergranular attack on long term strength properties. The first specimen, tested at 1100°F and 60,000 psi failed in 1.1 hours at a strain of $0.2\% \Delta D/D$ with a small longitudinal crack.
- The LIFE 2 fuel element modeling code is now available for use on a local IBM computer. The sample problem has been run and compiler differences are being corrected.
- Gamma scans of a fueled rod loaded with four radioactive fission products and heat treated in a thermal gradient indicate that cesium and tellurium combine, and that they may form separate compounds with fuel and that cesium and iodine combine as CsI. Molybdenum does not seem to react with any of these volatile additives.
- The preliminary results of the first post-irradiation examination of GE fuel rods at the LASL facility have been received and are being evaluated.
- Pulsed eddy current testing has been conducted on 18 rods heat treated in a thermal gradient to investigate conditions of fission product attack. The results will guide the metallographic examination and will provide a body of correlative data for appraising the value of this non-destructive test.
- The C4M preliminary data package for ANL/GE/HEDL round-robin calculations was distributed. GE will compute predicted fuel performance for the upcoming ANL H-3 test as part of the round robin with ANL and HEDL. Results of these calculations will be discussed at a scheduled modeling meeting.
- A model for axial fuel motion was incorporated into the TOFF-II fuel failure program.
- In an initial test, hyperstoichiometric fuel was exposed to high temperature (1060°F) sodium for one week; a 6 mil ΔD ($2\% \Delta D/D$) was measured.
- Two experiments (F9C-L and F9C-2) containing preirradiated fuel rods began irradiation to failure in EBR-II. The irradiation is continuing with minimal perturbation of the prior operation conditions.
- The Fuel Void Deployment Experiment (F3B) and the High Cladding Temperature Experiment (F5) have been reinserted in EBR-II after the interim examination for continuation of the irradiation to high burnup.

1. SUMMARY

1.1 TASK B—DEFECTED FUEL BEHAVIOR

The design basis specification for capsule B9B was approved and issued. Preparation of the final design is in process. Capsule B9B will investigate the performance of mixed oxide fuel defected at high burnup and measure the rate of gas release from the plenum of a high burnup rod.

The post-irradiation neutron radiographs and gamma scans of capsule B9D were completed. The results indicate no gross swelling of the fuel, and no relocation of fuel or fission products within the sodium loop. Examination of the capsule hardware has been suspended until July 1972.

The design basis specification for the B11 series of capsules has been approved and issued. This series of tests is designed to determine the change of fuel rod diameter as a function of time caused by fuel-sodium reactions and the volume changes associated with the reactions.

A test proposal has been prepared and is being reviewed internally for a series of experiments to determine the post-failure behavior of mixed oxide fuel rods under typical LMFBR steady-state operating conditions.

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

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

C4M capsule fabrication is continuing. All major machining is complete; the pressure vessel head assembly is nearing completion; dummy fuel rods are ready for assembly of the test array and assembly operations have been initiated. Remote reencapsulation of the C4M test rod has been delayed until FY-73.

The C4M preliminary data package for ANL/GE/HEDL round-robin calculations was distributed.

A Design Basis Specification for C6C/D tests is in internal review. Physics studies of axial and radial power distributions for the Series 6 capsule assembly (three independent inner capsules) are nearly complete.

Due to the termination of the SEFOR Follow-On program, all work ceased for the Task C-1 SEFOR TOP-1A capsule, and project termination functions were completed.

GE will compute predicted fuel performance for the upcoming ANL-H-3 test as part of the round robin with ANL and HEDL. Results of these calculations will be discussed at an upcoming fuel modeling meeting (tentatively scheduled for May 17 through 18, 1972).

A two-dimensional transient heat transfer model of the ANL S-6 test, using the THTE heat transfer code, has completed checkout. This model, coupled with the TOFF-2 transient overpower failure model, will attempt to predict actual time of fuel rod failures that occurred during that test.

GE has supplied HEDL with experimental determinations of transient axial power profile of the TREAT reactor.

GE is preparing a revised version of the format for formal exchange of experimental data for discussion during the above mentioned meeting on fuel modeling.

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

Table 1-1
TASK B-IRRADIATIONS

Design Parameters													
		Fuel							Operating Conditions				Status
Capsule	Objective	O.D. (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 rods)	Fuel-sodium compatibility. Sodium-bonded fuel central temperature versus exposure time.	0.216	2	0.010	80	25	2.00	347	V-Raft GETR	800	14	14,500	Reported in GEAP-13729
		0.216	2	0.010	83	25	1.93	347					
B8C (2 rods)	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel- sodium-clad compatibility.	0.144	2.2	0.006	86	25	1.97	316	V-Raft GETR	1100	21	113,500 57,000	Reported in GEAP-13729
		0.204	2.25	0.017 *	84	25	1.97	316					
B8D	Fuel-sodium compatibility versus density, stoichiometry and Pu content	0.214	1.75	0.0065	80-88	40	1.97	316	V-Raft GETR	950	17.5	41,000 43,500 41,400	Reported in GEAP-13729
		0.214	1.75	0.0065	80-88	25	1.97						
		0.214	1.75	0.0065	80-88	25	1.94						
B9A (2 rods) axially aligned)	Defected fuel rod. Fuel swelling versus stoichiometry in flowing sodium	0.216	5	0.005	89	25	1.99	316	Special V-Raft GETR	1019 971	23.4 23.4	8,600 8,600	Reported in GEAP-13730
		0.216	5	0.005	89	25	1.95						
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 rods)	Swelling behavior of defected fuel rods in flowing sodium	0.216	4.25	0.004	89.6	25	1.97	316 20% CW	Special V-Raft GETR	1015	20.0	~7300 **	Test terminated
		0.216	4.25	0.004	89.6	25	1.97	316 Annealed		1030	20.0	~7300 **	Test terminated
B9D (2 rods)	Two defected fuel rods. 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	~7300	Examination in process.
B9B	Performance of rod defected at high burnup.	0.215	24.0	0.005	85.0	25	1.97	316 20% CW	Special V-Raft GETR	1150	16.0	~30,000	Design in progress.
B11A	Rate of diameter change of defected fuel rod.	0.219	5.0	0.001	87.0	25	2.00	316 30% CW	Trail Cable	1200	23.0	~ 5,000	Design in progress.
B11B	Rate of diameter change of defected fuel rod.	0.219	5.0	0.001	87.0	25	1.97	316 30% CW	Trail Cable	1200	23.0	~ 5,000	Design in progress.
B11C	Rate of diameter change of defected fuel rod.	0.219	3.0	0.001	87.0	25	1.97	316 30% CW	Trail Cable	1200	23.0	~ 5,000	Design in progress.

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

Table 1-2
TASK C-1-EXPERIMENTAL TESTS

Design Parameters										Status							
Capsule	Objective	O.D. (inch)	Fuel Length (inch)	Fuel Density (% TD)		Fuel Composition			Cladding Material	Fabrication	Pre-irradiation	TREAT Transient Irradiation		Examinations [†]			Report
				Pellet	Smear	%PuO ₂	%U-235	O/M				Calibration	Test	Nondestructive		Destructive	
														Before Transient	After Transient		
C4A	Transient failure threshold	0.2159	14.23	93.6	90.4	19.4	0.57	1.983	316SS								
C4B	for zero burnup, mixed-oxide	0.2155	14.21	94.5	91.2	19.4	0.57	1.983	316SS								
C4C	fuel as a function of	0.2164	14.19	83.1	80.4	20.9	0.91	1.983	316SS								
C4D	axial restraint density,	0.2154	14.22	83.7	80.1	20.9	0.91	1.983	316SS								
C4I	and form (pellet versus	powder	13.48	powder	83.5	20.2	0.89	1.973	316SS								GEAP-13562
C4J	powder).	powder	13.45	powder	83.5	20.2	0.89	1.973	316SS								
C4E	Comparison with C4A	0.2159	13.46	83.2	80.5	28	92.7	1.974	316SS	C	C*	P	P	C	P	P	
C4F	and C4B test results (high density pellet fuel)	0.2160	13.51	84	80.5	28	92.8	1.975	316SS	C	C*	P	P	C	P	P	
C4G	Comparison with C4A and	0.2154	13.50	94.4	90.5	28	89.3	1.973	316SS	C	C*	C	C	C	C	C	
C4H	B to determine effect of burnup on failure threshold (transient magnitude will be varied).	0.2152	13.45	94.4	90.4	28	89.3	1.976	316SS	C	C*	C	C	C	C	C	
C4K	Comparison with C4I and J	0.2205	13.50	Powder	84.1	28	93.0	1.972	316SS	C	C*	P	P	C	P	P	
C4L	to determine effect of burnup on failure threshold of powder fuel (transient magnitude will be varied)	0.2206	13.47	Powder	84.2	28	93.0	1.972	316SS	C	C*	P	P	C	P	P	
C4M	Failure threshold for low- power (~10 kW (t)), high burnup fuel (PNL-2-7).	0.2125	13.44	95.2	85.0	25	93.2	—	304SS	I	C	P	P	I	P	P	
C4N	Comparison to C4M (PNL-2-2).	0.2120	13.56	93.4	87.7	25	93.2	—	304SS	I	C	P	P	I	P	P	
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	C	NP	C	C	C	C	C	
C5A	Mobility of transient melted	0.214	24	95	91.5	0	12	2.000	316SS								
C5B	fuel; investigate potential for fuel relocation under prototypical conditions of length, restraint, and fission gas.	0.214	24	95	91	0	12	2.000	316SS								GEAP-13543
C6A-1	Transient performance of	0.212	24	86	80	0	10.0	2.000	316SS								
C6A-2	defected or sodium bonded	0.2223	24	powder	80	0	10.0	2.000	316SS								
C6A-3	fuel; relative performance	0.212	24	86	80	0	10.0	2.000	316SS								GEAP-13722
C6B-1	limits and failure mechanisms	0.212	24	86	80	0	10.0	2.000	316SS								
C6B-2	of pellet and powder fuel	0.2223	24	powder	80	0	10.0	2.000	316SS								
C6B-3	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								

P Planned
I Initiated
C Completed
NP Not Planned

* EBR II fast flux, burnup goal = 50,000 MWd/Te
† Consists of x-rays, neutron radiographs, de-encapsulation, cladding, diametral measurements, gamma transmission scan

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

Additional work has been done on the BUBBLE/VOIDUM code to predict the void formation and collapse detected in the coolant flow and pressure oscillations of the Mockup 1A experiment. The data retrieval algorithm and plotting routine using the GE 635 computer is operational. Mockup 1B data has been digitized, computer plotted, and the graphs are being checked for accuracy. The preliminary design of the C11A capsule has been reviewed and work is progressing into the final capsule design. Preparations to procure the long lead time items for the C11A capsule have been initiated. In response to a revised Task C-2 work scope, the work on the C12A test has been suspended until FY-73.

The status of experimental tests under Task C-2 is shown in Table 1-3.

1.4 TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

Gamma scans of fueled thermal gradient capsules with Cs_2O additions have shown that a cesia-fuel compound forms at O/M ratios > 1.985 . Similarly, Te additions have also shown that a compound forms with mixed oxide fuel.

Additions of Cs and Te show that both may combine with fuel and with each other and migrate down the temperature gradient.

Gamma scans of a fueled thermal gradient capsule containing additions of cesia, tellurium, molybdenum oxide and iodine suggest that during the 100-hour heat treatment the cesia combined with fuel, tellurium, and iodine; the tellurium also combined with fuel, and molybdenum did not react.

Microprobe examination of irradiated specimen E1H-30 showing extensive fuel-fission product-cladding reaction indicated that this six-mil reaction was probably caused by an initial oxide film on the cladding. Fuel and cladding components, along with cesium, tellurium and molybdenum were present. The oxide film may have been a factor in the very uniform $\text{Pu}/(\text{U} + \text{Pu})$ ratio that extended across the fuel pellet.

Capsule E2R, designed to investigate the time for central hole formation, has been irradiated a total of eight hours. Central holes developed in 88% T.D. pellets during the 11- to 26-minute exposure at 2500°C . No central holes have yet formed in the 94% T.D. pellets.

A total of 18 thermal gradient capsules have been tested using pulsed eddy current (PEC) as a nondestructive check of intergranular attack. Rods in this group have been tested from 100 to 1000 hours in temperature ranges of 550 to 870 and 550 to 1100°C . They are awaiting metallographic examination with the PEC traces providing guidance.

The correlation of parameters involved in intergranular fission product attack suggests some relationship with high carbon or high nitrogen content of the fuel. Accordingly, thermal gradient capsules containing fuel with 500 ppm carbon or 500 ppm nitrogen, with and without Cs_2O , have been treated 100 hours in a 550 to 1100°C gradient. They have also been pulsed eddy current tested and are being sectioned for metallographic examination.

1.5 TASK F—FAST FLUX TESTS AND SUPPORT

Fabrication of the F9F fuel rods, and of the mockup grid-spaced channel and fuel rods, is complete. EBR-II reported that the F9F xenon tag ampoules contained well above the minimum acceptable tag gas content. The mockup of the Mark S-1 subassembly designed specifically for grid spacer tests will be used for flow testing at GE and for practice assembly and disassembly at EBR-II.

Preliminary results of the postirradiation examination at the LASL facility of fuel rods F9C-13, F4A, F4E, F12P, and F12Q have been received and are being evaluated. The radiographs show no indications of failure, and the central voids are clearly defined. The F9C-13 rod at 7 at. % burnup is one of the highest burnup unencapsulated rods irradiated in EBR-II. Examination of a sister rod (F9C-31, 1025°F maximum cladding temperature, 84% theoretical fuel density) showed negligible attack at the cladding inside surface. The F4 encapsulated rods with 86% theoretical fuel density reached 9 at. % burnup. The F12P and F12Q rods with 10-mil and 15-mil cladding thickness were irradiated to $> 100,000$ MWd/Te in a thermal flux.

Table 1-3

TASK C-2 EXPERIMENTAL TESTS

Test Number	Objective	Test Facility	Number of Rods	Design Parameters							Steady-State Test Conditions			Status
				o.d. Diameter (inches)	Heated Length (inches)	Fuel Density % TD		Fuel Composition		Cladding Material	Peak kW/ft	Sodium Temperature (°F)	Burnup MWd/Te	
						Pellet	Smear	UO ₂ %U-235	O/M					
C10X	Fuel-cladding response, sodium-fuel interaction, and capsule containment requirements during transient undercooling.	GETR	1	0.250	2	94 94	89 89	25 93*	2 2	316 SS	18 21.6	~1275	Nil	Examination Completed
C11A	Voiding, fuel movement, gas release from a transient undercooling accident	GETR	1	0.230	18	90.4	85.5	35	2	316 SS	12.5	1000	5 to 10,000	Design in Progress
Mockup 1A	Dynamic void behavior during transient undercooling in an out-of-pile loop.	Out-of-Pile Loop	3	0.250	24	←Resistance Heaters→			NA	304 SS	12	800	NA	Report in Progress
Mockup 1B	Flow and void formation during transient undercooling.	Out-of-Pile Loop	3	0.250	24	←Resistance Heaters→			NA	309 SS	12	950	NA	Test Completed

*One pellet at lower end of fuel column to provide hot spot to initiate boiling

NA = Not Applicable

The F9C-1 and F9C-2 experiments, containing preirradiated fuel rods began irradiation to failure in Mark E19 subassemblies. The fuel rods had previously reached 7 at. % burnup, with 14 kW/ft peak start of life power and 1050°F maximum cladding temperature, as part of the F9C experiment. The irradiation is continuing with minimal perturbation of the prior operating conditions.

The F3B (Fuel Void Deployment) and F5 (High Cladding Temperature) experiments were reinserted in EBR-II for Run 55. The experiments had reached 10 at. % and 6 at. % burnup, respectively, before the interim examination. The fuel rods have been combined into a single subassembly (X151) for continuation of the irradiation to high burnup.

The topical report of the F3A experiment has been prepared (GEAP-13735). The F3A experiment containing 16 unencapsulated fuel rods with 1150°F maximum cladding temperature was destructively examined at 6 at. % burnup. Significant attack at the cladding inside surface was observed.

The audit of the F9E experiment fabrication requested by EBR-II is complete. The irradiation of F0A and F0C to 9 at. % is complete, and the fuel rods have been discharged from the reactor. The microprobe examination of fuel rod F5Q is complete, with detailed results reported under Task E. Fabrication of the F2O experiment, which had been stopped temporarily to speed fabrication of the higher priority F9F experiment, has been resumed. Fabrication of the Type-321 stainless steel clad replacement fuel rods for F11A is under way.

The status of the fast flux irradiations in progress under Task F is summarized in Table 1-4. Thirteen subassemblies are currently being irradiated in EBR-II.

1.6 TASK G-6—CORE DESIGN ANALYSIS

The master schedule for grid-spaced driver fuel assembly development has been updated.

Topical report GEAP-13765, "Design Basis of the F9E Grid Spaced LMFBR Diagnostic Assembly for Irradiation in EBR-II," has been transmitted to the AEC for review.

Engineering support was provided for the fabrication of a mockup of F9F, the second in a series of GE grid-spaced fuel assembly experiments planned for irradiation in EBR-II. The mockup was successfully assembled.

An outline of the System Design Description (SDD) of the FFTF grid-spaced driver fuel assembly has been prepared and a draft of several sections is in preparation.

Two 37-rod grid assemblies continued operation at 1000°F in the Breeder Test Facility for a total of approximately 4400 hours at a maximum sodium velocity of 22.3 feet/second. This test will be completed in April. The post-test cleaning and examination will be initiated in May.

A series of single-rod hydraulic tests, designed to measure the amplitude of vibration of a fuel rod in a simulated bundle containing staggered grid spacers were completed. The fabrication of the 217-rod instrumented hydraulic flow test assembly is near completion.

1.7 TASK G-10—SAFETY METHODS AND ANALYSIS

A model for axial fuel motion was incorporated into the TOFF-II fuel failure program.

Reformulation of the VOIDUM loop transient hydraulics program was completed. The new formulation provides for systematic inclusion of valves, pumps, pipe-breaks, etc., in the loop model.

Coupling of TOFF and BUBBLE to FREADM was begun.

Exploratory analyses were performed using PISCES IDL with an energy addition rate given by a functional approximation to that of an ANL MFCI model. The results appear to be valid and suggest an MFCI hydrodynamics model using a sodium equation of state.

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

The assigned GE resident engineer at ANL has continued to provide a link for early exchange of technical information on development programs between ANL and GE-BRD. He has continued his informal communications and exchange of documents and has obtained computer material from ANL.

1.9 TASK L—CLADDING DEVELOPMENT

Irradiation of four L-15 capsules and the MT-4 capsule is continuing in EBR-II S/A X148. The total fluences accumulated to date for these capsules are 3.5×10^{22} n/cm²-GE and 9.1×10^{22} n/cm²-GE, respectively.

Irradiation of the L-16 capsules in EBR-II S/A X098 was terminated when the goal fluence of 3.8×10^{22} n/cm²-GE was attained. Four Type-304 capsules and four Type-316 capsules from X098 will be included with 11 new capsules for reconstitution into S/A X098A, the next subassembly in the L-16 series.

Fabrication of three L-16D capsule inserts is complete. Fabrication of twelve (12) L-16B capsules containing Type 321 was initiated. These capsules are required for reconstitution of S/A X098A.

Transmission electron microscopy (TEM) examination of irradiated samples of 50% cold worked Type-316 cladding (Rod F5Q irradiated at 526°C (979°F) and 625°C (1158°F) to approximately 3×10^{22} n/cm² E > 0.1 MeV-GE) showed very few voids. A significant degree of recovery was detected by means of electron microscopy at 526°C (976°F) and 625°C (1158°F). Analysis of TEM foils from EIH-26 (solution annealed 316) shows that nearly every M₂₃C₆ particle was associated with a void.

Stress-rupture tests at 593°C (1100°F) have been initiated on sections cut from F3A fuel cladding to investigate the influence of intergranular attack on the long-term strength properties of fuel cladding. The initial specimen, tested at 60,000 psi, failed in 1.1 hours. The failure was a small longitudinal crack with a strain of 0.2% $\Delta D/D$. This program has been suspended until FY-73.

The high-pressure stress-rupture rig is being developed to provide a facility for testing fuel rod cladding and unirradiated tubing at higher pressure and temperatures closer to the actual temperature-of-irradiation of Task F fuel rods. This facility will provide stress-rupture data more representative of the as-irradiated fuel cladding by minimizing the thermal effects on cladding microstructure which occur when postirradiation tests are performed at temperatures higher than the temperature of irradiation. Assembly of this test system is in progress.

Biaxial creep tests at 482°C (900°F) on material irradiated to fluences up to 7.2×10^{22} n/cm² E > 0.1 MeV-GE have been terminated after 4,000 hours. One specimen stressed to 35 ksi had failed. A comparison of the creep rates of these specimens with those from an earlier series (fluence 3×10^{22} n/cm² E > 0.1 MeV-GE) confirms that continued irradiation to higher fluences produces a still further reduction in creep rate.

Characterization of four lots of air-melted Type-321H tubing is continuing. Burst tests (high strain rates) were completed on four lots which included annealed, 10% and 20% cold worked material. The results show that increasing the annealing temperature from 1065°C (1950°F) to 1148°C (2100°F) decreases both yield and ultimate strength and increases ductility at temperatures to 593°C (1100°F). The cold work increased hoop strength and decreased ductility. The 20% cold worked tubing was slightly stronger than prototypical FFTF Type-316 20% cold worked tubing. Metallography shows that the tubing annealed at 1065°C (1950°F) has a finer grain size (ASTM-10) than tubing annealed at 1148°C (2100°F) which has a grain size of ASTM-6.

Fabrication of an order for 1500 feet of vacuum-melted Type-321H tubing by Cartech is nearing completion. The 1204°C (2200°F) solution anneal has been completed and the tubing is in final drawing stages.

An order for 2,000 feet of tubing to be fabricated from the Titanium modified Type-316 material available at ORNL has been initiated. The tubing will be ordered to the RDT-13-8T standard with a process sequence believed to represent the best compromise between improved high temperature strength and good ductility after irradiation.

