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SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

FORTY-THIRD QUARTERLY REPORT

MAY-JULY 1972

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

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

Forty-Third Quarterly Report

May—July 1972

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Prepared for the
United States Atomic Energy Commission
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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	=	(GE) Breeder Reactor Division
BUBBLE	=	coolant boiling and channel voiding code, a subroutine of FREADM
co-ppt	=	coprecipitated
CSEWG	=	Cross Section Evaluation Working Group
C.W.	=	cold worked
D Plant	=	Demonstration Plant
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
FSAR	=	Final Safety Analysis Report
FTR	=	Fast Test Reactor
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
ℓ	=	angular momentum states
lb _f	=	pounds force
lb _m	=	pounds mass
LASL	=	Los Alamos Scientific Laboratory
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
mg	=	milligram
Mod	=	modified
mole	=	gram molecular weight
MRB	=	Material Review Board
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
% Δk	=	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
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
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

PREFACE

A major contract revision was made effective July 1, 1972, in which the tasks under Project Agreement (PA) 10 are redefined and reassigned. This 43rd quarterly report is the last one to be written under the old task definitions and assignments. The report of each individual task includes an indication of its future situation under the new definitions and assignments.

In a formal sense PA-10 continues under the title "Core Engineering," and the new PA-10 will include many of the old topics. An additional new project agreement, "Safety Engineering," will encompass other topics of the old PA-10 as well as some newly defined ones. These changes are broadly indicated by the following diagram; the task outlines for the new PA-10 and the Safety Engineering project are as follows:

CORE ENGINEERING (PA-10)

TASK 1 — Engineering Coordination and Liaison

Administration, project engineering, liaison with national laboratories, quality assurance and standards, scheduling.

TASK 2 — Fuel Rod Engineering

Test fuel rod design, in-pile tests and evaluation—normal operation, material properties evaluation.

TASK 3 — Fuel Assembly Engineering

Grid and wire wrap test assembly design, operation, and evaluation, out-of-pile testing, operating limits analysis.

TASK 4 — Core Restraint Engineering

Restraint analysis, test mechanism design, scoping tests—2D, and planning for 3D tests.

TASK 5 — Control and Blanket Assembly Engineering

Conceptual design and test planning, integration with ARD, AI, and HEDL programs.

TASK 6 — Defected Fuel Testing

Design, operation, and evaluation of in-pile experiments in GETR and EBR-II.

SAFETY ENGINEERING (PA-XX)

Coordination and Liaison

Administration, project engineering, liaison with national laboratories, quality assurance and standards, scheduling.

Design Analyses

Accident analyses through HCDA, comparison with FFTF results, assessment of development priorities.

Data and Methods Application

Acquisition and evaluation of computer methods, feedback of experimental data into models, comparison of methods, specification and direction of experiments in national laboratories.

Documentation

Completion of final report on computer codes previously developed under PA-10—FREADM, TOFF, etc.

ABSTRACT

This report is the forty-third 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 May through July 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.

1. SUMMARY

1.1 TASK B—DEFECTED FUEL BEHAVIOR

Design of the B9B capsule and fabrication of fuel for the capsule was completed. Tests to determine the effect of fuel geometry and the size of the fuel-cladding gap on the release of gas from defected fuel for correlation with the test results to be obtained from the B9B experiment were initiated and completed. These tests show that the mass flowrate of gas escaping from a failed fuel rod is greatly affected by the loss of pressure due to friction and to a lesser degree by any temperature increase of the gas.

Post-test examination of capsule B9D continued. The fuel rods were removed from the capsule for examination. Visual examination and diameter measurements of the fuel rods showed that essentially no deterioration or swelling of the rods had occurred following the 3300 MWd/Te irradiation in the defected condition. Maximum diameter increase was ~ 0.001 inch.

Fabrication of the sodium fuel reaction study capsules B11A, -B, and -C continued. Fabrication of capsule hardware was initiated and nearly completed. Fuel fabrication was completed and the fuel rods loaded and welded shut.

Preliminary review of the Run-Beyond-Failure (RBF) test proposal was completed. A meeting was held with ANL/EBR-II personnel to discuss the feasibility of the test program and technical problems associated with doing tests in EBR-II.

Work was continued on the development of engineering expressions for prediction of dimensional changes of defected fuel rods. Present work is aimed at development of expressions to define the effects of the fuel rod defect geometry and the change in volume of the fuel resulting from the formation of fuel-sodium reaction product on the surface of the fuel.

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

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

High-Burnup Fuel Experiments

Capsule C4M machining operations and the open-shop Detailed Assembly Instructions package have been completed. Assembly operations have ceased at a controlled work-stoppage point to await AEC-RDT fiscal-year 1973 direction relating to disposition of the C4M capsule and its test plans. All control documentation and capsule components are presently in bonded storage in a completely traceable form.

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

Comparative Tests

Detailed physics calculations have resulted in a recommended design modification to the C6 comparative test assembly (three independent inner capsules) that will allow testing of EBR-II pre-irradiated test specimens with an axial power shape comparable to EBR-II. The modifications involve incorporation of a boron carbide flux filter, including collars near the ends of the fuel columns for suppressing end flux peaking and providing axial power shape control. The basic concept of the design is similar to that already in use on ANL Mark II loop flux-filtered tests (e.g., H-3 test).

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/Ta)	
					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-cad compatibility.	0.144	2.2	0.006	86	25	1.97	316	V-Raft GETR	1100	21	113,500	Reported in GEAP-13729
		0.204	2.25	0.017*	84	25	1.97	316				57,000	
B8D	Fuel-sodium compatibility versus density, stoichiometry and Pu content	0.214	1.75	0.0065	80-88	40	1.97	316	V-Raft GETR	950	17.5	41,000	Reported in GEAP-13729
		0.214	1.75	0.0065	80-88	25	1.97					43,500	
		0.214	1.75	0.0065	80-88	25	1.94					41,400	
B9A (2 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	23.4	8,600	Reported in GEAP-13730
		0.216	5	0.005	89	25	1.95			971	23.4	8,600	
E1J-B1	Thermal expansion clad defect mechanism	0.221	5	0.0007	94	25	2.00	316	Trail Cable	835	17.5	50	Examination completed
E1J-B2	Thermal expansion clad defect mechanism	0.221	5	0.0007	92	25	2.00	316	Trail Cable	860	19.0	300	Examination completed
B9C (2 rods)	Swelling behavior of defected fuel rods in flowing sodium	0.216	4.25	0.004	83.6	25	1.97	316	Special V-Raft GETR				Test terminated
		0.216	4.25	0.004	83.6	25	1.97	20% CW 316 Annealed					
B9D (2 rods)	Two defected fuel rods. Fuel swelling in clean sodium (< 5 ppm Oxygen).	0.215	4.25	0.005	83.5	25	1.97	316 20% CW	Special V-Raft GETR	1120 1100	16.5	~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 complete. Test Terminated
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	Fabrication 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	Fabrication 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	Fabrication in progress.

* Large gap required to accommodate sampling device

Table 1-2
TASK C-1-EXPERIMENTAL TESTS

Design Parameters										Status							
Capsule	Objective	O.D. (inches)	Fuel Length (inches)	Fuel Density (% TD)		Fuel Composition			Cladding Material	Fabrication	Pre-irradiation	TREAT Transient Irradiation		Examinations ¹			
				Pellet	Smear	%PuO ₂	%U-235	O/M				Calibration	Test	Nondestructive		Destructive	Report
														Before Transient	After Transient		
C4A	Transient failure threshold for zero burnup, mixed-oxide fuel as a function of axial restraint density, and form (pellet versus powder)	0.2159	14.23	93.8	90.4	19.4	0.57	1.983	316SS								GEAP-13562
C4B		0.2155	14.21	94.5	91.2	19.4	0.57	1.983	316SS								
C4C		0.2164	14.19	83.1	80.4	20.9	0.91	1.983	316SS								
C4D		0.2154	14.22	83.7	80.1	20.9	0.91	1.983	316SS								
C4I		powder	13.4E		83.5	20.2	0.89	1.973	316SS								
C4J		powder	13.4E		83.5	20.2	0.89	1.973	316SS								
C4E	Comparison with C4A and C4B test results (high density pellet fuel)	0.2159	13.46	83.2	80.5	28	92.7	1.974	316SS	C	C	P	P	C	P	P	Project terminated
C4F		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 B to determine effect of burnup on failure threshold (transient magnitude will be varied).	0.2154	13.50	94.4	90.5	28	89.3	1.973	316SS	C	C	C	C	C	C	C	
C4H		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 to determine effect of burnup on failure threshold of powder fuel (transient magnitude will be varied)	0.2205	13.50	Powder	84.1	28	93.0	1.972	316SS	C	C	P	P	C	P	P	Project terminated
C4L		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/ft) 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	Project interrupted; awaiting final AEC-RDT disposition.
C4N	Comparison to C4M (PNL-2-2).	0.2126	13.56	93.4	87.3	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 fuel; investigate potential for fuel relocation under prototypical conditions of length, restraint, and fission gas.	0.214	24	95	91.5	0	12	2.000	316SS								GEAP-13543
C5E		0.214	24	95	91	0	12	2.000	316SS								
C6A-1	Transient performance of defected or sodium bonded fuel; relative performance limits and failure mechanisms of pellet and powder fuel under transient irradiation conditions. Sodium loaded 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								GEAP-13722
C6A-2		0.2203	24	powder	80	0	10.0	2.000	316SS								
C6A-3		0.212	24	86	80	0	10.0	2.000	316SS								
C6B-1		0.212	24	86	80	0	10.0	2.000	316SS								
C6B-2		0.2203	24	powder	80	0	10.0	2.000	316SS								
C6B-3		0.212	24	86	80	0	10.0	2.000	316SS								

P Planned
I Initiated
C Completed
NP Not Planned

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

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Model Development

Task C-1 participated in a joint meeting with ANL/HEDL/RDT on comparative fuel modeling. The main areas of discussion involved:

1. Comparative fuel-modeling analyses of the upcoming ANL-H3 tests
2. Presentation of the revised GE-proposed format for exchange of experimental data
3. The effect of non-prototypic testing conditions on test planning and interpretation

In addition, there was a discussion of HEDL's proposed future testing program requirements for FFTF FSAR submittal (Volume III of the FFTF Safety Testing Program).

Detailed two-dimensional ($r-\theta$) heat transfer calculations have been completed for the ANL S-6 test. The preliminary results differ from those published by ANL. Work will continue under PA-XX (Safety Engineering) in an effort to understand the apparent differences.

Administrative

Task C-1 has prepared the draft work plans for the new Safety Engineering Project (PA-XX). The plans are currently in internal review.

As a result of AEC-RDT redirection of the safety testing program, concentrating all fast-reactor safety tests at national laboratories, the C-1 transient overpower task has been deleted from the PA-10 program effective July 1, 1972.

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

The final report for the Mockup 1A test (GEAP-13851, *Loss of Flow Experiment in an Electrically Heated Sodium Cooled Test Facility*) has been completed. Work on the C11A test has been terminated. The shutdown status of C11A is:

- Detailed capsule and fuel rod drawings have been issued.
- The final design review has been completed and minutes of the meeting issued.
- Draft copies of fabrication specifications have been completed.
- Several fuel rod parts (cladding tube, reflector, bottom end plug, and insulator pellets) have been completed.
- The two dimensional (THTD) model of the capsule for steady-state thermal analysis and the fuel rod stress analysis model (CREEP/PLAST) model have been developed.

Programming was completed for the free-standing version of the VOIDUM code with added options and the checkout is in progress. Work was initiated on a proposed procedure for coordinating national laboratory work in support of the Safety Engineering Base Program (PA-7X). Effort on Task C-2 funded activities terminated at the end of June in response to the PA-10 work scope for 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

Detailed plans for the destructive examination of capsule E2R have been submitted to Los Alamos Scientific Laboratory (LASL), where the work will be done. Center holes were observed in the 88% TD pellets after an exposure of 11 to 26 minutes. None developed in the 94% TD pellets despite a total exposure of 8 hours at a center temperature of $\sim 2500^{\circ}\text{C}$.

Table 1-3

TASK C-2 EXPERIMENTAL TESTS

Test Number	Objective	Test Facility	Number of Rocs	o.d. Diameter (inches)	Heated Length (inches)	Design Parameters					Steady-State Test Conditions			Status
						Pellet	Smear	Fuel Composition		Cladding Material	Peak kW/ft	Sodium Temperature (°F)	Burnup MWd/Te	
								UO ₂	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	Test Deferred
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 1E	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

Cesium and tellurium caused intergranular attack when loaded into thermal gradient capsules at 550 to 1100°C with stoichiometric fuel. No attack occurred under similar conditions with hypostoichiometric fuel.