Table 1-4
TASK F FAST FLUX TESTING PROGRAM IN EBR-II (a)

Group	Objective	Total Number of Rods in Group	Date In	Projected Date Out	Actual Date Out	Number of Rods in S/A	S/A	Position	Range of Average Rod Powers (b) kW/ft	Target at. %	Burnup MWd/Te	Average Calculated Burnup (c)		Peak Calculated Fluence (d) (10 ²² n/cm ²)		Status
												at. %	MWd/Te	Total	E > 0.1 MeV	
FO	Low power swelling rate of high density fuel at various burnup levels	5	7/16/65		8/15/68	1	XG02	7A1	7.76	3.91	35000	3.34	30199	2.76	1.99	Post exam complete
			7/16/65		8/ 8/70	2	XG03	7D1	5.47– 5.92	8.95	80000	6.88	62247	5.93	4.32	Reconstituted into XG03A
			7/16/65		8/ 8/70	2	XG04	7B1	5.44– 6.24	11.18	100000	7.62	68916	6.62	4.90	Reconstituted into XG04A
			9/14/70			2	XG03A	8D2	5.49– 6.12	8.95	80000	8.77	79354	7.71	5.58	Irradiation complete
			9/14/70	3/22/73		2	XG04A	8A7	5.34– 6.08	11.18	100000	9.48	85760	8.40	6.16	In reactor
F1	Short time performance and proof test	6	5/ 6/65		5/23/65	6	XG01	4F2	12.32–14.49	0.22	2000	0.21	1882	0.17	0.15	Reported in GEAP-5570
F2	Medium power scoping irradiation. Major parameters include smear density, stoichiometry, cladding-coolant interactions and burnup	21	9/ 3/65		3/ 6/68	9	XG05	4C2	11.42–13.91	8.39	75000	6.69	60543	5.69	4.88	Post exam complete
			9/ 3/65		2/20/67	12	XG06	4E2	8.48–14.83	5.59	50000	5.02	45451	4.22	3.62	Reported in GEAP-13549
			2/23/69		5/18/70	4	X050	4C2	10.73–12.86	11.19	100000	11.75	106358	10.34	8.87	Post exam complete
F3A	Bare rod irradiation to evaluate fuel cladding and cladding-coolant interaction	16	8/15/68		6/30/69	16	X040	5B2	12.87–14.65	5.62	50000	2.88	25959	2.36	2.00	Reconstituted into X040A
			10/ 4/69		5/18/70	16	X040A	5B2	12.53–14.25	5.62	50000	5.17	46620	4.30	3.65	Reported in GEAP-13735
F3B	Density and stoichiometry effects for encapsulated fuel rods	18	11/21/67		5/18/70	18	X027	4B3	11.48–12.80	13.49	120000	8.86	79922	7.54	6.47	Six pins destructively examined
			3/10/72	7/17/73		8	X151	6A4	9.89–11.74	13.49	120000	9.23	83281	7.89	6.76	In reactor
F4	Low density fuel behavior at high heat ratings	9	5/ 9/66		6/28/67	9	X011	2F1	13.74–14.53	5.65	50000	3.64	32758	2.98	2.57	Reconstituted into X039
			8/15/68		12/24/68	7	X039	3C2	13.41–13.83	9.04	80000	4.85	43633	4.00	3.44	Reconstituted into X070
			1/10/70		8/ 8/70	7	X070	3E1	12.72–13.37	9.04	80000	8.25	74217	6.95	5.97	Post exam in progress
F5	Fuel performance at peak FCR cladding temperatures	19	5/28/69		8/ 8/70	19	X064	4F2	11.63–13.75	11.24	100000	5.76	51970	4.80	4.12	Three pins destructively examined
			3/10/72	7/17/73		11	X151	6A4	11.06–12.97	11.24	100000	6.17	55633	5.16	4.42	In reactor
F6	Fuel-cladding interactions under accelerated swelling	4	3/24/66		4/24/69	4	X010	7F3	8.59– 9.34	5.59	50000	5.71	51645	4.80	3.91	Post exam complete
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	12.54–12.65	5.64	50000	5.40	48599	4.56	3.90	Transient tests in TREAT
			1/23/69		5/18/70	4	X050	4C2	11.91–12.25	5.63	50000	5.43	48914	4.49	3.84	Transient tests in TREAT
F8A	Effect of density on fuel swelling rate	16	1/13/67		8/ 2/69	7	X019	6D2	8.53– 9.15	5.62	50000	4.97	44777	4.12	3.42	Post exam complete
			1/13/67		1/31/70	9	X020	6B5	8.37– 9.11	11.25	100000	6.15	55398	5.24	4.38	Reconstituted into X081
			5/24/70	9/3/72		9	X081	7F6	6.64– 8.25	11.24	100000	9.98	89975	8.66	6.98	In reactor
F8B	Effect of void deployment on fuel cladding reaction	19	7/25/68		8/24/70	19	X036	7E1	4.99– 6.39	11.24	100000	4.73	42600	4.01	3.03	Reconstituted into X117 and X118
			2/24/71	6/17/75		9	X117	8B2	4.31– 5.49	11.25	100000	5.79	52161	5.01	3.65	In reactor
			2/24/71	5/ 1/74		10	X118	8D7	4.97– 6.16	11.24	100000	6.72	60612	5.81	4.32	In reactor

Table 1-4 (Continued)

Group	Objective	Total Number of Rods in Group	Date In	Projected Date Out	Actual Date Out	Number of Rods in S/A	S/A	Position	Range of Average Rod Powers (b) kW/ft	Target at. %	Burnup MWd/Te	Average Calculated Burnup (c) at. %	MWd/Te	Peak Calculated Fluence (d) (10 ²² n/cm ²)	Total	E > 0.1 MeV	Status
F9A	High power, high burnup, unencapsulated fuel rod irradiations	37	2/20/69 10/27/70	2/10/73	5/18/70	37 37	X043 X043A	4D2 7D4	12.06–13.36 8.01–10.28	11.25 11.25	100000 100000	5.57 8.23	50189 74213	4.64 7.02	3.98 5.96	Reconstituted into X043A In reactor	
F9B	Medium power, high burnup, unencapsulated fuel rod irradiations	37	5/23/69	2/10/73		37	X062	7F3	7.84–10.23	11.24	100000	8.28	74684	7.07	5.89	In reactor	
F9C	High power, high burnup, unencapsulated fuel rod irradiations	37	4/ 3/69 10/27/70 2/ 2/72 2/ 2/72	4/ 7/72 3/28/73	7/16/70 5/13/71	37 37 19 19	X056 X056A X143 X144	5C2 6B2 6A2 6B4	10.50–11.55 10.20–11.71 10.64–11.70 11.01–12.03	11.24 11.24 11.24	100000 100000 100000	5.67 6.68 7.05 7.18	51157 60207 63546 64740	4.72 5.61 5.95 6.05	4.01 4.76 5.05 5.14	Reconstituted into X056A Reconstituted into X143 and X144 In reactor In reactor	
F9D	Medium power, high burnup, unencapsulated fuel rod irradiations	37	4/24/69	6/19/73		37	X058	7B6	6.86–9.33	11.24	100000	7.55	68072	6.48	5.18	In reactor	
F9E	High power, high burnup, unencapsulated fuel rod irradiations in a grid spacer subassembly	37				37	X145									Shipped to EBR-II	
F9F	High power, high burnup, unencapsulated fuel rod irradiations in a grid spacer subassembly	37				37	X155									Fabrication in progress	
F10A	Cladding performance at peak cladding temperatures	19	2/2/72	7/22/64		19	X121	6D3	11.38–14.34	11.01	100000	1.08	9904	0.90	0.76	In reactor	
F10B	Cladding performance at peak cladding temperatures	19	12/5/71	5/31/74		19	X122	6D4	12.01–14.10	11.25	100000	1.68	15105	1.39	1.17	In reactor	
F10C	Fuel rod performance at peak cladding temperatures	19														Design in progress	
F11A	Cladding performance at peak FCR cladding temperatures	37				37	X141									Shipped to EBR-II	
F20	Irradiation to probe operating limits	19														Fabrication in progress	

(a) All calculated power, burnup, and fluence numbers are based on the fission rates provided in the "Guide for Irradiation Experiments in EBR-II, revision 4," February 1971.

(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 rod 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 rod in the subassembly located closest to the center of the reactor core. Values are based on data supplied by EBR-II.

The open literature on the physical metallurgy of stabilized Cr-Ni austenitic stainless steels has been reviewed with emphasis on Type 321H (Titanium stabilized). It was concluded that the strength of stabilized grades of austenitic steels is highly dependent on prior thermal and mechanical histories. Thus, properties reported as typical for these alloys must be viewed with skepticism as most of the reported data have been generated from steels that were not heat treated for maximum mechanical properties.

The current status of the irradiation program under Task L is shown in Table 1-5.

1.10 TASK P—FUEL ELEMENT MODELING

The LIFE 2 fuel element modeling code is now available for use on a local IBM computer. The sample problem has been run and compiler differences are being corrected.

Planning has been initiated for the development of a fuel element cladding structural code to be used in conjunction with a fuel modeling performance code to compare fuel rod structural design and lifetime criteria with predicted performance. The preliminary development plan will utilize CREEP-PLAST for several exploratory analyses.

Table 1-5
SUMMARY OF TASK-L CAPSULES IN EBR-II

Group	Purpose & Major Parameters	Total Rods In Group	Number of Specimens		Date In	Date Out	Row-S/A	Core Position	Reactor Exposure (MWD)	Target Fluence	S/A Average Fluences ⁽²⁾		Status
			Tubular	Flat Tensile						10^{22} n/cm ² Total	10^{22} n/cm ² Total	>0.1 MeV	
L-2 ⁽³⁾	P.I. Mech. Prop. Eval. of Type-304, 316L, 321, 347, & Incoloy-800.	5	15	80	09/03/65	02/20/67	4-XG06	4E2	9,317	4.0	3.3	2.9	GEAP-10066
L-4	P.I. Mech. Prop. Eval. of Type-316.	2	10	20	03/29/66	11/14/66	4-X009	4A2	5,355	2.0	1.9	1.6	GEAP-10066
L-4 ⁽³⁾	P.I. Mech. Prop. Eval. of Type-304, 321, 347, & Incoloy-800	5	25	50	07/17/66	04/10/67	2-X014	2D1	3,674	2.0	1.6	1.4	GEAP-10066
L-10	P.I. Mech. Prop. Eval. of Type-304, 316, 321, 347, & Incoloy-800	10	70	60	01/13/67	09/ /68	4-X016	4D3 & 4F3	7,894	2.0	2.8	2.4	Report complete
L-14	P.I. Mech. Prop. Eval. of Type-304, 316, 321, 347, & Incoloy-800	19	133	114	10/10/67	12/ /68	4-X025	4E2	8,399	2.0	2.9	2.5	Post exam complete
L-21	P.I. Mech. Prop. Eval. of Type-321	1	3	16	09/03/65	03/06/68	4-XG05	4C2	12,632	7.5	5.4	4.6	Post exam in progress
L-2	P.I. Mech. Prop. Eval. of Type-304, 316, 347, and Incoloy-800	4	12	64	09/03/65	03/06/68	3-X039	3C2	2,532	7.5	9.0	7.5	Post exam in progress
					02/23/69	05/18/70	4-X050	4C2	11,706				
MT-3 ⁽³⁾	P.I. Mech. Prop. Eval. of Incoloy-800, Inconel-625 & Hastelloy-X	4	None	48	12/13/65	06/26/69	4-XA08	4F2	19,280	10.0	6.4	5.5	GEAP-13738
L-6	P.I. Mech. Prop. Eval. of Type-304, 316, 347 & Incoloy-800	3	63	126	12/06/66	01/02/69	2-X018	2B1	10,353	20.0	7.8	6.7	Post exam complete
					04/23/69	06/26/69	4-X018A	4E2	2,334				
					10/02/69	09/10/70	4-X018B	4E2	10,351				
L-15	P.I. Mech. Prop. Eval. of Type-304, 316, 321, 347 & Incoloy 800	4	None	288	08/15/68	12/24/68	3-X039	3C2	2,532	3.0	3.5	3.1	In reactor
					08/24/70	06/10/71	4-X080	6E5	5,351				
					10/20/71	01/00/73 ⁽¹⁾	6-X148	6B3	5,989				
MT-4 ⁽³⁾	P.I. Mech. Prop. Eval. of Incoloy-800, Inconel-625, & Hastelloy-X	1	None	48	12/13/65	06/26/69	4-XA08	4F2	19,280	20.0	9.1	7.9	In reactor
					08/24/70	06/10/71	6-X080	6E5	5,351				
					10/20/71	01/00/73 ⁽¹⁾	6-X148	6B3	5,989				
L-16A	Meas. γ -heating rates for L-16 Type Capsule design	19			—	—	4			0.1			Fabrication in progress
L-16B	Meas. Swelling in Type-316 & 304 Annealed and C.W.	19	76	None	11/16/70	01/00/76 ⁽¹⁾	4-X098	4F2	12,041	3.0	3.8	3.3	In reactor
L-16C	Meas. Temp. Dependence of Swelling in Clad Alloys (Replacement capsule for L-16B)	7		None	—	—	—	—	—				Design in progress
L-16D	Reencapsulation of previously irradiated specimens	3	36	None	—	—	—	—	—		9.0 ⁽⁴⁾	7.5 ⁽⁴⁾	Design in progress

1. Projected assuming EBR-II continues to operate at 62.5MW with 50% Load Factor.

2. Fluence calculated as described in GEAP-10028-37, pages 68-69.

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

4. Fluence prior to reencapsulation.

2. TASK B—DEFECTED FUEL BEHAVIOR

Task Leader: C. N. Craig*

2.1 GENERAL

The use of a sealed plenum fuel rod of a conservative design with proven quality and built with careful attention to detail provides a high level of assurance that the fission products in the fuel will be isolated from the primary cooling system. However, it is also desirable to understand the consequences of failure so that, should failure occur, adequate information is available to the plant operator to permit him to decide upon the appropriate mode of continued plant operation. It is therefore necessary to develop information on the post-failure performance of failed fuel in support of the reactor design, operating limits, and safety analysis.

The specific purpose of Task B is to: 1) determine analytically and experimentally the consequences on the operation of the reactor of failure of a fuel rod under normal or near normal LMFBR operating conditions; 2) provide experimental and analytical data to permit selection of fuel rod 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.

2.2 FORCED CONVECTION CAPSULES (FCC)

2.2.1 B9B Capsule

Capsule B9B is designed to investigate the performance of a defected mixed oxide fuel rod operated beyond failure in contact with high purity (<10 ppm O_2) flowing sodium. The capsule will contain a single mixed oxide fuel rod which will be defected at high burnup ($\sim 25,000$ MWd/Te). After defecting, the rod will be operated for approximately one month to determine the effect of continued operation of a defected fuel rod on the dimensional change of the rod and the release of fuel and fission products from the rod. Two pressure transducers will be connected to the fuel rod plenum to measure the rate of release of fission product gas from the rod after defecting.

The design specification for capsule B9B was completed, approved, and issued. Drafting of the final design was initiated.

The capsule design will include a Fission Gas Analysis System (FGAS). A sketch of the FGAS is given in Figure 2-1. The system will be capable of: 1) monitoring and recording the gross gamma activity of the capsule cover gas; 2) sampling the cover gas above the sodium to enable gamma spectroscopic analysis for short half-life isotopes, and 3) sweeping the cover gas from the capsule or circulating the cover gas past the gross gamma monitor.

2.2.2 B9D Capsule

The purpose of the B9D capsule experiment was to investigate the performance of mixed oxide fuel rods defected at power in high purity (<5 ppm O_2) high temperature (1050 to $1100^\circ F$) sodium and operated for several days after defecting. The capsule contained two mixed oxide fuel rods which were axially aligned. A summary of the fuel and operating parameters has been presented in a previous quarterly report.⁽²⁻¹⁾ The irradiation of capsule B9D was successfully completed in December 1971. All instrumentation in the capsule operated within the design requirements throughout the irradiation.

*Contributors are listed in Appendix B.

Figure 2-1. Schematic of B9B Fission Gas Analysis System (FGAS)

The postirradiation (P.I.) examination of the capsule was initiated during February. All work activities are being performed to a system of documented procedures which meet the Quality Assurance requirements of PA-10. The fuel neutron radiographs, capsule gamma scanning, and removal of the primary containment fission gas were all completed prior to inserting the capsule body into the alpha cell facility at GE-Vallecitos. On March 10, 1972, all destructive examination work on the capsule was suspended until July 1972. The capsule will be stored in the VNC alpha hot cell during the March-July period. Inspection and surveillance on its condition will be conducted weekly until the examination is resumed.

The results of the capsule B9D irradiation test represent the first U.S.-generated data on the operation of failed mixed-oxide fuel elements in a clean flowing sodium coolant. The failure of both fuel rods was confirmed by the postirradiation neutrographs. Figure 2-2 compares the pre- and postirradiation exposures in the 0-degree orientation. The defect plug located on the upper fuel rod (B9D-1) was displaced $\sim 3/4$ inch by the defecting process, while the defect plug on the bottom fuel rod was moved $\sim 1/8$ inch. These numbers agree with the anticipated results. Examination of the postirradiation neutron radiographs indicates that gross fuel-sodium reactions or fuel cladding deformation did not occur. This is in contrast to the results of a similar FCC irradiation test⁽²⁻²⁾ where peak diametral changes of up to 23% were found in the fuel rod cladding. In this earlier test the oxygen level in the sodium was estimated to range from 70 to 226 ppm during irradiation. The B9D preirradiation sodium was determined to have less than 5 ppm of oxygen.⁽²⁻³⁾ A uranium foil getter was incorporated into the design to maintain this oxygen level during the irradiation. A niobium wire in the capsule sodium will be retrieved during the P.I. examination and analyzed to confirm the final sodium purity in the capsule.

The neutron radiographs indicate only one area of localized radial fuel swelling. Immediately adjacent to the bottom fuel rod defect (B9D-2), a 5 to 6 mil bulge of sodium fuel reaction product appears to have extruded into the original defect hole. This represents a localized diameter increase of 2.5%.

Two significant features were noted in the top fuel column (B9D-1): 1) the mixed oxide pellet at the top of the column was cracked during irradiation, and 2) a significant increase in the fuel column length was detected. The 4.25 inch long section of unrestrained fuel increased 0.050 to 0.060 inches in length. The crack in the top pellet was insufficient to account for this increase in length. No change was observed in the post-irradiation length of the axially restrained B9D-2 fuel column. These preliminary results are summarized in Figure 2-3. A more detailed evaluation of the neutron radiographs is in process. The results will be checked by visual examination of the B9D fuel rods when the P.I. examination resumes in July 1972.

The entire capsule body was scanned for gross gamma activity above 100 keV and for selected fission products. The results of the gross gamma scan are shown in Figure 2-4. From the gross gamma scan it is evident that the two fuel rods did not receive the same flux. The upper fuel rod (B9D-1) scan is relatively flat with a center-to-end ratio of 1.04. The lower fuel section (B9D-2) shows considerable variation in the gamma activity from end-to-end of 1.24. Fission product scans of the lower half of the capsule showed no indication of fuel or fission product relocation. Scans of the upper capsule indicated a low concentration of fission gas in the capsule primary containment cover gas. The only other detected activity was a small quantity of Cs-137 located near the sodium-cover gas interface. Sodium samples will be taken during the destructive portion of the P.I. examination and chemically analyzed to verify these findings.

2.3 TECHNICAL SUPPORT FOR IN-PILE DEFECTED FUEL TESTING

2.3.1 Heat Pipe Development.

A review of the B9B capsule design concept and instrumentation was held to evaluate the need for a heat pipe in the B9B capsule. During the review, it was concluded that if pump failure (loss of two phases) was added to the other signals which initiate scram, then the coolant will not boil and will not be a safety hazard. Calculations will be made to confirm this conclusion. In addition, calculations will be made to determine whether the sodium in the capsule will freeze and block further natural convection prior to the time required to remove the decay heat from the capsule. Development work on the heat pipe was discontinued until these calculations are completed and it can be determined if the heat pipe is required.

Figure 2-2. B9D Fuel Rod Neutron Radiograph (0° Orientation)

Figure 2-3. Capsule B9D - Fuel Swelling Data

Figure 2-4. Capsule B9D Gross Gamma Scan

2.3.2 Defect Mechanism Development

No additional work was performed during the past quarter. The results of two scoping tests were reported previously.⁽²⁻⁴⁾

2.4 SODIUM FUEL REACTION STUDIES

The purpose of this series of experiments, designated the B11 series, is to determine:

- The change of fuel rod diameter as a function of time resulting from sodium-fuel reactions.
- The effect of initial fuel O/M on the rate law for reaction between sodium and fuel and the magnitude of fuel rod diameter changes resulting from sodium fuel reactions.
- The reproducibility with which sodium fuel reaction rates may be measured in-pile through measurement of cladding diametral changes.

The design basis specification for capsules B11A, B, and C were approved and issued. The fabrication specification and capsule and fuel rod drawings were completed and these documents were submitted for approval to Engineering and Quality Assurance.

Preparation of the powder for fuel fabrication has begun.

2.5 RUN-BEYOND-FAILURE TESTS

A test proposal was completed for a series of tests to determine the consequences of post-failure operation of mixed-oxide fuel rods under typical LMFBR steady-state operating conditions. The test proposal is undergoing internal review. These tests will determine the consequences of post-failure operation of mixed-oxide-fueled subassemblies containing defected rods operating under typical LMFBR steady-state conditions.

The capabilities and feasibility of conducting run-beyond-failure tests in the EBR-II, SEFOR, ETR-FEFPL, FFTF, and GETR-FCC were studied and a report was transmitted to AEC-RDT.

REFERENCES

- 2-1. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program*, Forty-First Quarterly Report, November 1971—January 1972, February 1972, (GEAP-10028-41), pp 2-1 through 2-9.
- 2-2. Murata, R. E., Craig, C. N., Pfefferlen, H. C., and Novak, P. E., *Effect of Stoichiometry on the Behavior of Mixed-Oxide Fuel During Extended Operation in Failed Pins*, October 1971, (GEAP-13730).
- 2-3. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program*, Forty-First Quarterly Report, November 1971—January 1972, February 1972, (GEAP-10028-41), p 2-9.
- 2-4. Ibid, p 2-9 through 2-14.

3. TASK C-1—SAFETY EXPERIMENTATION (TRANSIENT OVERPOWER)

Task Leader: G. R. Thomas*

3.1 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 and controllability of the movement of molten fuel 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.