Cesium and tellurium in a ratio of 1 reacted with cladding in the presence of stoichiometric fuel after 100 hours at 850°C. After 1000 hours similar attack took place at ~750°C, indicating a time-temperature effect.

Carbon or nitrogen may actively participate in intergranular attack reactions when present with stoichiometric fuel. With hypostoichiometric fuel, or in the presence of cesium, no attack occurs.

Metallographic examination has shown that Type-316 stainless steel with a typical silicon concentration of 0.6% more effectively resists carbide or nitride diffusion from the fuel into the cladding than does the same alloy with low (0.17%) silicon.

Experiments exposing fuel to high temperature sodium indicate that the geometry of the opening can affect the reaction-induced swelling.

Quantitative metallographic studies were made on a solid pellet rod and an annular pellet rod. Radial porosity distributions indicated up to five distinct density regions.

1.5 TASK F—FAST FLUX TESTS AND SUPPORT

Preliminary results of the metallographic examination of fuel rod F9C-13, in progress at LASL, indicate that attack at the cladding inside surface was negligible. The maximum increase in diameter was 1 mil (0.4% $\Delta D/D$).

The fuel-cladding mechanical interaction behavior of cold-worked Type-316 stainless steel, solution-annealed Type-316 stainless steel, and solution-annealed Type-321 stainless steel clad fuel rods with a smeared density of ~90% of theoretical was compared. At a burnup of ~6 at.%, the maximum cladding diameter change of 0.54% in the cold-worked Type-316 stainless steel clad rod was entirely plastic deformation. At a similar burnup, the solution annealed Type-316 stainless steel and Type-321 stainless steel clad rods exhibited some cladding swelling and a smaller amount of plastic deformation.

The results of the gamma scanning of fuel rods F9C-13, F12P, F12Q, F4A, and F4E at LASL show that the cesium isotopes, Ru 103 and Rh 106 migrated axially, with concentrations observed in the inter-pellet gaps, at various locations along the fuel column, and at the top and bottom of the fuel column.

The F9E grid-spaced subassembly test was successfully assembled and flow tested at EBR-II, and began irradiation with Run 56.

Fabrication of the F9F grid-spaced channel is complete. Final inspection, parts fit-up, and the Quality Assurance audit of the fabrication are in progress. The water flow test of the mockup F9F grid-spaced subassembly is complete. This test provides the basis for sizing the orifice for the actual F9F subassembly.

Fabrication of the F11A replacement and spare fuel rods is complete. Final inspection and documentation is in progress.

Damage to fuel rod F8B-5 was discovered during reconstitution of the experiment at EBR-II. The rod cannot be reinserted in-pile and is being stored at EBR-II pending shipment to LASL for destructive examination.

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, and three are being fabricated.

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/t	Target Burnup		Average Calculated Burnup ^(c)		Peak Calculated Fluence ^(d) (10^{12} n/cm ²)		Status
										at. %	MWd/Te	at. %	MWd/Te	Total	E > 0.1 MeV	
F0	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

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Table 1-4 (Continued)

Group	Objective	Total Number of Rods in Group	Date In	Projected Date Out	Actual Date Out	Number of Rods			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
						in S/A	S/A	Position				at. %	MWd/Te	Total	E > 0.1 MeV	
F9A	High power, high burnup, unencapsulated fuel rod irradiations	37	2/20/69	2/10/73	5/18/70	37	X043	4D2	12.06–13.36	11.25	100000	5.57	50189	4.64	3.98	Reconstituted into X043A
			10/27/70		37	X043A	7D4	8.01–10.28	11.25	100000	8.23	74213	7.02	5.96	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	11.25	100000	8.28	74684	7.07	5.89	In reactor
F9C	High power, high burnup, unencapsulated fuel rod irradiations	37	4/ 3/69		7/16/70	37	X056	5C2	10.50–11.55	11.25	100000	5.67	51157	4.72	4.01	Reconstituted into X056A
			10/27/70		5/13/71	37	X056A	6B2	10.20–11.71	11.25	100000	6.68	60207	5.61	4.76	Reconstituted into X143 and X144
			2/ 2/72	4/ 7/72	19	X143	6A2	10.64–11.70	11.25	100000	7.05	63546	5.95	5.05	In reactor	
			2/ 2/72	3/28/73	19	X144	6B4	11.01–12.03	11.25	100000	7.18	64740	6.05	5.14	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	11.25	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									In reactor
F9F	High power, high burnup, unencapsulated fuel rod irradiations in a grid spacer subassembly	37				37	X155									Fabrication complete
F10A	Cladding performance at peak cladding temperatures	19	2/2/72	7/22/64		19	X121	5D3	11.38–14.34	11.25	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	5D4	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 were based on the fission rates provided in the "Guide for Irradiation Experiments in EBR-II, revision 4," February 1971, and were valid as of Run 55. Burnup values will be updated upon receipt of revision 5 of the guide.

(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.

1.6 TASK G-6—CORE DESIGN ANALYSIS

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

Final assembly of F9E was completed at ANL under surveillance of the design engineer. The assembly was placed in EBR-II core position 6C4 in preparation for Run 56.

A formal request for Approval in Principle (AIP) for the Fast Test Reactor (FTR) grid-spaced driver fuel assemblies was submitted to the AEC.

A preliminary draft of the System Design Description (SDD) for the FTR grid-spaced driver fuel assembly was completed.

The two 37-rod assemblies from the high-temperature sodium flow endurance tests are being prepared for cleaning and examination.

The 217-rod full-scale hydraulic test has been postponed as a result of grid dislodgements encountered in the final fabrication stage of the grid spacer assembly. These dislodgements have been investigated and a revised program has been formulated to avoid this problem in the future.

1.7 TASK G-10—SAFETY METHODS AND ANALYSIS

A test case of the TOFF-II pre-failure axial fuel motion model was computed using reference D-Plant design data. The results showed significant axial fuel relocation for the assumed conditions; axial motion is a sensitive function of plenum gas pressure and cladding failure threshold.

Final specifications were developed for coupling BUBBLE to FREADM. This included formulation of the BUB2 and BUB3 subroutines for computing film thickness and vapor pressure.

1.8 TASK G-12—ANL SAFETY CORE DESIGN GUIDANCE

The GE liaison engineer at ANL has continued to communicate recent advances in the ANL development programs and has established frequent contacts between staff of both organizations. A substantial amount of documents and computer material was received from ANL.

1.9 TASK L—CLADDING DEVELOPMENT

The four L-15 and the MT-4 materials capsules have accumulated a total fluence of 3.5×10^{22} and $9.1 \times 10^{22} \text{ n/cm}^2$, respectively, in EBR-II. The L-15 capsules were discharged and shipped to VNC for post-irradiation examination. The MT-4 will be included in the reconstituted S/A X148A for continued irradiation.

Nine hot L-16 capsules previously irradiated in S/A X098 and ten cold L-16 capsules were incorporated into S/A X098A for irradiation in Row 4 of EBR-II beginning with Run 56. The cold capsules contain Type-321H specimens and sodium-exposed specimens to provide a broader data base for the L-16 clad alloy swelling experiment.

The effects of irradiation-induced void formation on the elastic moduli of stainless steels have been evaluated using a dynamic resonance technique. Rings, approximately 0.25 inch high, were cut from solution-annealed Types-316 and -347 fuel rod claddings irradiated in the GE-F2 fuel rod test in EBR-II. These samples had volume increases of up to 10% as measured by the immersion density method. The results indicate that 10% swelling in Type-316 may reduce the room temperature Young's Modulus by more than 20%.

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

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	Fiat Tensile						10 ²² n/cm ² Total	10 ²² n/cm ² Total	>0.1 MeV	
L-2 ¹	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 ¹	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 08/15/68	03/06/68 12/24/68	4-XG05 3-X039	4C2 3C2	12,632 2,532	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 02/23/69	03/06/68 05/18/70	4-XG05 4-X050	4C2 4C2	12,632 11,706	7.5	9.0	7.5	Post exam in progress
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 04/23/69 10/02/69	01/02/69 06/26/69 09/10/70	2-X018 4-X018A 4-X018B	2B1 4E2 4E2	10,853 2,334 10,351	20.0	7.8	6.7	Post exam complete
L-15	P.I. Mech. Prop. Eval. of Type-304,-316,-321,-347 & Incoloy 800	4	None	288	08/15/68 08/24/70 10/20/71	12/24/68 06/10/71 01/00/73 ⁽¹⁾	3-X039 4-X080 6-X148	3C2 6E5 6B3	2,532 5,351 5,989	3.0	3.5	3.1	In reactor
MT-4 ⁽³⁾	P.I. Mech. Prop. Eval. of Incoloy-800, Inconel-625, & Hastelloy-X	1	None	48	12/13/65 08/24/70 10/20/71	06/26/69 06/10/71 01/00/73 ⁽¹⁾	4-XA08 6-X080 6-X148	4F2 6E5 6B3	19,280 5,351 5,989	20.0	9.1	7.9	In reactor
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.

Sections of solution-annealed AISI 321 fuel rod cladding irradiated to approximately $3 \times 10^{22} \text{n/cm}^2$ ($E > 0.1 \text{ MeV-GE}$) at 527, 595, and 627°C have been evaluated by transmission electron microscopy. Void sizes are similar to those previously observed in annealed AISI 316 but the void number densities are substantially lower in the Type-321 cladding.

1.10 TASK P—FUEL ELEMENT MODELING

The LIFE-2 fuel element modeling code is in production status on the H6000 system.

The EIH-26 fuel rod was modeled using the BEHAVE-2 code. Improvements were sought in ability to predict diametral increases by investigating various possible mechanisms. Stress and plastic strain hypotheses were analyzed and rejected. The hypothesis that swelling is enhanced by intragranular carbide precipitation provided a rationalization of some aspects of the data.

Gap conductance and axial length increase studies were performed using BEHAVE-2.

A plan was formulated for development of a structural design code capable of predicting fuel rod cladding failure.

1.11 QUALITY ASSURANCE

The PA-10 Plan was revised in response to RDT comments and is now being reviewed internally.

A presentation of scope and plans for FY-73 PA-10 activities was made to RDT.

A series of seminars on codes and standards was begun. A series on the functional classification system was also initiated.

An investigation of L-16 capsules was performed as a result of shipping problems.

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 purposes of Task B are to:

- Determine analytically and experimentally the consequences of continued reactor operation following fuel rod failure under normal or near normal LMFBR operating conditions
- Provide experimental and analytical data to permit selection of fuel rod design parameters and operating conditions which limit the consequences of fuel failure
- Provide experimental data on failure kinetics to permit modeling and prediction of fuel failure behavior for the projected range of LMFBR operating conditions
- 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 mixed-oxide fueled rod irradiated to approximately 25,000 MWd/Te, defected, and then operated for approximately 30 days after defection. The fuel will be in an environment of high-purity (<10 ppm O_2) flowing sodium. The pressure in the rod before, during, and following failure will be monitored by pressure transducers. A Fission Gas Analysis System (FGAS) is being developed for the capsule which will permit monitoring and sampling of the cover gas in the capsule plenum.

Design of the B9B capsule and fabrication of fuel for the capsule was completed. A drawing of the capsule design is shown in Figure 2-1.

Development testing of the B9B capsule Fission Gas Analysis System (FGAS) was initiated. A mockup of the plenum region of the capsule and of the FGAS was fabricated and is shown schematically in Figure 2-2. During the first test run, the mockup operated for 160 hours with gas flowing continuously through the 3/8-inch diameter sampling line before plugging occurred. Upon examination, it was found that the unused backup lines were also plugged. A modification of the baffles protecting the ends of the sampling lines was made and a second test run initiated. During the second test, the 3/8-inch sampling line plugged after 98-1/2 hours of continuous gas flow. Flow was then switched to the 1/2-inch line. The 1/2-inch diameter sampling line plugged after 48-1/2 hours.

* Contributors are listed in Appendix B.