3.2 HIGH-BURNUP FUEL EXPERIMENTS

The transient overpower failure thresholds and subsequent fuel-sodium interactions are being investigated for preirradiated 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 $\sim 50,000$ MWd/Te fast-flux burnup in the EBR-II reactor. Included in this inventory are six GE rods irradiated at 13 to 15 kW/ft, and two HEDL-supplied FFTF rods (PNL-2-2 and 2-7) which operated at a lower power level of ~ 10 kW/ft.

3.2.1 Capsule Fabrication and Assembly

Machining operations are essentially complete for the C4M capsule and assembly operations have been initiated. The pressure vessel head assembly has completed a preliminary hydrostatic pressure test (to 5,000 psia), heat treatment and final machining operations (surface finishing). The head assembly will be qualified for TREAT use by an additional hydrostatic test to 15,000 psia. Dummy fuel rods previously used in the C4X test (identical capsule) have completed upgrading for use in the C4M capsule. The inner capsule upper bulkhead has completed assembly steps involving the mounting and welding of brazed tubes for instrumentation and sodium-filling penetrations (17 total penetrations). Braze qualification for these penetrations is nearly complete.

The major portion of the Detailed Assembly Instructions for the C4M capsule assembly have been written and approved by Task C-1 and BRD QA.

The completion date for the C4M capsule has moved from a late June date to a probable September date. Efforts involving preparation for remote reencapsulation have been suspended until after the end of FY-72 (June 30, 1972). In addition, C4M capsule fabrication is now planned to include an orderly interruption, with resumption of assembly effort (primarily the remote reencapsulation portion) at the beginning of FY-73.

The Series 4 capsule design has been modified to eliminate the pressure transducer and associated instrumentation. The pressure transducer system was removed because technical evaluation indicated that: (1) the signal obtained during a test would be severely distorted due to interfering capsule geometry; (2) the combination of TREAT Hodoscope and capsule flowmeter provides sufficient data for determination of sodium and fuel movement and interaction. In addition, removal of the pressure transducer significantly decreases the complexity of remote fabrication. The removal of the pressure transducer improves the ultimate safety of the capsule, and improves quality assurance.

*Contributors are listed in Appendix B.

3.2.2 TREAT Irradiation Test Planning

The preliminary data package for the C4M test round-robin calculations has been distributed to HEDL, ANL, and AEC-RDT. This package contains sufficient detail for HEDL/ANL to set up complete thermal analysis models, including a complete description of radial power (both neutronic and gamma contributions) for the entire inner capsule. A second package will be issued at a later date, and will include a specific TREAT power history (shaped) and recalculated physics prediction for radial power profiles for the inner capsules based on the actual C4M fuel rod (PNL-2-7) fabrication/irradiation history as described in a data package received from HEDL during March (HEDL CTS TM: 72:01).

3.2.3 Postirradiation Analysis

Microprobe analysis of three specimens from the C4G fuel column is complete except for final data reduction. The results (Pu-U migration) will be used in analysis of radial power generation in the C4G-H tests.

A coupled EBR-II TREAT modeling prediction package using THTE and BEHAVE-2 has completed initial checkout and is being tested on the C4G EBR-II-TREAT irradiation history. The purpose of the coupled package is to analytically predict the cladding deformation history from as-fabricated condition to completion of the TREAT irradiation including a checkout of the suspected cladding damage (pre-fuel melting) resulting from the rapid startup characteristics of TREAT irradiation.

3.3 COMPARATIVE TESTS

A draft of the C6C/D capsule assembly design basis specification is in internal review. A physics analysis of the capsule assembly TREAT characteristics is nearly complete, including definition of flux filter configurations for axial power shaping.

It is planned that these two capsule assemblies (each containing three independently encapsulated fuel rods) will be used to establish the differential effect on cladding damage/failure threshold resulting from different steady-state power levels. The fuel rods will be chosen from those available from EBR-II irradiations, with nominal values of $\sim 50,000$ MWd/Te burnup and steady-state power levels ranging from ~ 7 to 15 kW/ft.

3.4 SEFOR CAPSULE TESTS

As a result of cancellation of the SEFOR Follow-on Program, work on the Task C-1 SEFOR capsule TOP-1A has been discontinued. The project has completed termination phase consisting of a consolidation of existing documentation. Specific areas of the completed capsule design and the existing TOP-1A fuel rod hardware will be evaluated for potential use in future Task C-1 test series or other PA-10 applications.

3.5 MODEL DEVELOPMENT

As a result of discussions with RDT/HEDL/GE, Task C-1 will participate in round-robin calculations on the upcoming ANL H-3 test. The resulting round-robin calculations (independent ANL, GE, and HEDL input) will be subjected to comparative analysis and discussion during the next transient fuel behavior modeling meeting (tentatively scheduled for May 17 and 18 at GE). Data packages describing the H-3 test were received from HEDL in mid-April. These packages are being used as the basis for the GE analysis of H-3.

The ANL S-6 test has been modeled in two dimensions on the THTE program to provide input for TOFF. The thermocouple trace for the S-6 test has been received from ANL. Preliminary calculations on the THTE model have shown that it will be necessary to include the thermocouple in the THTE model in order to match the experimental thermocouple trace. A definition of the thermocouple configuration has been received from ANL, and a more sophisticated fission/gamma heating model has been completed by BRD physics. A TOFF-2 code checkout using the S-6 preliminary THTE model (one-dimensional unit cell) resulted in a very promising initial correlation of predicted versus actual failure time. When ready, the above described two-dimensional model will be applied to TOFF-2.

3.6 LIAISON

Efforts have been initiated to establish the identity of potential EBR-II-irradiated fuel specimens as candidates for use in future transient-overpower tests (e.g., C6 capsule assemblies or Series 4 capsules).

A memo has been distributed to HEDL describing experimental measurements of the integrated TREAT axial power profile (determined during the C4B and C and C6A and B tests).

A revised format for formal exchange of experimental data from LMFBR safety development tests is being prepared for discussion at the upcoming transient fuel behavior modeling meeting.

Task C-1 is involved in discussions with ANL on test objectives and schedule for the upcoming ANL S-11 and -12 tests investigating the interactions of vapor-driven molten fuel with coolant.

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

Task Leader: H. C. Pfefferlen*

4.1 GENERAL

A major safety-related area that is of concern to both the FFTF and the LMFBR programs is the potential for rapid reduction or loss-of-coolant flow to localized portions of the core. Such coolant starvation might be a result of local blockages, fuel distortion, debris, or failure of the pumps. The fuel response to such starvation, the propagation of malfunction to neighboring rods, 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.

Various analytical studies have investigated the effects of loss of flow, 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.

4.2 OUT-OF-PILE TRANSIENT UNDERCOOLING TESTS

The out-of-pile experiments are designed to simulate features of subsequent in-pile transient undercooling tests. The out-of-pile test assembly consists of a concentric loop with an EM pump and three replaceable 24-inch-long resistance heaters to simulate fuel rods. The loopsule is instrumented to measure temperature, pressure, flow, and vibrations (accelerometers).

4.2.1 Mockup 1A

Mockup 1A was the first out-of-pile transient undercooling test.⁽⁴⁻¹⁾ The purpose of this experiment was to check out the loop operation and instrumentation response during a loss-of-flow test.

The Mockup 1A test conditions were input into the BUBBLE/VOIDUM code for transient thermal hydraulic analysis. The initial results of the analysis overestimated the extent of voiding. Additional work was done on the BUBBLE/VOIDUM code and the program was modified to account for coolant slug reentry after it had been expelled from the top of the channel. In its current form the code predicts the void formation and collapse phenomenon observed in the Mockup 1A test. However, the code predicts a shorter period of void formation and collapse than is inferred from the coolant flow and pressure oscillations measured in the Mockup 1A experiment. Possible causes are being investigated and the code is currently being revised.

*Contributors are listed in Appendix B.

4.2.2 Mockup 1B

Mockup 1B was used to investigate the initial stages of loss-of-flow accidents in an out-of-pile loop experiment utilizing the same facility as the Mockup 1A test. Three fuel rods, operating at ~ 12 kW/ft, were simulated in this test by resistance heaters with 24-inch heated lengths. The investigation as reported earlier,⁽⁴⁻²⁾ included the flow transient as well as the initial stages of void-formation.

Mockup 1B data reduction and interpretation is continuing. The data retrieval algorithm and plotting routine using the GE-635 computer and plotter is operational. Temperature, pressure, and flow data, which has been digitized from oscillograph records, has been plotted and the graphs are being checked for accuracy and resolution of significant details.

A draft of the procedure for the removal of the Mockup 1B test section has been prepared and is under review. The procedure includes draining the sodium from the loop and provides for "moth balling" the facility.

4.3 SERIES C11

The C11 series of tests will investigate transient undercooling accidents by utilizing stagnant sodium capsules and the GE Test Reactor Facility. Capsule C11A is an in-pile vehicle for the study of the latter stages of a transient undercooling accident. This is a small, nonflowing capsule in which boiling will be initiated in a manner similar to that used on the C10X capsule. In this test the upper portion of a fuel rod will be simulated, including the fission gas plenum. This test will investigate the voiding process, fuel rod failure mechanism, post-failure fuel movement and fission gas release rate resulting from a local undercooling condition.

The design basis specification for Capsule C11A was sent out for review and comment under the new Quality Assurance Project Plan PA-10. Engineering change notices incorporating the comments are being prepared to update the document.

The Product Quality Plan for use on Task C-2 GETR tests was prepared, reviewed, and approved in accordance with the Quality Assurance Project Plan PA-10.

The C11A capsule has proceeded through its preliminary design. The design review has been completed and the modifications are being incorporated into the final capsule design. Work is progressing on the two-dimensional heat transfer analysis of the capsule for both steady-state and transient test conditions. The thermal analysis will also be used to assess the potential for damage due to sodium solidification and remelt cycles during the experiment. The analysis of the fission gas distribution in the C11A fuel rod at the target burnup has been initiated. The analysis and design for the annular EM flowmeter for the C11A capsule was completed.

Procurement of instrumentation for the C11A capsule was initiated. A vendor for the C11A pressure transducer system was reviewed for qualification to RDT standard C6-3T. A procurement specification, which includes RDT standard C7-67T, is being prepared for the required thermocouples.

The C11A Fuel Rod Fabrication Specification has been prepared and is being reviewed. The dies have been ordered for fabricating the enriched UO_2 fuel pellets. The following FFTF Reactor grade materials and parts for the fuel rod were requested from HEDL: cladding (RDT E13-8T), reflectors (RDT E13-10T), plenum spring (RDT E13-12T), plenum spacer (RDT E13-11T), insulator pellets (RDT E13-7T), and end cap material (RDT E13-9T). These materials and parts identified as acceptable for a FFTF Driver Fuel Assembly were shipped from HEDL to BRD and are awaiting receiving inspection.

The fuel rod will be pressurized to simulate the plenum pressures expected at high burnups. The fuel rod pressurization and pinchoff technique has been demonstrated successfully on a mockup.

4.4 SERIES C12

The C12 series of tests were planned to investigate transient undercooling accidents using flowing sodium multirod loops.

The first test in this series, C12A, was planned to be run in an ANL loop in the TREAT reactor. Both the ANL Mark IIA loop and the "R" loop were being evaluated with regard to C12 test objectives. A draft of the Experiment Proposal has been reviewed, and the comments received for preparation of the document in final form. A draft of the Design Basis Specification has been prepared. In response to a revised project plan, the work on the C12A test has been discontinued and no additional effort is planned.

4.5 DATA RETRIEVAL SYSTEM

A procurement specification for a Digital Data Retrieval System for transient undercooling tests has been completed. As a result of revised project funding, the procurement of this system will not be initiated.

REFERENCES

- 4-1. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Thirty-Third Quarterly Report, November 1969—January 1970, February 1970, (GEAP-10028-33).*
- 4-2. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Fortieth Quarterly Report, August—October 1971, November 1971, (GEAP-10028-40).*

5. TASK E—FUEL PROPERTIES AND PERFORMANCE EVALUATION

Task Leader: B. F. Rubin*

5.1 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 used 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 and identified with an electron probe microanalyzer, electron microscopes, X-ray diffractometer, and by simulated out-of-pile testing. Methods of limiting these reactions are being explored. The radial and axial migration of impurities, fission products, and fuel constituents are being studied in irradiated samples. The behavior of mixed oxide in the presence of hot sodium is being monitored at several levels of fuel O/M, Pu content, sodium temperature, and sodium oxygen content.

5.2 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR

5.2.1 Central-Hole Formation at Reactor Startup—Capsule E2R

This capsule, containing 88% and 94% T.D. mixed-oxide pellets, is designed 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 inserted eight times: once each for periods of 1, 10, 15, 34, and 60 minutes, and three times for a period of 120 minutes, making a total exposure of eight hours. Neutrographs taken after each exposure revealed that the central hole had formed in the 88% T.D. pellets at some time during the 15-minute exposure, between 11 and 26 minutes. The central void measured 25 to 30 mils. No central hole had yet formed in the 94% T.D. pellets.

5.3 ELECTRON MICROPROBE EXAMINATIONS

5.3.1.1 Specimen E1H-30J of the F3B Test (~8 at. % analyzed burnup) (Sample location 13 inches from fuel bottom)

Initial results from the examination of this sample were reported last quarter.⁽⁵⁻¹⁾ This sample was notable for the broad bands, up to six mils in width, of fuel-fission product-cladding reaction product. It was clad in cold worked Type-304 tubing that had been given an extended heat treatment to improve its creep properties.**

The reaction product band was examined at the widest, narrowest, and one intermediate location for elemental composition and distribution. Point count concentrations for iron, chromium, nickel, cesium, tellurium, molybdenum, uranium, and plutonium were taken with the energy dispersive spectrometer system at all three locations and were run through the MAGIC-IV Computer Code which reduced the values to quantitative concentrations.

The results of the examination at the region where the band was widest are presented in Figure 5-1. This figure shows that the ring included fuel, cladding alloy elements and fission products. The urania and plutonia essentially maintained their original 3/1 ratio across the reaction zone and into a slight indentation of the cladding, as shown in the metallographic view. Point counting extended nearly a mil into the cladding. Within this cladding region, a compositional deviation from the matrix material was detected, with the chromium reduced from 19% to 14%, and iron proportionately higher.

The chromium may have been reduced by the formation of the thin surface film believed to be Cr_2O_3 that was observed on the cladding after the 1300°F heat treatment. Its formation was attributed to insufficient protection from the atmosphere. Chromium also migrated into the reaction zone, reaching its peak value (about 19%) near the middle of the band; the chromium concentration declined in the direction of the fuel, as had been indicated by the X-ray image in Figure 5-5 of Reference 5-1. The nickel content of the reaction band was nearly constant across the half closest to the

*Contributors are listed in Appendix B.

**The heat treatment consisted of 24 hours at 900°F, followed by 216 hours at 1300°F.

Figure 5-1. Elemental Composition of Fuel-Fission Product. Cladding Reaction Zone of E1H-30J. (Compositions are Normalized to 100%)

cladding, then it increased near the fuel interface in contrast to the behavior of iron and chromium. Molybdenum appeared constant at <1% while cesium and tellurium fluctuated across the band in no discernible relationship. A number of these elements existed as oxides. The radial oxygen distribution within this sample of (U-25 wt % Pu)O_{2.00} was also determined. The oxygen X-ray intensity was compared to the intensity of a standard of the same composition and the result is plotted in Figure 5-2. This intensity ratio was also converted to O/M values using the calibration curve described last quarter,⁽⁵⁻²⁾ and the oxygen distribution is shown in these units as well. The experimental precision of the O/M is estimated to be ± 0.02 . The wide fuel-fission product-cladding reaction band exhibits a high oxygen concentration, peaked near the center; this ring apparently served as an oxygen sink, collecting oxygen from all regions of the fuel. The low oxygen content of the fuel near the center may have prevented any uranium oxidation and consequent vaporization, so that a flat profile of Pu/(Pu+U) resulted.

5.3.1.2 Specimen E1H-30F (~7.7 at. % analyzed burnup)
(Specimen location 7-1/2 inches from fuel bottom)

This sample showed a reaction layer at the fuel-cladding interface that was similar to, but narrower than, that of the previous sample displayed in Figures 5-1 and 5-2. The radial Pu/(Pu+U) ratio was determined and is plotted in Figure 5-3. This shows a slightly greater compositional change than in the E1H-30J specimen, possibly due to the steeper thermal gradient at this location near the peak flux position. The narrower reaction rim may have been a factor also since the oxygen concentration of the fuel, which participates in the actinide migration, would not have been reduced to the low extent of sample E1H-30J; both had the same O/M original ratio.

5.3.1.3 Specimen E1H-23K (~7 at. % analyzed burnup)
(Specimen location 11 inches from fuel bottom)

The radial Pu/(U+Pu) ratio for this annular pellet was determined and is plotted in Figure 5-3(b). The figure shows the more characteristic increase in the plutonium concentration at the center. A dip appears near the mid-radius position (the outer extremity of the columnar grain region).

5.3.1.4 (Specimen F5Q-H1 (4.9 at. % analyzed burnup)
(Specimen location 9.8 inches from fuel bottom)

This metallographic specimen, which was clad in 50% cold worked Type-316 stainless steel, was illustrated in a previous report.⁽⁵⁻³⁾ It included a unique metallic-appearing ring adjacent to the cladding. The results of the microprobe examination are presented in Figures 5-4 through 5-7. Figure 5-4 includes a view of the ring along with X-ray images of uranium and plutonium. These show that contrary to their behavior in the reaction band of E1H-30, these elements did not participate in the chemical reaction; the fuel components are clearly delineated from the metallic ring. The uranium appears in the narrow region between the rim and the cladding, possibly as a vapor-deposited oxide. In other regions that were examined, uranium or plutonium appeared within the ring.

Figure 5-5 shows spatial concentrations of iron, chromium, and nickel in the vicinity of the rim. The X-ray images show that the rim is composed primarily of iron and nickel. Surprisingly, chromium does not appear within the rim or in the nearby fuel.

Figure 5-6 shows X-ray images of fission products tellurium, molybdenum, cesium, and palladium. These indicate that cesium has deposited on both sides of the metallic ring but is not appreciably within it. Tellurium is confined to the side of the ring adjacent to the cladding. Molybdenum and palladium primarily appear within the gap next to the cladding.

The actual composition of the ring and adjacent areas was determined by point counting at $\sim 8 \mu\text{m}$ intervals in the manner used to develop Figure 5-1. These results are presented in Figure 5-7 after normalizing the compositions of the nine analyzed elements to 100%. The elemental distribution is in agreement with the X-ray images of the earlier figures. The ring is essentially free of fuel and consists of iron (to 73%), nickel (14%), and chromium (5 to 12%), and a small percentage of metallic fission products. For the most part, the fission products are concentrated outside and on both sides of the metallic ring. The cesium, tellurium, and molybdenum have atom ratios of about 6:1:20. These three elements are commonly detected in microprobe investigations of cladding attack.

Figure 5-2. Photomicrograph of Specimen E1H-30J and the Corresponding Radial Distributions of Oxygen and Pu/(U + Pu)

(a) SPECIMEN E1H-30F

(b) SPECIMEN E1H-23K

Figure 5-3. Photomicrographs of Specimens E1H-30F and E1H-23K and the Corresponding Radial Distributions of Pu/(Pu + U)

NOTE: LINE a IN TOP LEFT VIEW IS LOCATION WHERE ELEMENT ACTIVITY, PLOTTED IN FIGURE 5-7, WAS COUNTED

Figure 5-4. Photomicrograph, Specimen Current Image and Uranium and Plutonium X-Ray Images of Area Showing the Metallic Ring in F5Q-H1

2MA-21A

500X

POSITIVE SPECIMEN CURRENT IMAGE

FUEL

METALLIC
RING

CLADDING

2MA-21D

500X

CHROMIUM X-RAY IMAGE

2MA-21C

500X

IRON X-RAY IMAGE

2MA-21E

500X

NICKEL X-RAY IMAGE

Figure 5-5. Specimen Current Image, and Iron, Chromium and Nickel X-Ray Images of Metallic Ring in F5Q-H1

Figure 5-6. Tellurium, Molybdenum, Cesium and Palladium X-Ray Images of Fuel Cladding Areas of F5Q-H1 Containing the Metallic Ring

Figure 5-7. Results from Point Counting Across Cladding-Metallic Ring-Fuel at Position Indicated on Photomicrograph in Figure 5-4. These Nine Elements Were Normalized to 100%.

Figure 5-8 shows the Pu/(Pu+U) ratio across a radius of this sample. This trace is similar to traces observed in other samples, such as that of E1H-23 in Figure 5-3. Distinctive features are the dip in the columnar grain-region of the pellet and the increase near the center void which amounts to a total of 3 to 4%.

Figure 5-8. Photomicrograph of Specimen F5Q-H1 and Corresponding Radial Distribution of Pu/(Pu + U)

5.4 FUEL-FISSION PRODUCT-CLADDING REACTIONS

5.4.1 Fission Product Migration and Reactions

Reference 5-1 described the intergranular attack that free tellurium or tellurium exceeding a Cs_2Te relationship may cause in stainless steel cladding when it is heat treated at a suitably high oxygen potential ($\text{O/M ratio} \geq 2.00$). Additional experiments were conducted with radioactive tracers to investigate this reaction further. The testing and gamma counting results are summarized in Table 5-1. Test capsules 8-1 and 8-4 were loaded with Te-128 and substoichiometric and stoichiometric fuel. The 100-hour heat treatment over the 550 to 1100°C range of temperatures caused the tellurium distribution of Figure 5-9 to take place. This shows that the tellurium behaved identically in both oxygen environments and remained at the high temperature end of the capsule. Despite its high vapor pressure, the tellurium apparently combined with fuel as an uranium-plutonium tellurite or tellurate. The compound(s) is stable within the O/M range of 1.97 to 2.00.

Similar experiments were conducted with stoichiometric fuel and with cesium and tellurium additions in 1/1, 2/1, and 4/1 atomic ratios. The results from these three capsules are presented in Figure 5-10. This shows that one or both elements combine with fuel and with each other at the hot end and the cesium-tellurium compound(s) migrates to the cooler end. Discrete compounds apparently deposit en route at temperatures of about 900 and 1050°C. The gamma counts did not indicate any difference due to the different cesium-tellurium ratios.

Capsule 8-6 was prepared to investigate interrelations between the major elements identified in Figure 5-7: cesium, tellurium, molybdenum, and also iodine. The additions were in the cesium ratios of 4/1, 1/1, and 10/1. Results are shown in Figure 5-11. Cesium and tellurium once again combined and migrated together down the thermal gradient although some of the cesium has concentrated at the cold end of the capsule, where the iodine has deposited, as CsI. The molybdenum, added as MoO_3 , remained at the hot end of the capsule indicating that it was reduced to the stable metal early in the cycle. These findings show that cesium and tellurium react with mixed oxide and that cesium may also coexist as both the iodide and telluride or tellurite.