Figure 2-1. Drawing of Forced Convection Capsule B9B

1	101	FUEL PIN MODULE	289E655G1 (MOD. 1)
1	102	FLOWMETER MOD.	289E656G1 (MOD. 2)
1	103	SAFETY GAS MOD.	289E630G1 (MOD. 3)
1	104	JUNCTION BOX	195F125G1 (MOD. 4)
1	105	ACTUATOR MOD.	106D4394G1 (MOD. 5)
1	106	FLEX HOSE ASSY	106D4394G1
1	107	TUBE	289E660G4
2	108	GAS TUBE	289E660P8
1	109	OUTER SHROUD	289E660G1
4	110	RIB	163B5641P1
1	111	FLEX HOSE	(SEE BELOW)
1	112	FLANGE	289E660P18
2	113	TUBE	211A7820P8
1	114	BULKHEAD	289E660G2
1	115	TUBE	802X288 WALL 25 LG. 1" SOLID ST.
2	116	COUPLING	289E660P18
3	117	CLIP	SEE DETAIL
1	118	END PLUG	289E660P8
1	119	END CAP	P2
1	120	END CAP	P1
1	121	CAP	P4
1	122	END PLUG	289E660P10
1	123	TUBE	211A7881P1
1	124	COUPLING	289E660G3
1	125	LATCH PLATE	155C5679G2
1	126	CAPTURED BOLT	155B56412P17
1	127	LOCKWASHER	1/2" STN. STL.
2	128	WASHER	3/8-16 x 1 LG. STN. STL.
1	129	BULKHEAD	106D4394G2
3	130	ANGLE VALVE	153C4207P1
1	131	CAP	SWAGELOK N8 200-C-316
1	132	TUBE	802X288 WALL 11 LG. 25 LG. ST.
1	133	COUPLING	153C4210G1
1	134	COUPLING	MARMAN N8 507TH300

* FLEXONICS #0822-032-000 x 24" ± 1" LONG WITH 1/2" SCH. 40 WELD NIPPLE ENDS.

REFERENCE DWGS:
MASTER DWG. LIST 105X656

SECTION A-A

Figure 2-2. Schematic Diagram of FGAS Mockup Test

Work was completed on determining the rate of release of gas to be expected from defected fuel rods in conjunction with the B9B test. This was accomplished primarily on the basis of two sets of data. The first was the gas release data obtained from failed pin B3B. The second was a series of out-of-pile tests which were performed to provide data on the effects of fuel geometry and the size of the fuel-cladding gap on the release of gas from defected fuel rods. In the out-of-pile tests, fuel rods were simulated by dummy rods containing alumina pellets. Figure 2-3 illustrates the design of the dummy fuel rods. Figure 2-3 also illustrates the arrangement of the strain gauge pressure transducers which were mounted on the simulated fuel rods to measure the pressure drop along the rods. Tests were run by forcing pressurized helium into the top end of the rods and measuring the pressure drop at intervals along the rod and the gas flow rate through the rod. Table 2-1 summarizes the test parameters.

Table 2-1
SUMMARY OF PARAMETERS FOR GAS RELEASE TESTS

Test Number	Rod Number	Fuel Column Length L (in.)	Rod i.d. (in.)	Pellet o.d. (in.)	Diametral Gap D (in.)	Pellet Type
1	206	30	0.2201	0.2173	0.028	Solid Annular (0.05-inch Diameter Hole)
2	201	30	0.2204	0.2173	0.031	
3	209	30	0.2202	0.2173	0.029	
4	189	31.25	0.2201	0.2194	0.007	
5	154	30	0.2201	0.2173	0.028	
6	135	16	0.2201	0.2194	0.007	
7	154	30	0.2201	0.2173	0.028	
8	137	30.5	0.2201	0.2153	0.048	
9	163	30.5	0.2201	0.2153	0.048	
10	163	15	0.2201	0.2153	0.048	
11	154	15	0.2201	0.2173	0.048	
12	137	30	0.2201	0.2194	0.007	

Two different arrangements of the test apparatus were employed. In the first arrangement, which was employed for the first three tests, the flowmeter was upstream from the test rod and the gas stream exhausted through a hole in the side of the test rod. Figure 2-4 illustrates this arrangement schematically. Later, the flowmeter was moved downstream from the test rod to make it possible to go to higher system pressures. This second arrangement is illustrated in Figure 2-5 and was used for Tests 4 through 12.

A larger flowmeter was introduced into the system after the sixth test and was used for Tests 7 through 12. Figure 2-6 shows the test apparatus with the larger flowmeter installed.

Data collected from the first six tests are summarized in Table 2-2. Pressure readings at various distances from the top of the 30-inch long or, in the case of Test 6, 16-inch long, simulated fuel column are presented along with the corresponding mass flow rates of argon.

Analysis of the data has been separated into two parts:

1. Evaluation of the average friction factor over the length of the pellet column as a function of the Reynold's number
2. Evaluation of an exit flow coefficient as a function of Reynold's number

0.250 in. o.d. x 0.015 in. WALL TUBE
BALLIZED TO SIZE
LENGTH, ABOUT 60 in.

PHOTO SHOWING DETAILS OF BOTTOM END OF
TWO TYPES OF ROD AND PRESSURE TRANS-
DUCER, MOUNTING BLOCKS

TYPE 2

TYPE 1

Figure 2-3. Design Details of Simulated Fuel Rods

Figure 2-4. First Arrangement of Gas Release Test Apparatus

Figure 2-5. Second Arrangement of Gas Release Test Apparatus

Figure 2-6. Appearance of Gas Release Test Apparatus (Second Arrangement)