The capsules have been pulsed eddy current tested and are awaiting metallographic examination of the degree of intergranular attack.

Table 5-1
SUMMARY OF THERMAL GRADIENT CAPSULE EXPERIMENTS⁽¹⁾

	O/M	Temperature Range (°C)	Container	Additive	Results
Capsule 8-1 ⁽²⁾	1.97	550-1150	316 SS	$\text{Te}^{(4)}$	Te reacted with fuel, remained at hot end.
Capsule 8-2	2.00	550-1150	316 SS	$\text{Cs}_2\text{O} + \text{Te}^{(3,4)}$	Cs + Te combined with each other and fuel, drifted to cool end.
Capsule 8-3	2.00	550-1150	316 SS	$\text{Cs}_2\text{O} + \text{Te}^{(4,5)}$	
Capsule 8-4	2.00	550-1150	316 SS	$\text{Te}^{(6)}$	Te reacted with fuel, remained at hot end.
Capsule 8-5	2.00	550-1150	316 SS	$\text{Cs}_2\text{O} + \text{Te}^{(6,7)}$	Cs + Te combined with each other and fuel, drifted to cool end.
Capsule 8-6	2.00	550-1150	316 SS	$\text{Cs}_2 + \text{Te}^{(5)} + \text{Mo} + \text{I}$	Same as capsules 2, 3, and 5. Cs also combined with I as CsI. Mo did not react.
Capsule 9-1 ⁽⁸⁾	2.00	500-750	316 SS	Cs	
Capsule 9-2 ⁽⁸⁾	2.00	500-750	316 SS	$\text{Cs}_2\text{O} + \text{Te}^{(3)}$	
Capsule 9-3 ⁽⁸⁾	2.00	500-750	316 SS	$\text{Cs}_2\text{O} + \text{Te}^{(5)}$	
Capsule 9-4 ⁽⁹⁾	2.00	500-750	316 SS	Cs	
Capsule 9-5 ⁽⁹⁾	2.00	500-750	316 SS	$\text{Cs}_2\text{O} + \text{Te}^{(3)}$	
Capsule 9-6 ⁽⁹⁾	2.00	500-750	316 SS	$\text{Cs}_2\text{O} + \text{Te}^{(5)}$	
Capsule 9-7 ⁽¹⁰⁾	2.00	500-750	321 Stab	Cs_2O	
Capsule 9-8 ⁽¹⁰⁾	2.00	500-750	321 ST	$\text{Cs}_2\text{O} + \text{Te}^{(3)}$	
Capsule 9-9 ⁽¹⁰⁾	2.00	500-750	347 ST	$\text{Cs}_2\text{O} + \text{Te}^{(5)}$	

NOTES

1. Results of earlier testing in this program are tabulated on Page 5-19 of Reference 1.
2. These fueled capsules were heated 100 hours.
3. 1/1 ratio of Cs/Te.
4. Te addition was 12-15 mg.
5. 4/1 ratio of Cs/Te.
6. Te addition was 50 mg.
7. 2/1 ratio of Cs/Te.
8. Removed for destructive examination after 100 hours.
9. Test will last 1000 hours.
10. Test will last 900 hours.

Figure 5-9. Axial Distribution of Tellurium After a 100-Hour Heat Treatment in a Thermal Gradient with (U-25 wt % Pu) $O_{1.97}$ and (U-25 wt % Pu) $O_{2.0}$

Figure 5-10. Axial Distribution of Cesium and Tellurium After a 100-Hour Heat Treatment in a Thermal Gradient with (U-25 wt % Pu) O_{2.00}

Figure 5-11. Axial Distribution of Volatile Fission Products Cs, Mo, Te, I After a 100-Hour Heat Treatment in a Thermal Gradient with (U-25 wt % Pu) O_{2.00}

5.4.2 Cladding Modification Testing

The six capsules of the Series 6 test containing fuel and Cs_2O clad in Types-316, -321, and -347 stainless steel tubing had completed a 100-hour test in 510 to 810° or 510 to 870°C temperature ranges as reported last quarter.⁽⁵⁻⁴⁾ Pulsed eddy current tests were made after the initial run which indicated defects 0.5 to 1 mil deep, then the rods were reinserted for an additional 900-hour exposure. These rods have been re-eddy current tested to nondestructively determine the effect of the additional heat treatment on the extent of intergranular attack at this lower temperature level. The results are being analyzed and the rods will then be sectioned for metallographic examination.

5.4.3 High Magnification Examination of Cladding Attacked by Fission Products

Replicas were prepared from a metallographic sample of pin F3B-3 that had exhibited both general and intergranular cladding attack and that had received electron microprobe analysis as reported last quarter.⁽⁵⁻⁵⁾ The replicas were carbon coated and then either platinum-palladium alloy shadowed for transmission electron microscopy (TEM) or gold-shadowed for scanning electron microscopy (SEM). The high resolution of the TEM indicated in Figure 5-12 that the reaction product contained some small particles of unreacted cladding fragments. The SEM examination more clearly revealed the nature of the intergranular attack in Figure 5-13. The penetrating material apparently has a hardness of the same order of magnitude as the cladding. Also, the penetrating material seems to be outlined with a groove similar to the grain boundary grooves deep within the cladding.

A technique for preparing extraction replicas was successfully demonstrated on a sample of unirradiated cladding. This cladding was the same heat treated tubing that had a thin oxide coating housed E 1H-30, and was the subject of an electron microprobe examination in Sections 5.3.1.1 and 5.3.1.2.

The interior of the tube was filled with a liquid plastic (Technovit) that tightly adhered to the tube surface. The cladding was electrolytically dissolved leaving only the oxide film on the Technovit plug and revealing in (SEM) examination the original metal oxide interface. Several phases (probably oxides and grain boundary carbides) were present on the film and these, along with the oxide layer itself, could be identified by X-ray fluorescence or diffraction.

5.4.4 Fuel Impurity Effects

Investigations of fuel and cladding parameters of irradiated rods that displayed varied degrees of intergranular attack have suggested that there may be some correlation of the carbon or nitrogen content of the fuel with the degree of attack. Six thermal gradient capsules have been fabricated with 500 ppm carbon or nitrogen, with and without Cs_2O . The carbon was added as radioactive carbon-14 in BaCO_3 while nitrogen was added as UN. They have been heat treated for 100 hours in a 550 to 1100°C thermal gradient and then tested with pulsed eddy current for symptoms of attack; no significant indications were detected. The specimens have been sectioned and prepared for metallographic and carbon-14 beta autoradiographic examination.

5.5 SODIUM-FUEL REACTION KINETICS

The transducer-equipped stainless steel vessel for monitoring fuel expansion during exposure to liquid sodium operated for one week at 570°C (1060°F). It was loaded with 2-1/2 inches of $\text{UO}_{2.05}$ fuel housed in Type-316 stainless steel tubing. The tube contained a one-inch-long V-groove notch to allow sodium penetration. After exposure, it was found that the original defect had expanded 5 to 6 mils and that the expansion had cracked the cladding at each end as shown in Figure 5-14. The sodium contained ~10 ppm O_2 . Expansion of the tubing to the above extent, 2% $\Delta D/D$, included the 0.217-inch-diameter pellet expanding through the 1- to 2-mil clearance and beyond. The transducer did not follow the expansion although it was operational and the unit is being redesigned to prevent transducer slippage or hangup. The test will be repeated, but these first results establish the validity of the basic experimental plan.

Figure 5-12. Transmission Electron Micrograph from Replica of Intergranularly Attacked Cladding of Pin F3B-3

A682-18 OPTICAL MICROGRAPH, AS POLISHED 500X

1556 SCANNING ELECTRON MICROGRAPH 300X

1559 SCANNING ELECTRON MICROGRAPH 3000X

1563 SCANNING ELECTRON MICROGRAPH 3000X

Figure 5-13. Scanning Electron Microscope Study of Intergranularly Attacked Region of Fuel Pin F3B-3

2MA-19 BEFORE AFTER ~1.5X

(a) 0.25 in. CAPSULE IN BEFORE AND AFTER STATES

2MA-20 AFTER 3.5X

(b) HIGHER MAGNIFICATION VIEW TO EMPHASIZE THE 6-mil $\Delta D/D$ AND SLIT ELONGATION

Figure 5-14. $UO_{2.05}$ Swelling Upon Exposure to Sodium at $570^{\circ}C$

5.6 MEASUREMENT OF OXYGEN POTENTIAL OF MIXED OXIDE FUEL

Composite electrolyte tubes have been tested over a range of oxygen potentials. These probes are $\text{ZrO}_2\text{-CaO}$ coated with $\text{ThO}_2\text{-Y}_2\text{O}_3$ to extend the range of measurements into low O/M levels (~ -150 kcal/mole). Testing with $\text{Cr/Cr}_2\text{O}_3$ couples, or with $(\text{U-15 wt \% Ce})\text{O}_{2-x}$ that have nearly equivalent oxygen potentials at ≈ -135 kcal/mole, give emf readings that are nearly identical to values measured with pure $\text{ThO}_2\text{-Y}_2\text{O}_3$ probes. The readings from both sets of probes, however, were less than the theoretical values. This may be due to impurity interactions and inquiries have been made to obtain higher purity $\text{ThO}_2\text{-Y}_2\text{O}_3$ probes.

An initial design of a remotely operated fixture that will permit the measurement of oxygen potential in irradiated samples has been critically reviewed by hot cell operating personnel.

REFERENCES

- 5-1. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program*, Forty-First Quarterly Report, November 1971–January 1972, February 1972, (GEAP-10028-41), Section 5.3.1.2.
- 5-2. Ibid, Figure 5.9.
- 5-3. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program*, Thirty-Ninth Quarterly Report, May–July 1971, August 1971, (GEAP-10028-39), Figure 7-5.
- 5-4. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program*, Forty-First Quarterly Report, November 1971–January 1972, February 1972, (GEAP-10028-41), Section 5.4.2.
- 5-5. Ibid, Section 5.3.1.1.

6. TASK F—FAST FLUX TESTS AND SUPPORT

Task Leader: J. D. Stephen*

6.1 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, as well as subassembly variables, are determined under conditions simulating as closely as possible those expected in a fast ceramic reactor.

The test parameters for Task F are selected by a combination of design analyses which rely on basic properties data and on the results of pertinent experiments such as those in Task E and Task L. Model analysis (under Task P) has been applied quantitatively to interpretation of the results, and to aid in the selection of ranges of parameters for further tests. The results of prior fast flux irradiation tests are interpreted 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 design; and Task G-6, which is concerned with subassembly design and performance. Demonstration of high fuel burnup is required over a range of power levels. Fuel parameters that are under study in a fast flux environment include density, configuration, fabrication process history, stoichiometry, diametral gap, and external restraint. Fast flux tests are performed with both encapsulated and unencapsulated rods.

Selected irradiation tests in thermal test reactors have been 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 rod diameters without the complications of cladding swelling.

A summary of the fast flux irradiation tests in the EBR-II is given in Table 1-3. The "nominal" values of fluence, power, and burnup given in Table 1-3 were calculated using the fission rates provided in the latest revision (Number 4, February 1971) of the "Experimenters' Guide for Irradiation Experiments in EBR-II" as a basis.

6.2 HIGH BURNUP IRRADIATIONS

F0 Experiment (Low Power Operation with High Density Fuel)

The F0 experiment contains five encapsulated fuel rods with 93 to 95% theoretical smeared fuel density operating at less than 9 kW/ft.

Rod F0E was destructively examined at 4 at. % burnup. Fuel rods F0A and F0C completed irradiation to 9 at. % (nominal**) during Run 55 and are being discharged for destructive examination. At the request of EBR-II, the rods will remain at EBR-II to be examined in the new HFEF facility. Fuel rods F0B and F0D will be reconstituted for continuation of the irradiation in an outer row after installation of the reflector.

F3B Experiment (Fuel Void Deployment)

The F3B experiment, which contains 18 encapsulated fuel rods focused primarily on the effect of fuel void deployment, had reached 10 at. % burnup at 16 kW/ft peak start-of-life power with 1150°F maximum cladding temperature (nominal**) at the time of the interim examination.

Six rods were destructively examined at the interim. Eight of the remaining twelve F3B rods were combined with eleven F5 fuel rods into a single subassembly (X151) and reinserted in EBR-II position 6A4 on March 10, 1972 for Run 55 startup for continuation of the irradiation to high burnup. The remaining four rods are being held for insertion later.

*Contributors are listed in Appendix B.

**See general discussion end of Subsection 6.1.

F4 Experiment (Medium Density Fuel Performance at High Power)

The F4 experiment, which contains 7 encapsulated fuel rods with 84 to 87% theoretical smeared fuel density, was irradiated to 9 at. % burnup at > 16 kW/ft peak start-of-life power with 1150°F maximum cladding temperature (nominal*).

The fabrication and irradiation parameters for the F4 rods are summarized in Table 6-1. The nondestructive examination of the rods is complete. Two rods, F4A and F4E, are being destructively examined at the LASL facility.

The neutron radiographs, which were made at EBR-II, showed that a central void had formed over almost the full length of the fuel column in each rod. Several small areas containing material with a higher neutron cross section than the fuel were present in the voids. There were no indications of fuel rod failure. The fuel column length increases measured from the neutron radiographs are shown in Table 6-1.

The diametral profile of each of the capsules was measured at radial orientations of 0, 45, 90, 135, and 180 degrees, and with the capsule rotating. The maximum diameter increase for each is given in Table 6-1. Except for F4A and G, the capsules showed a periodic variation superimposed on the overall diameter increase (Figure 6-1). The periodic peaks occur at approximately one-inch intervals. The pitch of the spiral wire spacer was two inches, indicating that the peaks are associated with the wire. As seen in Figure 6-1, most of the periodic variation occurs on the lower temperature, bottom portion of the capsules below the peak flux region.

All of the capsules were scanned axially for gross gamma activity and for the activity of the isotopes Cs-134, Cs-137, Nb-95, Zr-95, and Ru-103. The plots are shown in Figures 6-2 to 6-9. The Cs-134 activity was near background. The Zr-95 activity and the Nb-95 activity in each rod were smoothly distributed with no spikes. The Cs-137 activity peaked at the ends of the fuel column. Spikes of Ru-103 activity are evident on the scans of F4G, H, and J. Most of the spikes occur in the bottom portion of the fuel. Those of G and H correspond to areas of high neutron absorption areas in the center void on the neutron radiographs. There was no correspondence with the high neutron absorption areas in the central void for the spikes on the F4J scan. Both F4H and J had a small spike in the gross gamma scans about 5-1/2 inches above the top of the fuel, which is above the blanket in the region of the spring. Both F4K and L had a large spike at the top of the slightly enriched blanket. Spectral scanning at these spikes indicated that they were due primarily to Cs-134, with a slight contribution from Cs-137.

Radiography (high energy X-rays, 22 MeV) of F4A and F4E at LASL is complete. There were no indications of fuel rod failure. The fuel pellets all contained transverse cracks and occasional longitudinal cracks. The central void in the fuel rod F4A, with Type-304 stainless steel cladding, starts in the bottom fuel pellet and increases to a maximum of 0.060 inch at ~ 3.3 inches from the top of the fuel column, diminishing to essentially zero at ~ 1.5 inches from the top. The central void in fuel rod F4E with Type-316 stainless steel cladding starts in the bottom fuel pellet with a diameter of 0.050 inch, increases smoothly to 0.075 inch near the fuel mid-point, then decreases smoothly to 0.050 inch in the top fuel pellet. The fuel column length of the F4A and F4E fuel rods increased ~ 0.33 and 0.66 inch, respectively. These results are consistent with the measurements from the earlier neutron radiographs.

F12P and F12Q Experiments (Effect of Cladding Thickness in a Thermal Flux Irradiation to High Burnup)

The F12P and F12Q fuel rods with fuel densities of 92% theoretical were irradiated to $\sim 100,000$ MWd/Te burnup in the GETR at cladding temperatures of $\sim 1150^{\circ}\text{F}$. The 2-inch fuel column, encapsulated rods are clad in annealed Type 316 with 15-mil and 10-mil wall thickness, respectively.

Neutron radiography, de-encapsulation, profilometry, and gamma scanning were completed at the GE Vallecitos Nuclear Center. The fuel rods were then shipped to LASL for complete destructive examination. Radiography at LASL is complete.

*See general discussion end of Subsection 6.1.

Table 6-1

SUMMARY OF FABRICATION AND IRRADIATION PARAMETERS FOR
GROUP F4 ENCAPSULATED FUEL RODS

	F4A	F4E	F4G	F4H	F4J	F4K ^(a)	F4L ^(a)
Fuel							
Wt% Pu (91% enriched in Pu-239)	20	30	20	20	30	20	25
U-235 Enrichment, %	92.9	91.4	92.6	92.8	89.3	91.3	92.6
Stoichiometry, O/M	2.00	2.00	1.98	1.98	1.98	2.00	1.98
Fabrication Process	← Coprecipitated →						
Configuration	← Solid Pellet →			← Powder →			
Average Pellet Density, % Theoretical	89.1	89.8	89.1	89.9	90.1	--	--
Smeared Density, % Theoretical	86.4	86.6	86.1	87.1	87.1	84.0	84.2
Diametral Gap Range, mils	2.6-4.9	3.2-5.3	2.5-4.5	2.7-4.2	3.4-4.2	--	--
Cladding							
Alloy	304 _b	316	316	I-800	316	316	I-800
Fabrication Method	← Welded and Drawn →						
Thermal Treatment	← Annealed →						
Mechanical Treatment	← Ballized →						
Wall Thickness Range, Inches	0.0141-0.0146	0.0149-0.0152	0.0152-0.0154	0.0148-0.0149	0.0149-0.0154	0.0150-0.0153	0.0147-0.0152
Nominal* Irradiation Conditions							
Peak Burnup, Atomic %	8.89	9.19	8.94	9.02	9.08	8.87	9.09
Peak Burnup, MWd/kg (192 MeV/fission)	80.4	82.6	80.9	81.7	81.6	80.2	82.0
Peak Linear Power kW/ft (SO ₂ /EOL) ^(b)	17.6/14.4	18.7/14.8	17.8/14.3	18.4/14.6	18.6/14.7	17.4/14.1	18.0/14.4
Peak Fluence, (>0.1 MeV) 10 ²² n/cm ²	5.87	5.97	5.90	5.96	5.97	5.90	5.92
Peak Cladding Temperature at Fuel Interface, °F	1130	1160	1130	1150	1160	1130	1140
Dimensional Changes							
Maximum Capsule Diameter Increase, mils (%)	3.6 (0.96)	3.7 (0.98)	3.7 (0.98)	3.7 (0.98)	3.5 (0.93)	4.0 (1.06)	4.1 (1.09)
Fuel Column Length Increase, Inches (%)	0.31 (2.2)	--	0.39 (2.7)	0.43 (3.0)	0.29 (2.0)	--	0.07 (0.5)

(a) The UO₂ blankets of pins F4K and F4L were 2.34% enriched in U-235.

(b) Start of Life/End of Life

* See general discussion end of Section 6.1

Figure 6-1. Diameter Profiles of Group F4 Capsules Over the Fuel Length

Figure 6-2. Gross and Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4A

Figure 6-3. Gross and Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4E

Figure 6-4. Gross and Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4G

Figure 6-5. Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4G

Figure 6-6. Gross and Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4H

Figure 6-7. Gross and Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4J

Figure 6-8. Gross and Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4K

Figure 6-9. Gross and Isotopic Gamma Activity Profiles Over the Fueled Region of Rod F4L

Profilometry has been performed at $\sim 10,000$, 20,000, 30,000, 40,000, 60,000, 78,000, and now 100,000 MWd/Te. The maximum diameter increases after 60,000 MWd/Te were 4.4 mils with 15-mil cladding and 7.4 mils with 10-mil cladding. The rate of diameter increase $[d(\Delta \bar{D})/d(\text{BU})]$ in mils/MWd/Te was significantly lower for the 15-mil cladding at burnups below 40,000 MWd/Te. Above 40,000 MWd/Te, the rates of increase for the 10- and 15-mil cladding converged and were essentially equal above 60,000 MWd/Te. The maximum diameter increases for F12P and F12Q at 100,000 MWd/Te were 10.2 and 10.7 mils, respectively.

The gross gamma scans of both fuel rods showed no activity in the plenum (bottom) area. The activity of the top insulator pellet was significantly higher than that of the bottom insulator pellet, especially for fuel rod F12P.

LASL radiographs showed no indications of fuel rod failure. The radiograph of F12Q at the 90-degree orientation is shown in Figure 6-10. The fuel pellets in both rods were cracked and all contained a central void. The center void was larger at the top of the rod and had a maximum diameter of 0.050 inch in F12P and 0.070 inch in F12Q. There was no change in the fuel column length of fuel rod F12P, but the F12Q fuel column length increased 0.06 inch.

6.3 ELEVATED CLADDING TEMPERATURE IRRADIATIONS

F3A Experiment (Unencapsulated Fuel Rod Irradiation at 1150°F Cladding Temperature)

The topical report evaluating the results of the destructive examination of the F3A fuel rods, which showed significant attack at the cladding inside surface, has been prepared (GEAP-13735). The 16 unencapsulated rods with annealed Types-304L and -316 stainless steel cladding in flowing primary sodium was irradiated at >16 kW/ft peak start-of-life power with 1150°F maximum cladding temperature to 6 at. % burnup (nominal*).

F5 Experiment (Encapsulated Fuel Rod Irradiation at 1200°F Cladding Temperature)

The F5 experiment contains 19 encapsulated fuel rods clad in Types-304, -316 (annealed and cold worked), -321, -347, and Incoloy 800, and had reached 6 at. % burnup at 16 kW/ft peak start-of-life power with 1200°F maximum cladding temperature (nominal*) at the time of the interim examination.

Three rods were examined at the interim. Fuel-cladding interaction was observed in the peak flux region of F5Q (50% cold worked Type-316 cladding). A detached rim of material was found between the fuel and cladding. Microprobe examination showed that the rim consists of Fe and Ni (Cr was not detected) with local zones of Pu. The detailed microprobe results are reported under Task E. Eleven rods have been reinserted in EBR-II as part of subassembly X151.

F11A Experiment (Wire-Spaced Subassembly Irradiation with 1275°F Cladding Temperature)

The F11A experiment will determine the performance of unencapsulated fuel rods with Types-304, -316, -321, and -347 stainless steel in several structural conditions. The fuel rods will be irradiated in a wire-spaced subassembly at $\sim 1275^\circ\text{F}$ cladding temperatures. Fabrication of three replacement rods and one spare rod with Type-321 and 20% cold worked Type-316 cladding, respectively, is under way.

6.4 OPERATING LIMITS IRRADIATIONS

F9C-1, F9C-2 Experiments (Run to Failure)

Irradiation of F9C-1 and F9C-2 is under way as part of the EBR-II run-to-failure program. Prior to reconstitution into two Mark E-19 subassemblies, the fuel rods had reached 7 at. % burnup at 14 kW/ft peak start-of-life power with 1050°F maximum cladding temperature (nominal*) as part of the F9C experiment. The irradiation is continuing with minimal perturbation of the prior operating conditions.

*See general discussion end of Subsection 6.1.

*Figure 6-10. Radiograph of F12Q at 90° Orientation
(22 MeV X-Rays, LASL Betatron)*

Analysis of the fission gas in the medium power fuel rod (F9C-31) indicates $\sim 74\%$ release based upon a *measured* burnup (peak) of 6 at. %. The nominal* operating conditions for this rod were 13 kW/ft peak start-of-life power with 1025°F cladding temperature, irradiated to 7 at. %.

The examination of F9C-13 at the LASL facility is under way. Preliminary results are being evaluated. This rod, with 87% theoretical fuel density, clad in annealed Type 316, reached 7 at. % burnup at 13 kW/ft peak start-of-life power with 950°F maximum cladding temperature (nominal*). The radiographs showed that the fuel pellets all contained transverse cracks and occasional longitudinal cracks. The first upper blanket pellet had a large longitudinal crack from end to end. The central void starts ~ 1.5 inches above the bottom of the fuel column, increases smoothly to a maximum of ~ 0.025 -inch diameter at the center of the fuel column, then decreases smoothly to zero at ~ 0.75 inch from the top of the fuel column. No significant change in fuel column length was discernible.

During assembly of F9C-1 and F9C-2, the spade tips of fuel rods F9C-1, -23, and -32 were found to be damaged, preventing assembly on the fuel rod T-bar grid. The rods apparently were dropped vertically a short distance as they were loaded into the EBR-II FEF storage magazines during the interim examination. Visual inspection at 4X magnification and neutron radiography indicated no additional damage. The end plugs were repaired by EBR-II.

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

The F20 experiment, containing 19 encapsulated fuel rods to be irradiated at >16 kW/ft for several days, will determine power-to-fuel center melting parameters and characteristics of operation with fuel center melting as a function of fuel density and void deployment for both fresh and preirradiated fuel.

Fabrication of the F20 fuel rods has been resumed. Work had been stopped temporarily to speed fabrication of the higher priority F9F experiment.

6.5 SUBASSEMBLY IRRADIATIONS

F9E Experiment (Unencapsulated Fuel Rods in a Grid-Spaced Subassembly)

The F9E experiment will study the performance of a 37-rod, grid-spaced subassembly operating at 1100°F cladding temperature to >11 at. % burnup (nominal*). The experiment was designed under Task G6 and is being built and irradiated under Task F.

An audit of the F9E experiment fabrication is complete. The audit included inspection at the EBR-II site of the actual F9E components. A joint GE/ANL/RDT meeting was held to discuss the interim and final examination requirements for the GE F9E and F9F grid-spaced subassemblies. ANL will factor GE's requirements into those of other experimenters testing grid-spaced subassemblies in order to standardize disassembly/examination equipment.

F9F Experiment (FFTF Diagnostic Irradiation of a Grid-Spaced Subassembly)

The F9F experiment, containing 37 unencapsulated fuel rods clad in 0.230-inch-diameter Type-316 20% cold worked with 85% theoretical fuel density, will test grid-spacer performance under near FFTF conditions. The experiment was designed under Task G6 and is being built and irradiated under Task F.

Fabrication of the F9F fuel rods is complete. The Fabrication Certification Report for the rods, including radiographs, weld samples, and xenon tag ampoules, has been forwarded to EBR-II for review. The tag gas ampoules have been analyzed by EBR-II, and found to contain between 1.40 to 1.50 cc xenon. This is well above the minimum acceptable level.

Fabrication of the mockup grid spaced inner channel and fuel rods for F9F is complete. This mockup of the Mark S-1 subassembly, designed specifically for grid-spacer experiments will be used for water flow testing at GE to size the F9F orifice and for practice assembly-disassembly at EBR-II.

*See general discussion end of Subsection 6.1.

Nuclear grade Type 316 10% cold worked strip material for the in-reactor F9F grid spacers has been obtained from Cartech, and is currently being rolled by Arnold Engineering of Marengo, Illinois.

6.6 IRRADIATIONS EVALUATIONS AND SUPPORT

RDT Standards

Proposed amendments have been prepared to RDT Standards E-13-1T covering FFTF Ceramic Grade Plutonium Dioxide and E13-4T covering FFTF Plutonium Nitrate Solution. In addition, it has been proposed that new standards be prepared for experimental fuel pellets and experimental insulator pellets. The amended and new standards are to be used in fuel experiments conducted under AEC Programs.

7. TASK G-6—CORE DESIGN ANALYSIS

Task Leader: P. R. Pluta*

7.1 GENERAL

The main purpose of this task is to develop a grid-spaced fuel assembly for the fast breeder reactor. The grid-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.

The master schedule for grid-spaced driver fuel assembly development has been updated and included in Figure 7-1. This schedule is focused on developing grid-spaced FTR driver fuel assemblies which will be available for testing in the first core of the FTR in June 1974.

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.

7.2 F9E DIAGNOSTIC GRID-SPACER EXPERIMENT IN EBR-II

The F9E group irradiation is the first in-pile test of a grid-spaced fuel assembly utilizing the wave-beam grid-spacer concept which was conceived by General Electric. F9E employs the following three different variations of the basic wave beam concept:

- Modified Wave Beam Full Grid
- Modified Wave Beam Staggered Grid
- Combination Modified-Unmodified Wave Beam Full Grid.

Task G-6 participated in an engineering and fabrication audit of F9E. Discussions were held with EBR-II personnel to resolve problems associated with interim and final examination requirements of F9E.

The desired range of F9E operating conditions was redefined to enable the EBR-II Project to place F9E in a core position with an adequate fast neutron flux level. Operating conditions requested previously did not include a flux requirement, since flux requirements were implicit in the requested power level. However, since the neutron energy spectrum will be changed by installation of a stainless steel reflector in EBR-II, the power level originally requested no longer implies the same fast flux level, and a change in the requested operating conditions is necessary. The ability of EBR-II to meet these guidelines and the actual position of F9E in the core are to be determined in May 1972.

A topical report entitled "Design Basis of the F9E Grid-Spaced LMFBR Diagnostic Assembly for Irradiation in EBR-II," GEAP-13765, has been transmitted to the AEC.

*Contributors are listed in Appendix B.

Figure 7-1. Task G-6 Grid-Spaced Fuel Assembly Development—
Focused on FTR Assembly Activities

7.3 F9F DIAGNOSTIC GRID-SPACER EXPERIMENT IN EBR-II

F9F is the second in a series of GE grid-spaced fuel assembly experiments planned for irradiation in EBR-II (F9E was the first). F9F is designed to provide fuel assembly test conditions and data of direct support to the optimized FFTF grid-spaced driver fuel assembly being developed by General Electric.

The F9F assembly is designed primarily to demonstrate:

- The adequacy of the design to withstand mechanical interaction between the bundle and the channel.
- The ability of the complete staggered-grid system to restrain rod bowing
- The adequacy of the tab-notch concept of fixing grids to channels.
- The behavior of grids made of C.W. Type-316 stainless steel in a reactor environment.

Engineering support was provided for the fabrication of a mockup of F9F which was successfully assembled.

Grid-spacer pushout tests were conducted on a model of the F9F channel grids to determine the load normal to the grid spacer which would cause the grid spacer tabs to disengage from the channel recesses. Such loads occur in-pile due to rod-to-channel differential growth as a result of friction between the fuel rods and the grid spacers as the rods slip through the grid spacers. A test fixture was used that allowed 37 hollow cladding tubes to prototypically fill the grid-spacer openings.

The axial pushout load was applied by an Instron Testing Machine using a rubber padded block to distribute the load. Tests were conducted on full grids and staggered grids. The load required for dislodgment was approximately twice that obtained in earlier tests on a model of the first grid-spaced EBR-II assembly F9E. The difference is attributed to the deeper recess and hence greater engagement of the F9F tab as well as the greater strength of the cold worked material. The load per tab for disengagement was approximately 10 pounds.

An X-ray technique was developed for inspection of the engagement of grid-spacer tabs in the channel recesses. This inspection technique will be used to verify the position of grids in F9E, F9F, and all future grid-spaced fuel assemblies.

7.4 FFTF GRID-SPACED DRIVER FUEL ASSEMBLY

A schedule for the FTR driver grid-spaced fuel assembly activities has been prepared. This is aimed at producing a complete mockup assembly and four reactor assemblies. The mockup is to be shipped to FFTF in May 1973 for trial assembly and disassembly. The reactor assembly shipping date agreed to with HEDL is February 1, 1974.

An outline for the System Design Description (SDD) of the FFTF grid-spaced driver fuel assembly has been prepared. The format used complies with the requirements of RDT Standard F1-2T. The basis for the SDD stems from SDD-31 which deals with the FTR wire-wrap driver fuel assembly. A draft of the first section of the System Design Description (SDD), which deals with the system functions and design requirements, has been prepared. A draft of the second section of the SDD document is in preparation. This deals mainly with the detailed design description and is responsive to design requirements set out in the first section. Other items such as system interfaces, examination procedures, etc., are dealt with.

Ordering of material and components for a total of six fuel assemblies (2 mockup and 4 for the reactor) has been initiated.

7.5 SODIUM FLOW TEST

A long-term sodium flow test is being conducted in the General Electric Breeder Test Facility Control Drive Test Loop to evaluate the endurance characteristics of the reference FTR Grid-Spacer Design and to provide Engineering information that will support the design bases for the F9E, F9F, and Reference FTR grid-spaced fuel assembly. The test assembly consists of two separate 37-rod dummy fuel elements positioned in series. The FTR shield/orifice region has been simulated for both assemblies to generate prototypical inlet turbulence in each bundle. One 37-rod assembly utilizes staggered grids, the reference for the GE grid-spaced FTR driver assembly. The other 37-rod assembly utilizes full grids, the backup concept for the FFTF driver fuel assembly.

The prime objectives of this endurance test are to:

- a. Demonstrate the mechanical integrity and hydraulic stability of a model FTR fuel assembly operating under conditions of turbulent flow at 1000°F.
- b. Determine the mechanical integrity of the reference FTR grid support system under typical operating conditions.
- c. Determine the effects of extended periods of flow-induced vibration on the clad-grid interface.
- d. Provide general information in support of the F9E and F9F EBR-II irradiation experiments.

The secondary objectives are to determine:

- a. The effects of various grid/channel clearances on the mechanical integrity of the support system.
- b. The stability of a grid spacer unsupported by the channel.

Details of the test assemblies are shown in Figure 7-2 (816E140). A complete description of the test parameters and test assemblies has been reported previously. ⁽⁷⁻¹⁾

The test is scheduled for approximately 6 months of operation in sodium at 1000°F with a peak coolant velocity of 22.3 fps. The test assembly was installed in the drive test loop on September 23, 1971 and operation began on October 5, 1971. Through the end of the quarter, approximately 4400 hours of operation at rated temperature and flow have been logged, exceeding the original test goals.

Removal, cleaning, and inspection of the assemblies will be initiated in May 1972. Cleaning of the assemblies will be accomplished using procedures similar to the EBR-II moisturized argon method. Facilities for this cleaning are now in preparation. Detailed destructive examination will follow cleaning in accordance with an approved test inspection plan.

7.6 HYDRAULIC FLOW TEST

7.6.1 Hydraulic Flow Test Background

Flow-induced vibration is of concern with any fuel bundle designed for extended periods of operation. Because of the difficulty of predicting the nature of vibration in the staggered grid support arrangement, a two-part test program is in progress to determine the flow-induced vibration response in the reference FTR fuel assembly. These two parts are as follows:

1. Single-rod water flow tests with simulated grids.
2. 217-rod, full-scale water flow tests containing 12 staggered grids and 6 full grids.

[816E140]

Figure 7-2. Sodium Flow Test Bundle

7.6.2 Single-Rod Tests

A series of single-rod hydraulic tests, designed to measure the amplitude of vibration of a fuel rod in a simulated bundle containing staggered grid spacers were completed.

The principal results of the single-rod tests are as follows:

1. The natural frequency of rod vibration with simulated staggered grid spacers on three-inch spacings was measured as 1250 ± 15 Hz. This is in excellent agreement with the value of 1243 Hz calculated for a rod simply supported at 3-inch intervals. The frequency of vibration was found to be identical at all flow rates and equal to that obtained by mechanically shocking the test channel. This is in agreement with theoretical calculations.
2. A staggered-grid-spacer system induces the same fundamental vibrational mode pattern as a full-grid-spacer system with the same spacing. This implies that the staggered-grid-to-rod contacts are true nodal points.
3. The maximum displacement occurs at mid-span location between grids. The displacements for flow rates between 12 and 22.4 ft/sec were 0.135 to 0.7 micro inch zero to peak. These displacements are within 30% of the theoretical predictions based on Reavis.
4. The vibration is unidirectional. This directionality implies that accelerometers and proximity probes must be fixed in the 217-rod bundle test with the resistive axis in the direction of the maximum displacement.
5. The fuel rod grid interaction displays a unique motion. The relative motion of grid-to-rod could cause fretting or wear. Further investigation of this phenomenon as well as comparisons to full grids will take place in the 217-rod test, and careful observation of the fuel rod at grid location will follow the sodium endurance test.
6. The miniature accelerometer has been established as a satisfactory sensor to measure interior fuel rod displacements of the order of greater than $0.05 \mu\text{inch}$ zero to peak, and its use is recommended in the 217-rod bundle flow test.

7.6.3 217-Rod Flow Test

The fabrication of an FTR 217-rod water flow test bundle will be completed in May. Remaining fabrication work includes insertion of the grid spacers into the channel, loading of the six accelerometers into dummy fuel rods (two have been successfully loaded) and insertion of the dummy fuel rods into the channel.

The maximum rod bending stress and grid-spacer contact load within the staggered grid region of the FTR bundle have been estimated from the maximum rod deflection observed from the single-rod flow test results. The maximum stress and load in the previous single-rod test have also been estimated based on the maximum deflection seen in that test. The results are listed below. For both rod deflections, the bending stresses and the grid spacer loads are negligible.

	Rod Displacement	Rod Bending Stress	Grid Spacer Load
Single Rod Flow Test	0.75×10^{-6} inch*	2.5 psi	0.0014 lb
217-Rod FTR Fuel Assembly	2.0×10^{-5} inch**	68 psi	0.037 lb

One of the primary objectives of the instrumented 217-rod hydraulic test will be to verify the vibration predictions for the reference FTR fuel assembly.

*Maximum single-rod flow test rod deflection (at 22.65 inches from the channel inlet).

**Maximum predicted rod deflection for staggered grid region of FTR 217-rod bundle (at 10.65 inches from the channel inlet).

7.7 GRID SPACER MECHANICAL TEST PROGRAM

A mechanical test program for 217-rod grid-spaced assemblies has been outlined and a tentative schedule established. A major test series in this program is the rod-grid sliding friction tests in 1000°F sodium.

A test plan has been drafted to conduct this series of tests to determine the effect of relative movement of rods and grids, as influenced by various mechanical factors, on the integrity of the grid spacers and rod cladding.

Two channels, each 30 inches in length, are planned for the tests. One channel will be fabricated to house eight staggered grids on a 3-inch pitch, the other to house four full grids on a 6-inch pitch. Two sets of rods will be fabricated—one set having a 0.230-inch outside diameter which is the nominal FTR dimension, and a second having a 0.2325-inch outside diameter which is representative of a worst rod-to-grid interference at end of life due to tolerances and differential swelling. All four combinations will be tested.

The test assemblies will be in a pool of 1000°F sodium during the tests. An oxygen level of no more than 5 ppm will be maintained.

Each test will consist of inserting the test assembly into the sodium pool, clamping the channel rigidly at fixed supports and connecting the rods to a holder. The holder is linked via a bellows and a tension/compression sensor to a drive mechanism. The mechanism, consisting of a motor, speed reduction box, and a cam, will oscillate the rods axially. The force required to move the rods will be continuously monitored by a recorder. For those tests with full bundles, the rods will be oscillated 100 cycles with a period of 2 minutes and strokes of 0.065 inch.

The parameters that will be tested include:

- Rod/grid clearance
- Types of grid
- Rod straightness
- Types of rod (interior, edge, or corner)
- Sodium temperature

A tentative list of parameters and variables for each test is shown in Table 7-1.

Only tests 1 through 9 are planned for the first phase of testing. Selected tests beyond test 9 will be conducted as necessary pending the outcome of the first phase of testing.

REFERENCES

- 7-1. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program*, Fortieth Quarterly Report, August–October 1971, November 1971, (GEAP-10028-40).

Table 7-1
TEST PARAMETERS AND OTHER SPECIFICS OF
GRID SPACER MECHANICAL TESTS IN SODIUM

Test Number	Effect Tested	Number of Rods	Nominal Rod o.d.	Nominal Clearance	Rod Type	Grid Type	Rod Bow	Number of Cycles	Sodium Temperature	Remarks
1	Clearance	217	0.2325	-0.0025	All	Full	Nominal	100	100F	
2	Clearance	217	0.230	0	All	Stagg	Nominal	100	1000F	
3	Clearance	217	0.2325	-0.0025	All	Stagg	Nominal	100	1000F	
4	Rod Type	6	0.2325	-0.0025	Corner	Stagg	Nominal	100	1000F	
5	Rod Type	6	0.2325	-0.0025	Edge	Stagg	Nominal	100	1000F	
6	Rod Type	6	0.2325	-0.0025	Interior	Stagg	Nominal	100	1000F	
7	Rod Bow	6	0.2325	-0.0025	Interior	Stagg	0.010 in./ft	100	1000F	
8	Rod Bow	6	0.2325	-0.0025	Interior	Stagg	0.015 in./ft	100	1000F	
9	Channel Ballooning	217	0.2325	-0.0025	All	Stagg	Nominal	100	1000F	Top grid replaced
10	Rod Type	6	0.2325	-0.0025	Corner	Full	Nominal	100	1000F	
11	Rod Type	6	0.2325	-0.0025	Edge	Full	Nominal	100	1000F	
12	Rod Type	6	0.2325	-0.0025	Interior	Full	Nominal	100	1000F	
13	Rod Bow	6	0.2325	-0.0025	Interior	Full	Nominal	100	1000F	
14	Rod Bow	6	0.2325	-0.0025	Interior	Full	Nominal	100	1000F	
15	Temperature	217	0.230	0	All	Stagg ^(a)	Nominal	100	1000F	All new grids
16	Temperature	217	0.230	0	All	Stagg ^(a)	Nominal	100	1100F	All new grids
17	Temperature	217	0.230	0	All	Stagg ^(a)	Nominal	100	1200F	All new grids

^(a) New set of grids inserted in channel which was refabricated from that channel which housed full grids.

8. TASK G-10—SAFETY METHODS AND ANALYSES

Task Leader: J. H. Terhune*

8.1 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.

8.2 MODEL DEVELOPMENT

8.2.1 Fuel Failure Modeling

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

Until recently, the TOFF (Transient Overpower Fuel Failure) program has computed cladding failure thresholds neglecting axial fuel motion within the fuel rod. At high fuel burnup, the molten fuel fission gas pressure could be sufficient to overcome the upper axial blanket weight, the weight of solid fuel above the molten fuel region, the fission gas pressure in the gas plenum, and the frictional force due to fuel and cladding contact. Thus, an equation of motion has been added to the program to account for the molten fuel motion.

Prior to any fuel motion there are two fuel melting situations to consider: (1) whether the molten fuel region does not extend the full length of the molten fuel column; (2) whether the molten fuel does extend the whole length of the fuel column. In the first case the area over which the molten fuel fission gas pressure is acting is obtained from the uppermost central void radius whereas in the second case the area is obtained from the uppermost molten fuel radius. Figure 8-1 illustrates the first case.

For axial fuel motion to take place, the following inequality must hold:

$$P_2 > (W_1 + W_2 + F + P_1 A) / A_1 \quad (8-1)$$

where:

- A = cross-sectional area of the upper axial blanket
- A₁ = cross-sectional area of the uppermost part of the central void or the molten fuel area
- F = frictional force between solid fuel or blanket and cladding
- P₁ = gas plenum pressure
- P₂ = molten fuel fission gas pressure
- W₁ = blanket weight
- W₂ = solid fuel weight

*Contributors are listed in Appendix B.

Figure 8-1. Fuel Rod Prior to Axial Fuel Motion

At the onset of motion the area over which the molten fuel fission gas pressure acts is presumed to change instantaneously to correspond to the cross-sectional area of the blanket or the solid fuel. The equation of motion is then:

$$\frac{(W_1 + W_2)}{g} \frac{d^2 x}{dt^2} = P_2 A - P_1 A - F - (W_1 + W_2) \quad (8-2)$$

where P_1 and P_2 are functions of time, t , or displacement, x , and g is the acceleration due to gravity.

Since the molten fuel volume and fission gas molten fuel pressure change with time, P_2 is also a function of the heat transfer process. But to integrate the equation of motion, the approximation is made that P_2 is a function of displacement only during a heat transfer time step. If δt_0 is the heat transfer time step, then the integration range is $0 \leq t \leq \delta t_0$. In this time interval, the equation of motion can be written as:

$$\frac{d^2 x}{dt^2} = \frac{P_2^0 V_2^0 g A}{(W_1 + W_2) (V_2^0 + A x)} - \frac{P_1^0 A L g}{(W_1 + W_2) (L - x)} - g \left(1 + \frac{F}{W_1 + W_2} \right) \quad (8-3)$$

where

- L = the gas plenum length
- V_2^0 = the fission gas volume in the central void and molten fuel at the beginning of the heat transfer time step
- P_1^0 = the plenum pressure at the beginning of a heat transfer time step
- P_2^0 = the fission gas molten fuel pressure at the beginning of a heat transfer time step

Expansion of the fission gas in the molten fuel and compression of the fission gas in the plenum are assumed to be isothermal, and the friction force is assumed constant.