Table 2-2
SUMMARY OF RESULTS FROM GAS RELEASE TESTS

Test Number	Mass Flow Rate (gm/sec)	Pressure (psia)			
		At A*	At B*	At C*	At D*
1	$2.03(10^{-3})$	62	57		33
	$1.22(10^{-2})$	98	87		42
	$3.19(10^{-2})$	143	126		55
	$6.31(10^{-2})$	181	157		65
	0.110	226	195		82
	0.169	262	232		95
2	$8.19(10^{-4})$	39	37	24	24
	$3.69(10^{-3})$	63	54	41	28
	$1.90(10^{-2})$	110	88	67	36
	$5.31(10^{-2})$	161	123	93	46
	0.109	209	159	118	55
	0.169	256	193	148	67
	0.241	304	229	173	78
3	$5.54(10^{-4})$	41	37	24	24
	$3.07(10^{-3})$	66	54	28	28
	$1.92(10^{-2})$	114	89	61	38
	$5.25(10^{-2})$	166	127	90	49
	0.104	213	161	117	63
	0.162	261	196	143	80
	0.229	309	233	171	95
	0.253	327	246	193	103
4	$1.57(10^{-3})$	312	230		60
	$8.27(10^{-3})$	515	375		90
	$1.57(10^{-2})$	711	513		123
	$2.54(10^{-2})$	895	652		156
	$3.67(10^{-2})$	1090	789		186
	$4.80(10^{-2})$	1293	928		220
5	$3.86(10^{-3})$	64	45		27
	$1.76(10^{-2})$	114	82		38
	$4.30(10^{-2})$	163	118		50
6	$1.10(10^{-3})$		155	120	51
	$2.23(10^{-3})$		202	158	65
	$7.72(10^{-3})$		305	235	95
	$1.35(10^{-2})$		407	309	128
	$2.12(10^{-2})$		503	385	162
	$2.78(10^{-2})$		590	453	188
	$3.89(10^{-2})$		673	545	227
	$4.74(10^{-2})$		800	618	259

*A, B, C, and D refer to positions along test rod indicated in Figure 2-3.

The analysis was taken from Shapiro²⁻¹ for the case of adiabatic flow in a duct with friction. The essence of this analysis is summarized in the following paragraphs. The assumptions for the analysis are:

1. The gas acts as a perfect gas (i.e., $P = \rho RT$)
2. The flow process is adiabatic

Using these assumptions and applying the laws of conservation of mass, energy, and momentum, the following equations hold

$$\bar{f} = \frac{D}{4L} \left\{ \left[\frac{1 - M_1^2}{KM_1^2} - \frac{1 - M_2^2}{KM_2^2} \right] + \frac{K+1}{2K} \left[\ln \left(\frac{M_1^2 \left(1 + \frac{K-1}{2} M_2^2 \right)}{M_2^2 \left(1 + \frac{K-1}{2} M_1^2 \right)} \right) \right] \right\} \quad (2-1)$$

where

$$\begin{aligned} \bar{f} &= \text{average friction factor} = \frac{1}{L} \int_0^L f \, dx \\ D &= \text{hydraulic diameter} = \text{diametral gap} \\ L &= \text{length of pellet column from points 1 to 2} \\ \left. \begin{aligned} M_1 &= \text{Mach number at point 1} \\ M_2 &= \text{Mach number at point 2} \end{aligned} \right\} M &= \frac{v}{\sqrt{\frac{KRT}{M}}} \\ K &= \frac{C_p}{C_v} = 5/3 \\ R &= \text{universal gas constant} = 1545 \frac{\text{ft lbf}}{\text{lb-mole } ^\circ\text{R}} \\ M &= \text{molecular weight} = 39.948 \text{ lbm/lb-mole} \\ T &= \text{absolute temperature, } ^\circ\text{R} \\ v &= \text{local velocity, ft/sec} \\ f &= \text{local friction factor} = \frac{\tau_w}{\rho v^2 / 2} \\ \tau_w &= \text{wall shear stress, lbf/in}^2 \\ \rho &= \text{local gas density, lbm/ft}^3 \end{aligned}$$

$$P_0 = P \left(1 + \frac{K-1}{2} M^2 \right)^{\frac{K}{K-1}} \quad (2-2)$$

where

$$\begin{aligned} P &= \text{local pressure, psia} \\ M &= \text{local Mach number} \\ P_0 &= \text{local stagnation pressure, psia} \end{aligned}$$

$$T_0 = \text{constant} = 530^\circ \text{R} \quad (2-3)$$

where T_0 = local stagnation temperature

NOTE: This is a direct result of the assumption of adiabatic flow.

$$\frac{T_2}{T_1} = \frac{1 + \frac{K-1}{2} M_2^2}{1 + \frac{K-1}{2} M_1^2} \quad (2-4)$$

where

T_1 = absolute temperature at point 1, $^\circ \text{R}$

T_2 = absolute temperature at point 2, $^\circ \text{R}$

$$G = C \sqrt{\frac{KM}{R} \left(\frac{2}{K+1} \right)^{\frac{K+1}{K-1}} \frac{P_0}{\sqrt{T_0}}} \quad (2-5)$$

where

G = mass flux, $\text{lbm/ft}^2 - \text{sec}$

C = flow discharge coefficient

NOTE: This equation is based on choked flow.

The average friction factor as given by Equation 2-1 was evaluated by measuring the parameters D and L , and the variables of pressure and flowrate to evaluate M_1 and M_2 . The parameters D and L are given in Table 2-1 for the tests conducted. The flowrate and pressures are given in Table 2-3. Figure 2-7 shows the calculated values of f plotted against the Reynold's number in the gap. Note: $Re = \rho vD/\mu$, where (ρvD) is a constant for each run of a test and the viscosity, μ , is evaluated at the average temperature of the gas over the length of the pellet column. The values for f are reasonably approximated by the line, $f = 16/Re$ for $40 \leq Re \leq 1300$. The values of f evaluated for $Re < 40$ are larger than predicted by $16/Re$. However, data in this region are suspect because of the errors in measuring the flow and pressure at these low values. The pressure transducers are rated up to 1500 psia; however, at the low flows they were reading only 2 to 10% of full scale. The calibration curve for the flowmeter used was difficult to read at the low flows. While the flowmeter had a linear scale on the instrument the calibration curve shows two orders of magnitude change for approximately the first 20% of the scale and another order of magnitude for the remainder of the scale. Therefore the errors in reading the low flows were larger than for the higher flows. With the use of a second, larger flowmeter data have been collected for $Re > 1300$.

In Tests 1, 2, and 3 where a 1/16-inch diameter hole was used in a rod with an approximately 30-inch column of pellets, flow discharge coefficients were evaluated using Equation 2-5. These values as functions of the Reynold's number based on the hydraulic diameter of the pellet-cladding gap are shown in Figure 2-8.