A first integral of Equation 8-3 may be obtained in closed form, and is:

$$\frac{dx}{dt} = \left\{ \frac{P_2^0 V_2^0 g^2}{(W_1 + W_2)} \ln \left(\frac{V_2^0 + Ax_0}{V_2^0 + Ax} \right) + \frac{2P_1^0 ALg}{(W_1 + W_2)} \ln \left(\frac{L - x_0}{L - x} \right) - 2g \left(1 + \frac{F}{W_1 + W_2} \right) (x_0 - x) \right\}^{1/2} \quad (8-4)$$

where x_0 is the value of x at final equilibrium, and is the positive root of:

$$x^2 - \left(\frac{P_1^0 AL}{W_1 + W_2 + F} + \frac{P_2^0 V_2^0}{W_1 + W_2 + F} + L - \frac{V_2^0}{A} \right) x + \frac{P_2^0 V_2^0 L}{W_1 + W_2 + F} - \frac{P_1^0 V_2^0 L}{W_1 + W_2 + F} - \frac{V_2^0 L}{A} = 0. \quad (8-5)$$

This equation is obtained from the equilibrium force balance.

Integration of the displacement velocity equation is carried out numerically using a third order Runge-Kutta approximation. For reactor fuel elements, the fission gas volume, V_2^0 , is usually small and the plenum fission gas volume is large. This results in x_0 being much smaller than the plenum length. Consequently, the equilibrium displacement is usually reached within a time interval short compared to that for heat transfer to occur. This is consistent with the assumption that $t \leq \delta t_0$ used above to justify the isothermal approximation.

8.2.2 Loop Hydraulics and Channel Voiding

The purpose of this subtask is to develop computer models to describe the transient flow throughout the primary coolant system and to describe the boiling process and void distribution within the reactor core under hypothetical LMFBR accident conditions.

The equations describing isothermal liquid sodium hammer have been reviewed in preparation for inclusion in VOIDUM. The one-dimensional equations for metal pipes given by Streeter and Wylie are: (8-1)

Conservation of Mass:

$$a^2 \frac{g_c}{g} \frac{\partial V}{\partial X} + V \left(\frac{\partial H}{\partial X} + \sin \alpha \right) + \frac{\partial H}{\partial t} = 0 \quad (8-6)$$

Conservation of Momentum:

$$\frac{g}{g_c} \frac{\partial H}{\partial X} + V \frac{\partial V}{\partial X} + \frac{\partial V}{\partial t} + \frac{fV|V|}{2D} = 0 \quad (8-7)$$

where

$$a^2 = \frac{K/\rho}{1 + \frac{CKD}{eE}}$$

$$K = \text{bulk modulus of elasticity of the fluid} = \rho \left(\frac{\partial p}{\partial \rho} \right)_T$$

ρ = mass density of fluid

P = pressure

C = $1.25 - \mu$ for a pipe which is free to expand axially

C = $1 - \mu^2$ for a pipe which is axially restrained

C = 1 for a pipe with expansion joints throughout

μ = Poisson's ratio for the pipe material

D = pipe diameter

e = pipe wall thickness

E = Young's modulus for the pipe material

V = fluid velocity

X = axial distance along pipe

H = fluid head at elevation Z above datum plane = $\frac{g_c}{g} \frac{P}{\rho} + Z$

α = angle between horizontal and axis of pipe

t = time

f = Darcy friction factor = $\frac{8g_c \tau_w}{\rho V^2}$

τ_w = shear stress at pipe wall.

Application of the method of characteristics to equations 8-6 and 8-7 yields four total differential equations. These are:

$$\frac{g}{ag_c} \frac{dH}{dt} + \frac{dV}{dt} + \frac{g}{ag_c} V \sin \alpha + \frac{fV|V|}{2D} = 0 \quad (8-8)$$

associated with the characteristic curve

$$\frac{dx}{dt} = V + a \quad (8-9)$$

and

$$-\frac{g}{ag_c} \frac{dH}{dt} + \frac{dV}{dt} - \frac{g}{ag_c} V \sin \alpha + \frac{fV|V|}{2D} = 0 \quad (8-10)$$

associated with the characteristic curve

$$\frac{dx}{dt} = V - a \quad (8-11)$$

Referring to Figure 8-2, with superscripts representing time steps and subscripts representing physical grid locations on that figure, equations 8-8 and 8-10 in finite difference form are:

$$V_n^j = 0.5 \left[V_r^{j-1} + V_s^{j-1} + \frac{g}{ag_c} (H_r^{j-1} - H_s^{j-1}) - \frac{g\Delta t}{ag_c} \sin \alpha (V_r^{j-1} - V_s^{j-1}) - \frac{f\Delta t}{2D} (V_r^{j-1} |V_r^{j-1}| + V_s^{j-1} |V_s^{j-1}|) \right] \quad (8-12)$$

$$H_n^j = 0.5 \left[H_r^{j-1} + H_s^{j-1} + \frac{ag_c}{g} (V_r^{j-1} - V_s^{j-1}) - \Delta t \sin \alpha (V_r^{j-1} + V_s^{j-1}) - \frac{ag_c}{2g} \frac{f\Delta t}{D} (V_r^{j-1} |V_r^{j-1}| - V_s^{j-1} |V_s^{j-1}|) \right] \quad (8-13)$$

In these equations, V_r^{j-1} , V_s^{j-1} , H_r^{j-1} , and H_s^{j-1} are linearly interpolated values on the physical grid, i.e.,

$$V_r = \frac{V_n^{j-1} - (a) \frac{\Delta t}{\Delta x} (V_n^{j-1} - V_{n-1}^{j-1})}{1 + \frac{\Delta t}{\Delta x} (V_n^{j-1} - V_{n-1}^{j-1})} \quad (8-14)$$

$$V_s^{j-1} = \frac{V_n^{j-1} - (a) \frac{\Delta t}{\Delta x} (V_n^{j-1} - V_{n+1}^{j-1})}{1 - \frac{\Delta t}{\Delta x} (V_n^{j-1} - V_{n+1}^{j-1})} \quad (8-15)$$

$$H_r^{j-1} = H_n^{j-1} - \frac{\Delta t}{\Delta x} (V_r^{j-1} + a) (H_n^{j-1} - H_{n-1}^{j-1}) \quad (8-16)$$

$$H_s^{j-1} = H_n^{j-1} + \frac{\Delta t}{\Delta x} (V_s^{j-1} - a) (H_n^{j-1} - H_{n+1}^{j-1}) \quad (8-17)$$

Given a steady-state flow situation subjected to a disturbance (e.g., a flow blockage or valve partial closure), equations 8-12 through 8-17 can be stepped through in time to compute pressures and velocities throughout the system. To assure convergence of the solution it is necessary that:

$$\Delta t \leq \frac{\Delta x}{(V + a)} \quad (8-18)$$

Since Δt is common to all pipe runs in the system, this requirement provides the criteria for selecting Δx in each pipe run.

Figure 8-2. Time-Space Grid Illustrating Nomenclature for Finite Difference Equations

Boundary conditions can be introduced at each end of each pipe run by solving the equation specifying the boundary condition simultaneously with the equations for the end increments of the pipe runs. For instance, consider a valve with pipe run "a" upstream and pipe run "b" downstream. For the last (downstream) section ($n = n_{\max}$) of pipe "a", equation 8-8 can be written in finite difference form as:

$$V_{a,n_{\max}}^j = -\frac{g}{g_c a_a} \left(H_{a,n_{\max}}^j - H_{a,r}^{j-1} \right) + V_{a,r}^{j-1} - \frac{g \Delta t}{a_a g_c} V_{a,r}^{j-1} \sin \alpha_a - \frac{\Delta t f_a V_{a,r}^{j-1} |V_{a,r}^{j-1}|}{2D_a} \quad (8-19)$$

where the first subscript refers to the pipe run and the second to the section within the run. For the upstream section ($n = 1$) of pipe run "b", equation 8-10 can be written in finite difference form as:

$$V_{b,1}^j = \frac{g}{a_b g_c} \left(H_{b,1}^j - H_{b,s}^{j-1} \right) + V_{b,s}^{j-1} + \frac{g \Delta t}{a_b g_c} V_{b,s}^{j-1} \sin \alpha_b - \frac{\Delta t f_b V_{b,s}^{j-1} |V_{b,s}^{j-1}|}{2D_b} = 0 \quad (8-20)$$

Steady-state flow through a valve may be expressed as:

$$V_{a,n_{\max}}^o = \left(\frac{D_b}{D_a} \right)^2 V_{b,1}^o = C_d^o \sqrt{H_{a,n_{\max}}^o - H_{b,1}^o} \quad (8-21)$$

where C_d^o is the flow coefficient at steady state.

During a transient, C_d is a function of time, so at time-step j :

$$V_{a,n_{\max}}^j = C_d^j \sqrt{H_{a,n_{\max}}^j - H_{b,1}^j} \quad (8-22)$$

Dividing equation 8-22 by 8-21 and defining the dimensionless valve coefficient as $\tau^j = \frac{C_d^j}{C_d^o}$ yields:

$$\frac{V_{a,n_{\max}}^j}{V_{a,n_{\max}}^o} = \tau^j \sqrt{\frac{H_{a,n_{\max}}^j - H_{b,1}^j}{H_{a,n_{\max}}^o - H_{b,1}^o}} \quad (8-23)$$

Equation 8-23 may be solved simultaneously with 8-19 and 8-20 to obtain:

$$V_{a,n_{\max}}^j = \frac{C_1^2 C_2 (\tau^j)^2}{2} + C_1 \tau^j \sqrt{\left(\frac{C_1 C_2 \tau^j}{2}\right)^2 + C_3^{j-1}} \quad (8-24)$$

where

$$C_1 = \frac{V_{a,n_{\max}}^o}{\sqrt{H_{a,n_{\max}}^o - H_{b,1}^o}} \quad (8-25)$$

$$C_2 = -g_c/g \left[a_a + a_b \left(\frac{D_a}{D_b} \right)^2 \right] \quad (8-26)$$

and

$$C_3^{j-1} = H_{a,r}^{j-1} - H_{b,s}^{j-1} + \frac{g_c}{g} \left(a_a V_{a,r}^{j-1} + a_b V_{b,s}^{j-1} \right) - \Delta t \left(V_{a,r}^{j-1} \sin \alpha_a + V_{b,s}^{j-1} \sin \alpha_b \right) - \frac{\Delta t g_c}{2g} \left(\frac{f_a a_a}{D_a} V_{a,r}^{j-1} |V_{a,r}^{j-1}| + \frac{f_b a_b}{D_b} V_{b,s}^{j-1} |V_{b,s}^{j-1}| \right) \quad (8-27)$$

Equation 8-24 expresses the transient velocity at the downstream end of pipe run "a" as a function of conditions known at the previous time step and at steady state. The valve coefficient $\tau = \tau(t)$ is specified as an input variable.

Other boundary conditions within the system can be handled similarly (e.g., nozzles, pumps, pipe breaks, response to HCDA transients, variable coolant level in outlet nozzle, etc.). Future extensions of this formulation will allow treatment of the demonstration plant reference design primary loop.

8.2.3 FREAMD Update

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

A VENUS Code⁽⁸⁻³⁾ which includes revision by ANL through December 1971 has been successfully modified for the GE-635. The ease of input in this version of VENUS, together with several options (such as blanket and reflector region identification, improved timestep control and treatment of void space) not available in the VENUS Code currently interfaced with FREADM, makes the code more advantageous to use when initiating a steady-state or transient start. This version had been applied by ANL earlier to the case of a large LMFBR with blanket regions. For these reasons, and because recent efforts have made the code cleaner from a programming point of view, this latest version of VENUS will be included in a planned FREADM-2 update.

Capability for performing heat transfer in sodium boiling accidents with evaporation and condensation is being added to the FREADM code. To effect this addition, the BUBBLE routine has been separated into three modules to be coupled directly to the FREADM heat transfer routines already in the code.

The first module computes heat transfer coefficients, including velocity-dependent film coefficients, and coefficients for heat transfer between sodium film and bubble where voiding has been initiated. The second module performs a heat balance on the liquid film and computes the thickness at the end of the heat transfer timestep. The third module performs a mass balance and a heat balance on the bubble and computes the pressure in the bubble in addition to the interface conditions between the bubble and the coolant slugs. Radial heat conduction calculations which were previously done in the BUBBLE routine are now done in FREADM. The BUBBLE capability is now available in FREADM either with or without coupling to the primary loop hydraulics (VOIDUM) calculation.

Programming of the TOFF code as a FREADM module was completed. To effect the coupling, a fuel melt fraction calculation was added to the FREADM heat transfer module and the original TOFF input was modified for compatibility with the FREADM DTA input routine. Also, the units of the TOFF input were changed to correspond to FREADM requirements, but the output was kept in the original engineering units. Testing of the coupled program is now in progress.

The TOFF subroutine will be "called", as a user option, in place of the present fuel cladding failure triggers. TOFF uses the single fuel region (steady-state) heat transfer model in FREADM to restructure the fuel within prescribed temperature limits. Fuel rod transient central pressures are computed using fission gas retention based on the restructured fuel volumes and transient temperatures are computed using the FREADM single fuel region heat transfer model.

Coupled FREADM-VENUS calculations have been executed to completion both from continuation of a FREADM calculation and from restart of a FREADM case. However, anomalies have been observed in the fuel temperature calculation, and the source of the anomalies is under investigation.

8.2.4 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.

The exploratory analysis using the PISCES IDL computer code has been concluded. The effort has been stopped at an expedient point consistent with overall program objectives. The experience with the code showed that it could be applied to analyzing a number of safety problems associated with LMFBR power plants. The documentation and user-oriented program design make the PISCES system of codes valuable tools for analysis. The most useful area of application is core disassembly analysis and analysis of high-pressure pipe breaks in the sodium steam generator. Special subroutines, easily inserted in the code, could provide very useful analytical tools for the problems associated with rapid high pressure loading accompanying core disassembly and steam generator high pressure pipe breaks involving sodium-water reactions.

Two PISCES-IDL computer runs have been completed out to problem run times of 50 msec. These runs model the system described in the 41st Quarterly, which is basically a column of sodium 721 cm long, heated uniformly over a 76-cm section representative of the core region. The pressure in the heated zone and the kinetic energy of the sodium slug above the heated zone are plotted as a function of time in Figures 8-3 and 8-4. The only difference between the

Figure 8-3. Pressure and Kinetic Energy Versus Time During Sodium-Fuel Interaction Using Pisces - 1DL Code

Figure 8-4. Pressure and Kinetic Energy Versus Time During Sodium-Fuel Interaction Using Pisces - 1DL Code

two problems is the boundary condition at one end of the column. The results in Figure 8-3 are from a problem having an unyielding boundary at the bottom of the column and a free surface at the top of the column. The results in Figure 8-4 are from a problem having a bottom boundary which yields at 2000 psi (~170 bars) and a free surface at the top. In both problems, energy is added to the 76-cm section at a rate given by:

$$\frac{dQ}{dt} = 223e^{-0.1024t} \frac{\text{cal}}{\text{gm msec}} \quad (8-28)$$

Equation 8-28 is an approximation to the energy addition rate determined for 200 micron molten fuel particles using an ANL MFCI mode. (8-2)

The results in Figure 8-3 are more complete results (further in time) for the problem for which preliminary information was reported in the 41st Quarterly, where problem output to 10 msec was all that was available. The results in Figure 8-3 indicate a different value (and unit) for the slug kinetic energy than in the 41st Quarterly. The previously reported results incorporated an error in units which lead to the conclusion that the slug kinetic energy was a factor of 100 which was greater than would have been expected from the ANL MFCI models. In fact, the slug kinetic energy in Figure 8-3 is quite comparable to the results from the ANL MFCI code even though the pressure is considerably higher. These results prompted the running of the second problem which has boundary conditions identical to the ANL MFCI problem from which the energy addition equation 8-28 was derived. The change in boundary condition reduced the peak pressure to some extent and substantially reduced the slug kinetic energy. The problem output, however, is still quite different than the ANL MFCI model results. This difference is presently unexplained but could be the result of equation-of-state difference in the two models or simply the difference in analysis methods.

The substantial sodium expansion noted in previous results was obtained in the two problems presented here. Vapor formation occurred at about 9 msec in Figure 8-3 and 7 msec in Figure 8-4. At these times, the specific volume of the sodium had changed by about a factor of 2 to 4 which means that considerable motion had occurred. This result is consistent with the model but may not relate directly to experiments such as the TREAT S-series tests, because the model does not include any incoherent turbulent mixing phenomena which may be present in the experiments.

The pressure-time histories in Figures 8-3 and 8-4 have rather unusual behavior because of the instantaneous nature of the initial energy addition. The initial pressure pulse is thought to be the result of this physically unrealistic step change in energy addition rate. The dip in pressure is the result of the rarefaction wave returning from the free surface and interacting with the initial pulse. The second pressure peak is the result of continuing energy addition in the heated volume.

The PISCES MFCI model investigated here has a sodium equation-of-state which is thermodynamically self-consistent. The problems analyzed assumed energy deposition rates obtained from ANL MFCI model, but this assumption could be avoided by using the heat transfer option in the PISCES IDL code. More elaborate models could be constructed which include both fuel (a fuel equation-of-state for the code has been developed) and sodium with heat transfer. Of course, these models would still have parameters with uncertain values, as do current models, but these uncertainties could be investigated more effectively with such an improved model.

REFERENCES

- 8-1. Streeter, V. and Wylie, E., *Hydraulic Transients*, McGraw-Hill, New York, 1967, p. 17.
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- 8-3. Sha, W. T. and Hughes, T. H., *VENUS: A Two-Dimensional Coupled Neutronics-Hydrodynamics Computer Program for Fast Reactor Excursions*, October 1971, (ANL-7701).

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

Task Leader: P. M. Murphy (Acting)*

9.1 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.

9.2 LIAISON

The liaison activities carried out by the GE resident engineer at ANL have been further expanded to cover the progress of a wider range of ANL LMFBR development programs. Communications established under this task between ANL and GE-BRD staff members have resulted in a better awareness of recent R&D activities and future plans of mutual interest.

Periodic reporting with comments on highlights of recent ANL development work related to GE's PA-10 program was continued and requests for information on technical matters were responded to routinely. Where appropriate, the liaison engineer has identified programmatic points of common interest to both organizations and has arranged for discussions between the responsible engineers.

Two computer codes, the NEWFCI and NA-HAMMER, were received from Reactor Analysis and Safety Division in a preliminary form and more than 50 documents were exchanged. A request was also issued for the REXCO-H Containment Analysis Code which should be supplemented by additional REXCO versions as they become available for release.

The subjects discussed during this quarter's communications to GE-BRD include activities carried out at the following ANL organizations: RAS, CEN, MSD, EBR-II, and TREAT.

9.3 PARTICIPATION IN ANL ACTIVITIES

Approximately one-half of the resident engineer's time has been spent in engineering work in support of the R-series flow transient TREAT tests.

Contributions included: supporting calculations for test vehicle design; modeling of loop hydrodynamic response to molten fuel-coolant interactions; modeling of the test vehicle tank blowdown system for support of design of gas relief system for flow coastdown control; and thermal-hydraulic calculations for best simulation of test fuel section thermal response to FFTF flow coastdown.

*Contributors are listed in Appendix B.

10. TASK L—CLADDING DEVELOPMENT

Task Leader: H. J. Busboom*

10.1 GENERAL

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

The objectives of the cladding materials development program are to:

1. 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.
2. Investigate the rate of alloy swelling and the environmental and metallurgical factors which influence and control it in LMFBR core materials and alloys which are potential future candidates for fuel rod cladding and core structural materials.

10.2 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 1-5.

10.2.1 Series L-15 and MT-4—Materials Capsules

The four L-15 capsules and the MT-4 capsule (Mark-A19 type) are continuing irradiation in EBR-II, Row 6, S/A X148. The total fluences accumulated to date are 3.5×10^{22} n/cm²-GE and 9.1×10^{22} n/cm²-GE, respectively.

10.2.2 Series L-16 Materials Capsules

A second group of materials capsules, designated the L-16 series, has been designed to provide quantitative data on the swelling behavior of candidate cladding alloys including Types 304, 316, and 321 as a function of temperature of irradiation, metallurgical condition, and fluence. The L-16 program plan has approval to utilize at least four consecutive EBR-II subassembly irradiations with an interim examination between each subassembly irradiation. Thus, new capsules will be incorporated into the next S/A to provide a means of maintaining up-to-date experiments on cladding alloy swelling. The initial subassembly in the L-16 program is X098. All future S/A's will be designated X098A, X098B, etc.

10.2.2.1 S/A X098 Capsules

These initial 19 (L-16B Type) capsules contain Type-304H and Type-316H tubular specimens as described earlier.⁽¹⁰⁻¹⁾ The irradiation exposure for this S/A was completed after Run 55 at 12,041 MWD's which resulted in an accumulated total fluence of 3.8×10^{22} n/cm²-GE. The S/A was discharged, dismantled, and four Type-304H and four Type-316H capsules will be included with eleven fresh cold capsules for reconstitution into S/A X098A. The detailed interim examination of the hot capsules to be included in X098A has been cancelled since relatively little swelling is anticipated at this fluence level and, in any case, this can be assessed from examination of discharged duplicate capsules. The remaining eleven hot capsules from S/A X098 will be destructively examined later to provide swelling data on Type-304H and -316H tubing at the 3.8×10^{22} n/cm²-GE total fluence level.

*Contributors are listed in Appendix B.

10.2.2.2 S/A X098A Capsules

The eleven new capsules to be incorporated into S/A X098A have been selected to provide a broader base for the L-16 program. A prime stabilized grade stainless steel Type-321H has been added to the existing two alloys Type-304H and -316H. This alloy will be added as double vacuum-melted material having been processed to optimize the creep properties for Type 321. The alloy will be irradiated in annealed, 10% and 20% cold-worked conditions at four temperatures of irradiation in eight L-16B-Type capsules. Air-melted Type 321H will also be included in these L-16B capsules as a minor variable. Three L-16D capsules will be included in X098A: one a cold capsule containing sodium-exposed ring samples and two hot capsules containing highly irradiated ring samples from previously irradiated fuel pin cladding. These materials were described previously.⁽¹⁰⁻²⁾

Type-321H (L-16B) Capsules

Fabrication of twelve L-16B capsules containing Type-321H vacuum-melted tubing has been initiated. The design of these capsules is identical to the earlier L-16B capsules.⁽¹⁰⁻¹⁾ Eight of these capsules are scheduled for irradiation and the additional four will be backup spares for the S/A X098A irradiation. The capsule identification and operating characteristics are given in Table 10-1 and the specimen material descriptions in Table 10-2.