Figure 2-7. Calculated Average Friction Factor as a Function of Reynolds Number

Figure 2-8. Flow Discharge Coefficient versus Reynolds Number for Tests 1, 2, and 3

Table 2-3
ANALYSIS OF FISSION GAS RELEASED TO THE B9D
CAPSULE PLENUM

Qualitative Sample Composition

Gas	Volume Percent
He	93.0
N ₂	5.5
O ₂	0.2
Xe	1.5
Kr	0.2
Ar	0.1
Acetone	0.04
CO ₂	0.02
Ne	0.002

Isotopic Abundance of Xenon and Krypton

Kr - 83	15.50 ± 0.55*
- 84	27.42 ± 0.78
- 85	7.77 ± 0.23
- 86	49.31 ± 0.79
Xe - 129	0.71 ± 0.04*
- 131	13.24 ± 0.08
- 132	18.23 ± 0.15
- 134	29.16 ± 0.16
- 136	38.66 ± 0.11

* Reproducibility of values calculated at 95% confidence limits.

The accuracy is ±2% of the reported value or the reproducibility, whichever is larger.

The flow coefficients for the diameter ratios shown were normalized to match the flow coefficient as given by Crane²⁻² at the highest Reynold's numbers for each test. The curve in Figure 2-8 shows the data have no apparent dependency on the diameter ratio. This is contrary to Crane's analysis where significant dependency on diameter ratio was obtained for ratios >0.4. Also the curve for C in Figure 2-8 rises smoothly from 0.006 to 0.7 in the range Re = 3 to Re = 1000 while Crane shows C going from 0.3 to 0.72 in the range Re = 4 to Re = 100, with a relatively flat plateau at C ~0.72 from Re = 100 to Re = 1000. (Note: this is for a diameter ratio = 0.5.)

The effect of $\bar{f}$ and C on the gas released from a defected rod is shown in Figures 2-9 and 2-10. Figure 2-9 shows the pressure distribution and flow for Test 2 (rod 201). This test had a 30-inch pellet column with an approximately 3.1 mil diametral gap and a 1/16-inch diameter orifice at the end. With the test open to atmospheric conditions all runs with exit stagnation pressures greater than approximately 30 psia would have choked flow. In this test all the runs except the two lowest pressure runs were choked.

Figure 2-9. Results from Test 2-Rod 201

Figure 2-10. Comparison of Actual Flow to the Idealized Isentropic Flow

Figure 2-10 shows a comparison of the highest pressure run for Test 2 and an idealized analysis of the same run. The idealized case was calculated for adiabatic, frictionless (isentropic) flow. The inlet pressure corresponds to the inlet pressure in the actual test. The exit pressure is given by

$$P = P_0 \left(\frac{2}{K+1} \right)^{\frac{K}{K-1}}$$

where

$$P_0 = 326 \text{ psia} = \text{stagnation pressure}$$

$$K = \frac{C_p}{C_v} = 5/3$$

therefore,

$$P = 156 \text{ psia.}$$

The idealized mass flowrate is given by Equation 2-5 and is 1.84 gm/sec. The stagnation pressure for the actual case can be calculated from Equation 2-2 and knowing the pressure and mach number just before the exit orifice. The pressure from Figure 2-10 is ~58 psia and the mach number was calculated as 0.12. Therefore, the stagnation pressure is ~58 psia. The large difference in stagnation pressure and discharge coefficient of 0.77 reduced the actual flow to approximately 14% of the idealized flow.

The effect of friction on the flow of gas from a failed LMFBR fuel rod will be to greatly restrict the mass flowrate of gases from a rod. For operating fuel rod failures the gas temperature will increase as it flows past the hot fuel. The increase in temperature of the gas causes the Reynold's number to decrease and the friction factor, f , to increase. In addition, the stagnation temperature will go up and cause the mass flow to decrease (see Equation 2-5). Raising the temperature of the gas will also cause the stagnation pressure to rise and the mass flowrate correspondingly to rise. The overall effect, however, will be that raising the temperature of the gas will further restrict the flow. For example, in the ideal case of frictionless flow with heating, a gas initially at 1000 psia and 1460°R and heated to 2145°R at the exiting choked condition has a stagnation pressure and temperature of 1112 psia and 2860°R, respectively. The effect of the heating on the mass flowrate compared to no-heating would be

$$\frac{(\text{Mass flow})_{\text{heated}}}{(\text{Mass flow})_{\text{adiabatic}}} = \frac{1112}{1000} \cdot \sqrt{\frac{1460}{2860}} = 0.795$$

Therefore, heating alone causes an approximately 20% decrease in flow, and increased friction will further decrease the flow.

The net result of the tests completed and analyses made is that the mass flowrate of gas escaping from a failed LMFBR fuel rod will be greatly affected by the loss of pressure due to friction and to a lesser degree by any temperature increase of the gas. The application of these results to the analysis of failed LMFBR fuel rods, however, requires additional testing to determine the effective flow area and flow path in a fuel rod after high burnup. One of the primary purposes of the B9B test is to provide these data.

Design and development work on capsule B9B was suspended on June 30, 1972. The design drawings and calculations and the fuel for the capsule have been archived so that work can be restarted if desired in the future. Work concerning capsule B9B will be reported under Task 6.

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 oxygen) high-temperature (1050 to 1100°F) sodium and operated for several days at typical peak power conditions for a Liquid Metal Fast Breeder Reactor (LMFBR).

The objectives of the experiment were to determine:

- The effect of fuel-sodium reaction on the cladding strain of defected fuel rods.
- The amount of fuel released from a failed fuel rod due to erosion or dissolution by the flowing sodium coolant.
- The amount and species of fission products released from the failed fuel to the sodium coolant.

The capsule design is illustrated in Figure 2-11. Details of the irradiation history and design of the fuel rods was presented in previous quarterly reports.^{2-3, 2-4} Post-irradiation examination of the capsule is currently nearing completion.

Sodium samples and hardware were removed from the capsule for examination and analysis. The status of these activities is summarized below.

2.2.2.1 Fuel Rod Examination

The two axially aligned fuel rods were removed from the capsule for examination. The sodium coolant was removed from the capsule section containing the fuel rods by melting in a specially designed argon atmosphere furnace. Final traces of sodium were removed from the fuel rod cladding surfaces with methyl alcohol and clean cotton swabs. Figure 2-12 shows the post-irradiation appearance of the fuel rods after cleaning. No evidence of gross cladding strain or the extrusion of fuel-sodium reaction products through the defects was observed.