Table 10-1
CAPSULE IDENTIFICATION AND OPERATION CHARACTERISTICS
FOR L16B CAPSULES

Capsule Identity	Drawing Number	Operating Temperature		Rod Location ⁽¹⁾ in Subassembly	Type ⁽⁴⁾ of Flux Wire
		(°C)	(°F)		
LP2AA ⁽²⁾	286E240-G4	450	842	16(0.96)	Ti
LP2BA ⁽²⁾	286E240-G3	525	977	12(0.97)	Fe
LP2CA ⁽²⁾	286E240-G2	600	1112	6(1.01)	Cu
LP2DA ⁽²⁾	286E240-G1	675	1250	2(1.03)	Ni
LP2EA ⁽³⁾	286E240-G2	600	1112	10(1.00)	Al-Co
LP2FA ⁽⁴⁾	286E240-G2	600	1112	5(1.02)	Fe
LP2GA	286E240-G4	450	842	Spare	Ti
LP2HA ⁽³⁾	286E240-G3	525	977	15(0.97)	Ni
LP2IA	286E240-G2	600	1112	Spare	Al-Co
LP2JA ⁽³⁾	286E240-G1	675	1250	9(1.01)	Cu
LP2KA	286E240-G2	600	1112	Spare	Fe
LP2LA	286E240-G2	600	1112	Spare	Cu

- (1) Based on core position 4E2, number in parentheses is rod fission rate normalized to S/A average.
- (2) Capsules AA, BA, CA, and DA are four capsules designed for nondestructive examination. These capsules will continue irradiation to 3×10^{23} n/cm² or until saturation of swelling is established.
- (3) Capsules EA, HA, and JA are to be destructively examined and replaced by new capsules in later subassembly loading.
- (4) Capsule FA will be irradiated until swelling rates for air melted Type 321H is established $\sim 6 \times 10^{22}$ n/cm² and then destructively examined.

Table 10-2

SPECIMEN IDENTIFICATION FOR THE L16B CAPSULES (TYPE 321H)

Capsule Identity	Specimen Identification ⁽¹⁾ for Various Regions ⁽²⁾ in Capsules							
	A	B ⁽³⁾	C	D ₁	D ₂	D ₃	D ₄	E
LP2AA	1AAK	SiC	1A1C	1AAA	11BA	12CA	1ADA	1A2C
LP2BA	1ADK	SiC	1A1D	1AAB	11BB	12CB	1ADB	1A2D
LP2CA	11BK	SiC	1A1E	1AAC	11BC	12CC	1ADC	1A2E
LP2DA	12CK	SiC	1A1F	1AAD	11BD	12CD	1ADD	1A2F
LP2EA	1AAM	SiC	1A1G	1AAE	11BE	12CE	1ADE	1A2G
LP2FA	1A2M	SiC	124C	1A1A	113A	124A	1A2A	113C
LP2GA	1AAL	SiC	1A1H	1AAF	11BF	12CF	1ADF	1A2H
LP2HA	1ADL	SiC	1A1I	1AAG	11BG	12CG	1ADG	1A2I
LP2IA	11BL	SiC	1A1J	1AAH	11BH	12CH	1ADH	1A2J
LP2JA	12CL	SiC	1A1K	1AAI	11BI	12CI	1ADI	1A2K
LP2KA	1AAN	SiC	1A1L	1AAJ	11BJ	12CJ	1ADJ	1A2L
LP2LA	1A2N	SiC	124D	1A1B	113B	124B	1A2B	113D

(1). Specimens are identified as follows:

First character designates alloy

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

Second character designates tube condition:

A = annealed; 1 = 10% cold work; 2 = 20% cold work.

Third character designates lot letter A, B, C, etc., or lot number 1, 2, 3, etc.

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

(2). Regions are shown on Drawing 286E240, Revision 3.

(3). Region B will contain SiC temperature sentinels listed in this column.

Lot identification:

Lot Identity	Heat Number	Melting Practice	Solution Annealing Temperature (°F)	Final Annealing Temperature (°F)	Structure
A	93132	VacMelt	2200	2000	Annealed
B	93132	VacMelt	2200	2000	10% CW
C	93132	VacMelt	2200	2000	20% CW
D	93132	VacMelt	2000	2000	Annealed
1	M1462	AirMelt	1950	1950	Annealed
2	M1462	AirMelt	2100	2100	Annealed
3	M1462	AirMelt	1950	1950	10% CW
4	M1462	AirMelt	1950	1950	20% CW

L-16D Capsules

Fabrication of the three L-16D capsule inserts is complete. These capsules are scheduled for encapsulation during April in order to have them available for reconstitution in S/A X098A.

10.3 CLADDING ALLOY SWELLING

The objective of this program is to measure cladding alloy swelling and S/A hardware volume changes in available irradiated specimens to determine the relationships between swelling, fluence, irradiation temperature, 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).

Examination of void formation in experimental fuel rod claddings irradiated in EBR-II using transmission electron microscopy (TEM) has continued. Investigation of two specimens of 50% cold-worked Type 316 from fuel rod F5Q revealed very few voids. The sections investigated were irradiated at 525 and 625°C (976 and 1158°F) to fluences of 3.1 and 2.4×10^{22} n/cm², $E > 0.1$ MeV-GE, respectively. Immersion density measurements⁽¹⁰⁻³⁾ had shown that each specimen had undergone a slight densification. Comparison of the postirradiation microstructures with that of an unirradiated archive sample showed that considerable recovery, as evidenced by a decrease in dislocation density, had occurred in both sections. The degree of recovery was greater in the specimen irradiated at 625°C (1158°F). Recrystallization had not occurred in either sample. Each specimen contained considerable carbide precipitation and the higher temperature specimen also apparently exhibited a few large sigma particles. Only isolated voids, generally within dislocation-free regions of the foils, were evident in either sample (Figure 10-1), whereas sections from companion rods clad with solution-annealed Type-316 and -321 examined previously,⁽¹⁰⁻²⁾ showed extensive void formation. Analysis of electron diffraction patterns from the cold-worked Type-316 confirmed the presence of $M_{23}C_6$ carbide particles and sigma phase in each sample. The specimen irradiated at the lower temperature also contained diffraction spots indicative of epsilon phase which is the precursor to martensite formation. Each of these cold-worked Type-316 specimens were slightly magnetic indicating that some martensitic transformation had occurred. Thermal control experiments are currently in progress to evaluate the degree of recovery of the cold work and extent of precipitation that occurs out-of-reactor. Initial TEM results from these tests show that very little recovery has occurred after aging at 593°C (1100°F) for 336 hours or at 649°C (1200°F) or 704°C (1300°F) for 24 hours. However, after 336 hours at 649°C (1200°F) the structure has partially recrystallized and after 336 hours at 704°C (1300°F), complete recrystallization has occurred as evidenced by the appearance of large grains.

The quantitative examination of two specimens of cold-worked and heat-treated Type-304 cladding from pin E1H-30 has been completed. This cladding had been cold worked 13% and then heat treated for 24 hours at 482°C (900°F) followed by 216 hours at 704°C (1300°F).

Immersion density measurements reported previously⁽¹⁰⁻³⁾ showed volume increases of 4.32% after irradiation at 503°C (935°F) to 4.7×10^{22} n/cm², $E > 0.1$ MeV-GE and 0.51% $\Delta v/v$ after irradiation at 593°C (1100°F) to 4.1×10^{22} n/cm², $E > 0.1$ MeV-GE. TEM examination of the unirradiated archive material showed a recovered microstructure containing extensive precipitation in slip traces. Extensive void formation had occurred in the recovered regions (see Figure 10-2). The specimens irradiated at 503°C (935°F) contained 1.4×10^{15} voids/cm³ with an average void diameter of 350Å, while the other specimens contained 6.8×10^{13} voids/cm³ with average diameter of 430Å. In each of these investigations, the void concentrations are based on foil thicknesses measured from stereo pairs.

The TEM examination of a section from rod E1H-26, solution-annealed Type 316 irradiated at 573°C (1064°F) to 4.8×10^{22} n/cm², $E > 0.1$ MeV-GE, has been concluded. This specimen had a volume increase of 4.6% $\Delta v/v$ as measured by immersion density. TEM examination revealed extensive populations of both voids and precipitate particles. Electron diffraction analysis identified the precipitates as $M_{23}C_6$. The sample region examined contained 1.4×10^{14} voids/cm³ with an average diameter of 830Å and 0.76×10^{14} precipitate particles/cm³ with an average dimension of 815Å. The particles were uniformly distributed within the grains with heavier concentrations in twin and grain boundaries. Stereo pair micrographs were used to determine the degree of coincidence of particles and voids. Nearly every precipitate particle (96%) had a void associated with it.

7028A

Figure 10-1. 50% CW Type-316 Cladding from Pin F5Q2 Irradiated at 525°C (976°F) to 3.1×10^{22} n/cm², $E > 0.1$ MeV-GE. Volume Change = -0.07%, Structure Shows Some Recovery with Isolated Voids (Arrowed).

7214A

Figure 10-2. Heat Treated Type-304 Cladding from Pin E1H-30, Section Irradiated at 593°C (1100°F) to 4.1×10^{22} n/cm², $E > 0.1$ MeV-GE. 6.8×10^{13} Voids/cm³ Mean Void Diameter = 430Å Volume Change = 0.51%.

10.4 CLADDING ALLOY MECHANICAL PROPERTIES

The objective of this subtask is to evaluate the postirradiation mechanical properties of available cladding alloy specimens, including precut tensile specimens from Task L and Task F fuel cladding and capsule tubing. Various uniaxial and biaxial techniques are being used to perform the mechanical property evaluations.

10.4.1 Stress Rupture Tests on F3A Fuel Rod Sections

Stress rupture tests were initiated on sections cut from remaining F3A fuel rods clad with either Type-316 or -304L stainless steel. These rods had been irradiated in EBR-II to fluences ranging from 2.3×10^{22} to 3.1×10^{22} n/cm², $E > 0.1$ MeV-GE with cladding temperatures of from 482°C (900°F) to 566°C (1050°F). The stress rupture tests were performed using the same equipment which was utilized in performing earlier burst tests on other F3A fuel rods.

Metallographic examination of cross sections of the fuel rod cladding indicated that intergranular attack of the cladding inside surface was as much as 7 mils in some of the specimens. The presence of this intergranular attack raised the possibility that the long-term mechanical properties of the cladding may be decreased by this attack. To ascertain this, it was decided to test some of the specimens with the mixed-oxide fuel inside the cladding while in other specimens the attacked region of the cladding would be removed electrochemically.

The first specimen was tested with the fuel inside the cladding at a temperature of 593°C (1100°F) at a hoop stress of 60,000 psi. This specimen had been irradiated at a temperature of 566°C (1050°F) with the peak fluence being 2.8×10^{22} n/cm², $E > 0.1$ MeV-GE. The specimen failed (small longitudinal crack) after 1.1 hours with the total strain ($\Delta D/D$) being 0.2%. The tests on the electrochemically treated cladding have been indefinitely delayed by a change in the task plan. The results from the only test performed are described above.

10.4.2 High-Pressure Stress-Rupture Rig

The high-pressure stress-rupture rig is being developed to provide a facility for testing fuel rod cladding and unirradiated tubing at higher pressure and temperatures closer to the actual temperature-of-irradiation of Task F fuel rods. This facility will provide stress-rupture data more representative of the as-irradiated fuel cladding by minimizing the thermal effects on cladding microstructure which occur when postirradiation tests are performed at temperatures higher than the temperature of irradiation. Assembly of this test system is in progress.

The Stress Rupture Unit III is modeled after a design implemented at WADCO.⁽¹⁰⁻⁴⁾ This design utilizes a closed, leak-tight high-pressure system rather than the slow-flowing system used in our earlier Units I and II. The new unit will be capable of testing 40 tubular specimens at one temperature from 538°C (1000°F) to 760°C (1400°F) and, since any number of specimens can be pressurized simultaneously, it is possible to have from one to forty different test pressures; 10,000 psi being the operational maximum. In practice, eight different pressures will be used.

Pressurization will take place after the specimens have reached equilibrium temperature. The exact pressure in each specimen will be determined with a high-precision (0.1% of full scale) 10,000 psi gauge connected to the supply line common to all the specimens. Each specimen will be isolated from the supply line with a manual valve and monitored with a 0 to 15,000 psi gauge. Forty such gauges will be mounted on a panel in rows of five; each row corresponding to a group of specimens enveloped by a cylindrical low-pressure jacket filled with argon at one atmosphere. The jackets are connected to a switch sensitive to small pressure increases so that failure of any one specimen within the jacket will activate the switch, causing an alarm to trip and a timer to stop. The pressure drop observed in the corresponding gauge will reveal the identity of the failed specimen. An isometric drawing of SRU-III is shown in Figure 10-3. The high-pressure supply system and the electrical control panel are independent of the main body to facilitate adaptation to hot cell operations.

The furnace has been modified to accept the eight low pressure jackets by drilling holes 4 inches in diameter through the lid and placing a stainless steel spider inside to support the low pressure jackets. Ceramic plugs with openings to accommodate the tubing involved fit into the large bore holes to prevent excessive heat losses at the top. The furnace has been fired for a period of 1-1/2 weeks.

Figure 10-3. Layout of Stress Rupture III

In order to determine the time and space dependence of temperature, a profile was run using five chromel-alumel thermocouples in conjunction with a 12-point mV recorder capable of recording emf from 0 to 150 mV and suppressing any amount within that limit. The span used was from 22.0 to 22.66 mV which corresponds to 532 to 546°C (990 to 1015°F). The temperature over a period of 67 hours varied only 1.7°C (2.9°F) well within the $\pm 3^\circ\text{F}$ required by ASTM.

10.4.3 Biaxial Creep of Type 304

Constant stress biaxial creep tests described previously^(10-5,-6) have now been terminated after 4,000 hours at 482°C (900°F). The six specimens tested had been fabricated from F2 and L2 Type-304 stainless steel capsule tubing. Irradiation conditions and test stress levels are given in Table 10-3. The test procedure (described in detail previously⁽¹⁰⁻⁵⁾) consisted of heating three-inch-long, sealed, prepressurized tube lengths to the test temperature. Diameter increases were measured by a contact profilometer; the initial and final diameter measurements were double checked with a second profilometry system. Unirradiated control specimens were tested in a separate facility. All the irradiated specimens had been irradiated at comparatively low temperatures (Table 10-3). A test temperature of 482°C (900°F) was chosen to reduce the annealing of irradiation damage and to avoid shrinkage effects due to swelling annealing.⁽¹⁰⁻⁶⁾ In addition, it was felt that it would be profitable to compare the postirradiation creep rate with in-pile creep rate data being generated in concurrent National Lab programs.

Table 10-3

900°F BIAXIAL CREEP SPECIMENS; IRRADIATION CONDITIONS AND STRAIN RATE ESTIMATIONS

Hoop Stress (ksi)	Specimen Number	Irradiation Temperature		Fluence x 10 ²²	Strain Rate
		(°C)	(°F)	E>0.1 MeV (GE)	(in./in./h x 10 ⁷)
Irradiated Controls					
35	F2G4	460	860	5.1 - 6.3	6.4
35	F2X1	391	735	5.3 - 6.0	2.5
35	L2A4	377	710	5.4 - 6.7	2.6
30	F2G2	391	735	6.5 - 7.1	2.3
30	L2G4	377	710	5.1 - 6.3	3.5
25	L2A2	377	710	6.8 - 7.2	0.8
35	B6	484	905	3.3	14.0
35	A2	424	795	3.2	7.0
25	I4	377	710	2.6	6.0
Unirradiated Controls					
35	C16				22.0
35	C17				22.0
30	C14				10.0
30	C15				10.0
25	C12				7.7
25	C13				7.7

Upon termination of the tests, the gas pressure remaining in the irradiated specimens was checked by puncturing each specimen and measuring the gas volume released. One of the specimens, F2G4, stressed at 35 ksi hoop stress, had failed. This specimen had been exposed to the highest irradiation temperature [460°C (850°F)] and exhibited a maximum strain (0.21%). Examination of the strain/time measurements indicates that the specimen had failed after approximately 2500 hours testing. Strain rate estimations have been made for each specimen and are reported in Table 10-3. As expected, all the strain rate values for the irradiated specimens are low. For the purpose of comparison, data from an earlier test series⁽¹⁰⁻⁶⁾ are included in Table 10-3 along with the unirradiated control values.

The control values are substantially higher than those previously predicted for unirradiated controls⁽¹⁰⁻⁶⁾ using a standard creep equation for Type-304 stainless steel. This discrepancy is attributed to the material composition and the analysis given for this material is being verified. Comparison of the high fluence ($\sim 6 \times 10^{22}$ n/cm², $E > 0.1$ MeV-GE) creep data with the lower fluence ($\sim 3 \times 10^{22}$ n/cm², $E > 0.1$ MeV-GE) creep results show a continued reduction in creep rate as a result of the increased fluence. Specimen F2G4, irradiation temperature 460°C (860°F), had a substantially higher creep rate than either of the two specimens F2X1 and L2A4 which, although tested at the same stress, had lower irradiation temperatures of 391°C (735°F) and 377°C (710°F), respectively. This is in agreement with the earlier data which showed specimen B6, the higher irradiation temperature specimen, deforming at approximately twice the rate of specimen A2, the low irradiation temperature specimen, Table 10-3.

In general, it appears that in this irradiation temperature region, a comparatively small difference in irradiation temperature or fluence has a more significant effect on creep rate than was previously suspected.

10.5 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.

10.5.1 Type-321H Air-Melted Tubing

A total of 3,000 feet of Type-321H air-melted tubing, including annealed, 10% and 20% cold-worked tubing, is undergoing incoming inspection and characterization. This tubing includes four different lots all from the same heat: 1065°C (1950°F) anneal, 1148°C (2100°F) anneal, 1065°C (1950°F) anneal +10% cold worked, and 1065°C (1950°F) +20% cold worked.

10.5.1.1 Burst Tests

Burst tests were performed on the four lots of tubing in air at 25, 371, 482, 593, and 704°C (78, 700, 900, 1100, and 1300°F). These tests were pressure-rate controlled at 1,000 psi per minute and both pressure and diameter increase were recorded during the test. The results (Table 10-4) are summarized in terms of stress-strain behavior for the transverse direction of the tubing, neglecting any stress or strain in the longitudinal direction.

The results show that increasing the annealing temperature from 1065°C (1950°F) to 1148°C (2100°F) decreases the yield stress (approximately 10 to 15 ksi) at all test temperatures and the ultimate burst stress (3 to 13 ksi) at temperatures below 593°C (1100°F). This reduction in hoop strength is related to a larger grain size and/or more dissolution of TiC precipitate, both of which occur more readily at the higher annealing temperatures. The reduction in strength, however, is accompanied by a slight increase in ductility.

The effect of cold work was to increase hoop strength and decrease ductility. Maximum burst strength was attained in the 20% cold-worked material which was slightly greater than prototypical FFTF Type-316 20% cold-worked tubing. The stability of the cold work decreases as test temperature increases as shown by the large negative slope in the yield strength versus test temperature curves, Figure 4. Extrapolation of the test data suggests that complete recovery resulting in as-annealed burst properties occurs between 760°C (1400°F) and 815°C (1500°F) in less than 30 minutes, the time required to perform these tests.

Table 10-4

**BIAXIAL BURST PROPERTIES OF TYPE-321H
AIR MELTED TUBING^(a)**

Specimen	Material Condition	Test Temperature (°F)	0.2% Offset Yield Stress (psi)	Ultimate Burst Stress (psi)	Burst Strain ($\Delta D/D\%$)	
					Chart ^(b)	Final ^(c)
1A24A	1950 Anneal	1300	33,750	44,600	13.3	13.6
1A24B	1950 Anneal	1300	35,000	45,700	10.1	10.0
1A24C	1950 Anneal	1100	41,500	56,150	11.7	10.4
1A24D	1950 Anneal	1100	41,500	56,800	10.4	11.6
1A24E	1950 Anneal	900	45,600	64,000	9.8	10.4
1A24F	1950 Anneal	900	46,000	63,500	7.6	8.0
1A24G	1950 Anneal	700	49,500	66,000	8.8	8.0
1A24H	1950 Anneal	700	49,600	66,300	8.8	8.4
1A24I	1950 Anneal	70	60,000	88,000	20.8	20.8
1A43A	2100 Anneal	1300	25,200	45,250	18.2	14.0
1A43B	2100 Anneal	1300	25,800	46,500	16.0	12.8
1A43C	2100 Anneal	1100	26,500	53,900	14.4	14.0
1A43D	2100 Anneal	1100	26,000	53,300	12.0	12.4
1A43E	2100 Anneal	900	29,500	57,500	12.2	12.0
1A43F	2100 Anneal	900	29,500	59,500	13.5	13.2
1A43G	2100 Anneal	700	31,400	60,300	14.4	13.2
1A43H	2100 Anneal	700	31,275	60,100	15.2	14.0
1A43I	2100 Anneal	70	43,200	75,000	28.8	24.0
11CW10A	10% CW	1300	51,400	60,250	4.9	5.2
11CW10B	10% CW	1300	50,300	60,250	5.2	4.8
11CW10C	10% CW	1100	72,600	75,500	1.2	1.6
11CW10D	10% CW	1100	71,500	75,500	1.7	2.0
11CW10E	10% CW	900	83,800	86,500	0.8	0.8
11CW10F	10% CW	900	83,600	85,500	0.7	0.8
11CW10G	10% CW	700	88,800	91,400	0.85	1.2
11CW10H	10% CW	700	86,700	91,500	0.77	0.8
11CW10I	10% CW	70	104,300	113,000	4.64	2.8
12CW9A	20% CW	1300	53,500	66,600	3.8	3.6
12CW9B	20% CW	1300	51,500	66,000	4.4	4.0
12CW9C	20% CW	1100	84,000	89,250	1.5	1.6
12CW9D	20% CW	1100	87,100	91,600	1.0	1.2
12CW9E	20% CW	900	106,500	106,700	0.68	1.2
12CW9F	20% CW	900	105,500	106,500	0.58	0.8
12CW9G	20% CW	700	112,800	113,600	0.62	0.8
12CW9H	20% CW	700	112,500	113,300	0.58	0.8
12CW9I	20% CW	70	131,200	139,000	2.3	2.4

(a) Pressure rate during test = 1,000 psi/min.

(b) Measured during test at time of rupture.

(c) Measured after test on burst sample.

Figure 10-4. Temperature Dependence of Yield Strength for Air Melted Type-321H Stainless Steel Tubing; Yield Stress was Measured in Transverse Direction Under Pressure Tests

10.5.1.2 Metallography

Both optical and transmission electron microscopy were performed on the four lots of tubing. Typical optical micrographs are shown in Figure 10-5. The major difference in these tubing lots is apparent in grain size. All tubing annealed at 1065°C (1950°F) had a grain size of ASTM No. 10* or finer. The tubing annealed at 1148°C (2100°F), however, had an average grain size of ASTM No. 6. The larger grain size tubing also exhibited a fine grain layer on the inside surface of ~ASTM No. 10 similar to the 1065°C (1950°F) annealed tubing. The cause of this layer is presently under investigation since it appears to be present in other Type-321H tubing⁽¹⁰⁻⁷⁾ lots purchased to date.

The inclusion content was also determined using ASTM standards for all four tubing lots. It is characterized mainly as Silicate Type (C) and Globular Type oxides (D). The ASTM inclusion ratings are as follows:

Tubing Lot	ASTM Inclusion Rate
1065°C (1950°F) Anneal	1-A-T, 2-C-T, 2-D-T
1148°C (2100°F) Anneal	1-B-T, 2-C-T, 1-D-T
1065°C (1950°F) Anneal + 10% C.W.	2-C-T, 2-D-T
1065°C (1950°F) Anneal + 20% C.W.	2-C-T, 1-D-T

The silicate-type inclusion is quite common in austenitic stainless steels and the 2-C-T rating represents a normal expected quantity and size for this type of inclusion. The globular inclusion is primarily Ti(C,N) titanium carbonitride, an inclusion found primarily in titanium-bearing steels.

*ASTM microgram size number per ASTM-E112.

71 PX 0660

71 PX 0665

71 PX 0670

71 PX 0675

100X

71 PX 0661

71 PX 0666

71 PX 0671

71 PX 0676

250X

Figure 10-5. Microstructure of Air-Melted Type-321H Tubing as Shown in Optical Microscopy; Etchant $\text{HC1} + \text{H}_2\text{O}_2$
a) 1950°F Anneal, b) 2100°F Anneal, c) 1950°F Anneal + 10% CW, d) 1950°F Anneal + 20% CW

Transmission electron microscopy has been used to characterize the dislocation and precipitate substructure in the four different tubing lots. Figure 10-6 shows the substructure present in the different lots.

Increasing the annealing temperature from 1065°C to 1148°C (1950°F to 2100°F) has resulted in a more complete solution of TiC; compare Figures 10-6a and 10-6b. However, the larger TiC particles have remained undissolved by the 1148°C (2100°F) treatment and presumably either longer time at this temperature or an increase in the annealing temperature would be necessary for complete dissolution. Thus, the creep-rupture strengths reported earlier⁽¹⁰⁻²⁾ for the 1148°C (2100°F) anneal, although improved over the 1065°C (1950°F) treatment, probably do not represent the maximum attainable for this material. Increasing the degree of cold work from 10 to 20% has simply caused a corresponding increase in dislocation density (Figure 10-4). Although the dislocation distribution appears somewhat heterogeneous and a tendency for dislocation cell formation exists in both 10 and 20% cold-worked specimens, other regions of the same foils exhibited a pronounced tendency toward planar slip.

10.5.2 Vacuum-Melted Type-321 Stainless Steel Tubing

A lot of 1500 feet of vacuum-melted Type-321 stainless steel tubing will be delivered next quarter. Transmission electron microscopy on sheet stock processed from the same heat as the tubing has confirmed that the 1204°C (2200°F) anneal given to one-half of the tubing will completely dissolve the TiC. Figures 10-7a and 10-7b illustrate the degree of TiC precipitation after 1093 and 1204°C (2000 and 2200°F) anneals, respectively. The kinetics of TiC precipitation at 1093°C (2000°F) are such that no TiC precipitation has occurred during a 1204°C (2200°F) +50% cold work + 1-minute 1093°C (2000°F) treatment, Figure 10-7c. This treatment is specified in the tubing order as it appears to represent a treatment that improves creep-rupture strength and permits satisfactory ultrasonic inspection (fine grain size).

The time at 1093°C (2000°F) after the 1204°C (2200°F) solution anneal is critical, as after only four minutes significant TiC precipitation has occurred (Figure 10-7d). Any TiC that precipitates during the 1093°C (2000°F) treatment will not be available for precipitation during creep resulting in a probable reduction of creep strength.

10.5.3 Ti-Modified Type-316 Stainless Steel Tubing

An order for 1000 feet each of 0.250- and 0.230-inch o.d. tubing to be fabricated from the Ti-modified Type-316 material available at ORNL has been initiated. The tubing is to be supplied in both the annealed and 20% cold work condition for both sizes. The tubing is ordered to RDT Standard E13-8T with approved modifications and with the following process sequence: the first two process annealing steps are to be performed at 1204°C (2200°F) (to dissolve all carbides) with each subsequent anneal (three to four) at 1050°C (1922°F). This process is believed to represent the best compromise between improved high-temperature strength and good ductility after irradiation.

10.5.4 Stabilized Grades of Austenitic Stainless Steels

The open literature on the physical metallurgy of the stabilized Cr-Ni austenitic stainless steels has been reviewed with emphasis on Type 321 (Ti stabilized). A summary of the mechanical property sections of this review is presented below.

The mechanical properties, both short and long term, for Type 321 are reported to be slightly inferior to the Mo-bearing Type 316, while slightly better or similar to the standard Type 304.⁽¹⁰⁻⁸⁾ A possible explanation of this and a discussion of its general validity follow.

Strong carbide forming elements, such as Ti or Nb, are added to the standard Cr-Ni steels in proportion necessary to react with the carbon in the alloy to preferentially form the MC-type carbide and suppress the formation of the grain boundary $(Cr,Fe)_{23}C_6$. Inhibiting the formation of this carbide reduces the alloy's susceptibility to sensitization. In addition to this obvious effect, Ti and Nb also have much more subtle effects. They (1) alter mechanical properties, (2) reduce the stacking fault energy,^(10-9,-10,-11) (3) promote stacking fault precipitation,^(10-9,-11) and (4) promote sigma phase formation.⁽¹⁰⁻¹²⁾

7133A

(a)

7144

(b)

7155A

(c)

7129

(d)

(ALL PHOTOGRAPHS HAVE THE SAME MAGNIFICATION)

Figure 10-6. Sub-Structure of Air Melted Type-321H Stainless Steel Tubing, a) 1950°F Anneal, b) 2100°F Anneal, c) 1950°F Anneal + 10% Cold Work, d) 1950°F Anneal + 20% Cold Work

(ALL PHOTOGRAPHS HAVE THE SAME MAGNIFICATION)

Figure 10-7. Illustration of TiC Dissolution in Vacuum Melted Type-321H Material Used for Fuel Cladding Tubing. a) 2000°F (1093°C) for One Minute, b) 2200°F (1204°C) for 15 Minutes, c) 2200°F (1204°C) for 15 Minutes + 50% Cold Work + 2000°F (1093°C) for One Minute, d) 2200°F (1204°C) for 15 Minutes + 50% Cold Work + 2000°F (1093°C) for 4 Minutes

The effect of Ti additions on strength depends upon (1) the Ti-to-C ratio, (2) total TiC available for precipitation, (3) amount of Ti and C in solution, and (4) the morphology of TiC when precipitation occurs. Clearly, one must characterize these parameters; if not directly, then indirectly by reporting such variables as chemical composition, prior thermal and mechanical histories, and grain size, along with any mechanical property data.

Freeman, et al.,^(10-12 to 10-14) have provided the most comprehensive elevated temperature mechanical property evaluation of Type-321 stainless steel. Their conclusions are that the creep-rupture strength of Type-321 stainless steel increases with (listed in order of decreasing effectiveness) (1) increased annealing temperature up to the temperature of complete TiC dissolution, (2) total amount of TiC available for precipitation, and (3) decreasing Ti/C down to a 1:1 atom ratio. They conclude that strain aging plays an important role in the creep strength of this alloy.

Increasing Ti/C over 1:1 and decreasing the annealing temperature promotes the formation of TiC in a needle-like morphology.⁽¹⁰⁻¹⁵⁾ Reportedly,⁽¹⁰⁻¹⁶⁾ this needle-like form of TiC does not interact with dislocations and, thus, strain aging will not occur. However, the granular form of TiC does interact strongly with dislocations.⁽¹⁰⁻¹⁶⁾ Figure 10-8 presents a compilation of the data available in the open literature on precipitation kinetics of Type-321 stainless steel in the form of a time-temperature precipitation (TTP) diagram. The exact positions of the various precipitation curves are a function essentially of Ti/C and the thermal-mechanical history. It can be seen from this diagram that proper thermal exposure can prevent formations of the needle-like TiC. Thus, if properly performed, postanneal aging is not always detrimental, but alone does not appear to offer any improvement in creep resistance. However, postanneal aging, when used in conjunction with plastic deformation, has been found to promote substantial increases in both short- and long-term mechanical properties.^(10-17,-18)

The short-term mechanical properties are less affected by thermal treatments than are long-term properties. As a rule, the yield strength decreases with a corresponding increase in ductility as the annealing temperature is increased.

Figure 10-8. Time-Temperature-Precipitation Diagram for Cr_{23}C_6 , TiC, and SIGMA (σ) in Type-321 Stainless Steel

In summary, the strength of the stabilized steels is so dependent on prior thermal and mechanical histories that properties reported as typical for these alloys must be viewed with skepticism since most of the reported data has been generated from steels that were not heat treated for maximum creep-rupture strength. The mechanical properties of the stabilized alloys do appear to be slightly inferior to those of the Mo-bearing grades, but this is believed to be due to the formation of the needle-like TiC precipitate during creep. If properly processed to inhibit this precipitation and promote a strengthening strain aging precipitation either before or during creep, it is believed that the strength of these alloys can be made comparable to the Mo-bearing grades. A development program has been initiated to identify the proper processing conditions necessary to achieve these ends.

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11. TASK P—FUEL ELEMENT MODELING

Task Leader: F. E. Gelhaus*

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.

The AEC-RDT has officially designated LIFE as the standard LMFBR fuel element modeling code. The LIFE-2 version is now available for use on a local IBM computer. The sample problem simulates the insertion of the rod into sodium at 700°F (48 hours), a rise to full reactor power (1 hour), two periods of steady-state operation at full power (500 hours each), a decrease to zero reactor power (1 hour), and a 24-hour cool-down to room temperature. The problem thus simulates a total of 1074 hours. The causes of the various differences noted between our output and the ANL run results are being sought and corrections are being made.

During this quarter, planning has been initiated for the development of a cladding structural code to be used in conjunction with a fuel modeling performance code to compare fuel rod structural design and lifetime criteria with predicted performance. The general purpose finite element computer codes, NASTRAN and MARC CDC, have been reviewed to determine the relative usefulness for the three-dimensional analysis of a fuel rod. For the present, the MARC CDC, with better element structure and elastic-plastic creep analysis capability, appears to be the more applicable. However, since the cost involved in running these codes will be high, the preliminary development plan does not involve the use of three-dimensional codes until the potential of the two-dimensional modeling has been fully explored. The modification of the two-dimensional code CREEP-PLAST to include flux-induced creep and cladding swelling was completed and sample problem involving creep under a radial thermal gradient was run to demonstrate capability. The preliminary development plan will utilize CREEP-PLAST in two important tasks. The first involves exploring the validity of using superposition versus a complete coupled fuel-cladding interaction analysis, and the other involves the validity of using a generalized plane-strain approximation to the three-dimensional problem.

*Contributors are listed in Appendix B.

12. QUALITY ASSURANCE

12.1 GENERAL

The revised PA-10 Quality Assurance Plan was issued during this period. A revision to the plan is being prepared to comply with comments received from RDT.

BRD-QA continued to implement the PA-10 QA Program during this period through its planning, engineering, surveillance, verification, and audit activities.

A BRD Management System for unplanned events (incidents, failures, nonconformances, and unsatisfactory conditions) in development programs was established during this reporting period by means of procedural controls.

12.2 SUMMARY OF QA ACTIVITIES

An audit of the F9E hardware and records was initiated because of uncertainties related to the configuration of the lower adaptor and lower shield. The audit revealed requirements for corrective action to assure that the hardware was satisfactory for experimental uses. These corrective actions have been performed.

Examinations of spot welds holding spacers to end plugs on F10 fuel rods revealed one broken weld. The broken weld was repaired; detailed inspection of all of the F10 fuel rods' spot welds was not required because the MRB determined these spot welds to be nonessential for completion of the experiment. These spot welds are intended to position the spacers during assembly operations. After assembly, the welds are not likely to fail and the positions of the spacers are assured by the assembly independent of the spot welds.

Material intended for fabrication of F9F grid spacers was rejected because of unusually high inclusion content. Inclusions for this strip stock are not controlled characteristics; however, high inclusion content was considered undesirable because it would add a variable of unknown importance to the experiment. Replacement strip stock will be vacuum melted and evaluated per the requirements of RDT Standards.

Reviews of design and fabrication documents for F20A fuel rods identified the need to change the design of the lower end plug to fit the test location in EBR-II.

A material review board determined that three F-11A fuel rods needed to be replaced because the end plug welds were considered undesirable for the intended service. Uncertainties of the quality of welds was the reason for rejection,

Nonconformance activities during this period consisted of:

Total Nonconformances	6
Nonconformances by Category	
MRB—Use as is	1
MRB—Repair	1
MRB—Scrap	2
1st Level Disposition	2
Nonconformances by Project Task	
PA-10 Task F	4
PA-10 Task C-1	1
PA-10 Task L	1

APPENDIX A. PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Nos. 1 through 115 from July 1959 through March 1972.

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 Report
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

GEAP-10028-39
Thirty-Ninth Quarterly Report
May–July 1971

GEAP-10028-40
Fortieth Quarterly Report
August–October 1971

GEAP-10028-41
Forty-First Quarterly Report
November 1971–January 1972

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)

GEAP-4469

Experimental Studies of Transient Effects
in Fast Reactor Fuels,

Series III,

Pre-Irradiated Mixed Oxide (PuO_2 - UO_2),

Irradiations Final Report,

Transient Irradiations;

J. E. Hanson, J. H. Field

(February 1968)

GEAP-4471

Calculated Nuclear Parameters and

Their Uncertainties in a 1000 MWe FCR;

P. Greebler, et al.

(October 1966)

GEAP-4472

User Requirements for Cross Sections

in the Energy Range from 100 eV to 100 keV;

P. Greebler and B. A. Hutchins

(1966)

GEAP-4580

FLUX-A Computer Code

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to Small Reactivity Oscillations;

P. Greebler, B. A. Hutchins, and M. D. Kelley

(March 22, 1965)

GEAP-4630

The Effects of Resonance Overlap on

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B. A. Hutchins

(August 1964)

GEAP-4675

Some Electrical Properties of

the Plutonium-Uranium Mixed Oxide System;

W. E. Baily

(July 1966)

GEAP-4770

A Technical and Economic Evaluation of

Vented Fuel for Sodium-Cooled Fast Ceramic Reactors;

G. L. O'Neill, J. Duffy, J. C. Gilbertson, F. W. Knight,

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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,
Series II;
J. E. Hanson, J. H. Field, and S. A. Rabin
(June 1965)

GEAP-4809

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

GEAP-4819

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

GEAP-4822

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

GEAP-4870

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

GEAP-4877

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

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

GEAP-4879

Thermal Conductivity of
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F. J. Hestzler, T. E. Lannin, K. J. Perry and E. L. Zebroski
(January 1968)

GEAP-4897

Irradiation Behavior of
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at 70,000 MWd/Te;
E. L. Zebroski, W. E. Baily, and D. P. Hines
(March 1965)

GEAP-4922

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

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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
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P. Greebler, P. Aline, B. Hutchins
(January 1967)

GEAP-5273

FORE II, A Computational Program for
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Reactor Performance;
B. E. Lawler, et al.
(September 1967)

GEAP-5340

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

GEAP-5344

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

GEAP-5355A

Burnup: A FORTRAN IV Code for Computing U and Pu
Fuel Burnup from U, Pu, and Nd Mass Spectrometric
Measurements.
B. F. Rider
(January 1970)

GEAP-5411

Turbulent Velocity Distribution in a Rod Bundle;
D. J. Bender, D. M. Switick, J. H. Field
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GEAP-5493

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

GEAP-5494

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

GEAP-5556

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

GEAP-5583

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

GEAP-5633

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

GEAP-5635

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

GEAP-5683

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

GEAP-5744

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

GEAP-5748

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

GEAP-5761

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

GEAP-5570

Short Term Fast Flux (EBR-II) Irradiation
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S. A. Rabin, W. W. Kendall, W. E. Baily
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GEAP-10046

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A Two-Dimensional Perturbation Module
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M. D. Kelley and T. R. Jones
(April 1969)

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Turbulent Heat Transfer in a Rod Bundle
with Liquid Metal Coolant;
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GEAP-10054

Compressive Creep Characteristics
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P. E. Bohaboy, R. R. Asamoto, A. E. Conti
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Mechanical Properties of Irradiated Type-304
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F. A. Comprelli, H. J. Busboom, M. F. Gebhardt
(June 1969)

GEAP-10066

Mechanical Properties Evaluation of Austenitic Stainless Steels Irradiated in EBR-II;
T. Lauritzen, A. Withop, G. P. Ferguson
(July 1969)

GEAP-10385

High Burnup Performance of Powder & Pellet Mixed-Oxide Fuel;
K. J. Perry, W. E. Bailly
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GEAP-10403

Axial Expansion of Mixed-Oxide Fuel Measured In-Pile;
D. A. Cantley, G. R. Hull, C. N. Craig
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The Melting Point of Plutonia-Urania Mixed Oxides Irradiated to High Burnup;
J. L. Krankota, C. N. Craig
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GEAP-13517

Empirical Analysis of Swelling of Austenitic Stainless Steels in Fast Reactors;
F. A. Comprelli
(September 1969)

GEAP-13537

User's Manual for DOT2DB: A Two-Dimensional Multigroup Discrete Ordinates Transport/Diffusion Code with Anisotropic Scattering;
R. Protsik, E. G. Leff
(December 1969)

GEAP-13543

Molten Fuel Movement in Transient Overpower Tests of Irradiated Oxide Fuel;
T. Hikido
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GEAP-13549

Performance of Plutonium-Uranium Mixed Oxide Fuel Pins in a Fast Reactor (EBR-II) to 50,000 MWd/Te;
R. C. Nelson
(August 1969)

GEAP-13550

The Influence of Surface Defects on Short-Term Mechanical Properties of Austenitic Stainless Steel Tubing;
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GEAP-13551

Effects of Nuclear Methods Approximations on LMFBR Analysis;
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GEAP-13553

An Evaluation of the Quality of Seamless Austenitic Stainless Steel Tubing;
T. A. Lauritzen
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GEAP-13556

BICYCL - A Computer Code for Two-Dimensional Analysis of Equilibrium Fuel Cycles;
P. C. Vaughn
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GEAP-13562

Transient Overpower Irradiation of Axially-Restrained, Zero-Burnup Fast Reactor Fuel Specimens;
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J. H. Field
(September 1969)

GEAP-13571

Evaluation of Physical and Mechanical Properties of Type-304 Stainless Steel After Irradiation to 3.9×10^{22} n/cm² Total Fluence;
H. J. Busboom, R. R. Asamoto
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GEAP-13582

Summary of (U,Pu)O₂ Properties and Fabrication Methods (November 1970);
B. F. Rubin
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GEAP-13591

Pu-239 Unresolved Resonance Data
for ENDF/B Version II;
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(April 1970)

GEAP-13592

ENDRUN-1, A Computer Code to Generate
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B. A. Hutchins and L. N. Price
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Fuel Zero to 3,000 MWd/Te;
R. R. Asamoto, P. E. Bohaboy,
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Liquid Metal Coolant;
K. E. Gregoire, P. E. Novak, R. E. Murata
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Testing Needs;
W. G. Meinhardt
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GEAP-13639-1, -2

Core Design Development Needs in Relation to
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Compiled by L. M. McWethy, P. R. Pluta, D. B. Sherer
(October 1970)

GEAP-13672

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M. D. Kelley
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GEAP-13686

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Fuel with Strong Radial Restraint;
R. C. Nelson, et al.
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GEAP-13703

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GEAP-13704

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B. A. Hutchins, C. L. Cowan, M. D. Kelley, J. E. Turner
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Pellet-Oxide Fuel Under Transient Overpower;
T. Hikido, J. H. Field
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GEAP-13729

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(September 1971)

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GEAP-13731

Measured Loss of Selected Fission Products from High
Burnup Mixed-Oxide Fuel in a Miniature Pumped
Sodium Loop;
D. E. Plumlee and P. E. Novak
(August 1971)

GEAP-13732

Compressive Creep of Urania-Plutonia Fuels;
S. K. Evans, P. E. Bohaboy, R. A. Laskiewicz
(August 1971)

GEAP-13733

Thermal Conductivity of Uranium-Plutonium Oxide;
R. A. Laskiewicz, G. F. Melde, S. K. Evans,
P. E. Bohaboy
(September 1971)

GEAP-13736

Characterization of Cold-Worked Tubing of Types-304H
and -316H Stainless Steel;
T. A. Lauritzen and H. J. Busboom
(August 1971)

GEAP-13738

Fast Neutron Damage in Type-304
Stainless Steel and High-Nickel Alloys;
P. W. Sandersky, P. J. Ring and R. T. Penrose
(December 1971).

GEAP-13740

TDOWN—A Code to Generate Composition and
Spatially Dependent Cross Sections;
C. L. Cowan, B. A. Hutchins, J. E. Turner
(August 1971)

GEAP-13765

Design Basis of the F9E Grid Spaced
LMFBR Diagnostic Assembly for
Irradiation Test in EBR-II;
P. R. Pluta and W. G. Meinhardt
(December 1971).

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