Diameter measurements on the fuel rods were made at intervals of 1/4 inch along their length at four different orientations. The results of the measurements are plotted in Figure 2-13. The large diameter fluctuations at the center line of each defect are due to the defect mechanism that was welded over the hole in each fuel rod cladding. These diameter fluctuations are not due to any cladding strain resulting from the formation of fuel-sodium reaction products. Both fuel rods exhibited a small amount of uniform cladding strain in comparison to the pre-irradiation cladding diameter of 0.2509 ± 0.002 inch. The plenum region of each fuel rod maintained its pre-irradiation diameter while the fuel sections of both rods exhibited cladding strain up to 0.001 inch (0.4%). Cladding deformation was only observed in the 0- and 45-degree orientations on fuel rod B9D-1. This is the only apparent difference between the performance of the two fuel rods. B9D-1 was defected in the gas plenum above the fuel column; B9D-2 was defected at the midpoint of the 4.25-inch fuel column. Under the conditions of this experiment it appears that the defect location has relatively little effect on the performance of the fuel rod after failure.

2.2.2.2 Fission Gas Analysis

Fission gas released from the failed fuel rods to the capsule plenum was collected during the post-irradiation examination and analyzed. The qualitative sample composition and isotopic abundance of xenon and krypton are given in Table 2-3. The total quantity of fission gas found in the primary containment was approximately 1.5 cc at STP. This represents about 10% of the total fission gas yield during the irradiation.

The gas plenum of each fuel rod was punctured and the gas collected and analyzed. The results are shown in Table 2-4. As expected, no fission gas was found in the plenum of B9D-1. However, in B9D-2 a small quantity of xenon and krypton was found (~0.2 cc at STP). This quantity of gas represents less than 5% of the total generated in the fuel rod. From the results, it appears that the plenum gas could not freely communicate with the defect hole in the cladding. The normal storage procedure for the fuel

Figure 2-11. Capsule B9D

Figure 2-12. Post-Irradiation Photographs of B9D Fuel Rods

Figure 2-13. Post-Irradiation Diameter Measurements of B9D Fuel Rods

rods would facilitate removal of this fission gas. In order to protect any sodium-fuel reaction product from exposure to air, the fuel rods were stored in a stainless steel container. This container was evacuated to <100 microns and backfilled with 3 to 5 psig of argon gas each time the container was opened. The fuel pin experienced a minimum of 21 such cycles prior to puncturing the gas plenum. The presence of fission gas and helium and a lack of any appreciable quantity of argon indicates that the plenum was well sealed. The air contamination found in the gas sample can be attributed to a measured small leak in the gas sampling system. A mass spectrometer analysis of gas from the B9D-2 plenum is in progress.

Table 2-4
ANALYSIS OF FISSION GAS RETAINED
IN B9D FUEL ROD PLENUMS

Gas	B9D-1 Volume (%)	B9D-2 Volume (%)
N ₂	61.1	21.2
O ₂	16.5	5.7
Ar	4.8	3.4
Kr	ND	3.2
Xe	ND	27.8
He	0.13	37.6
H ₂	17.8	ND

ND = Not Detected

2.2.2.3 Fuel Rod Metallography

The fuel rods were sectioned and four metallographic sections are being prepared (two from each rod). Figure 2-12 shows the location of each cut and the appearance of the exposed fuel surfaces. Detailed metallographic procedures are being followed in order to preserve the condition of any sodium-fuel reaction products. Kerosene and xylene liquids used in grinding, polishing, and storage of the fuel samples have been dehydrated by additions of sodium metal. Exposure of the samples to the atmosphere has been minimized wherever practical. Examination of the first fuel sample indicates that these techniques have been successful in retaining the reaction product. Figure 2-14 shows a 28X composite picture of fuel rod section H. This section is representative of the fuel about 1-1/4 inches below the plenum defect in fuel rod B9D-1. Reaction product 0.003 to 0.005 inches thick was observed in the fuel-cladding gap. Formation of this reaction product is probably responsible for the 0.001-inch cladding strain observed. Metallographic examination of three more fuel samples is in progress. One or more of the samples will be selected for microprobe analysis of the reaction phase composition.

2.2.2.4 Fuel Burnup Analysis

Fuel rod sections B and G (Figure 2-12) were submitted to Analytical Chemistry for burnup analysis. Results of this analysis are not yet available. Based upon sodium calorimetry, flux wire data, and gamma scans²⁻⁴ of the fuel rods, the estimated burnups for B9D-1 and B9D-2 are 7800 and 6600 MWd/Te, respectively. Information on the fuel, cladding, and sodium temperatures immediately adjacent to each fuel rod defect are given in Table 2-5.

2.2.2.5 Sodium Analysis (Oxygen Purity)

A niobium wire (6 inches long by 0.010-inch diameter) was included in the capsule design as a means of measuring the equilibrium oxygen activity in the sodium coolant during irradiation. This wire was retrieved from the capsule, cleaned of sodium and smearable alpha contamination and shipped to Argonne National Laboratory for analysis. Results of this analysis are not yet available.

Figure 2-14. B9D Fuel Rod Metallography (Section H)

Table 2-5
B9D CAPSULE FUEL ROD AND SODIUM TEMPERATURES

	Localized Linear Power (kW/ft)	Fuel Center Temperature (°F)	Fuel Surface Temperature (°F)	Clad i.d. Temperature (°F)	Bulk Sodium Temperature (°F)
B9D-1 (Plenum Defect)	15.8	3650	2300	1200	1081
B9D-2 (Fuel Defect)	13.9	3420	2180	1160	1057

A 0.0018-inch thick piece of uranium metal foil was included in the capsule sodium in order to maintain the oxygen impurity level below 5 ppm during irradiation. Analysis of the foil is being conducted in order to measure the amount of oxygen that was gettered. A section of the foil has been metallographically prepared and is shown in Figure 2-15. The foil was wrapped around a stainless steel rod and held in place by a stainless steel screen during irradiation. The post-irradiation thickness of the foil/oxide is fairly uniform. Two or three separate phases appear to be present in the 500X photographs. Samples of the foil have been submitted for x-ray diffraction analysis in order to identify the different phases.

2.2.2.6 Sodium Analysis (Fission Products and Fuel)

Three sodium samples were sectioned from the capsule hardware and analyzed for fuel and fission products released from the fuel rods. The analysis of the fission-product release is not yet complete. Results on the amount of released fuel are listed in Table 2-6. The numbers given in the table represent estimates of the total quantity of fuel released from both rods to the capsule primary containment. The estimates include fuel in the sodium and any plated out on the wall of the stainless steel hardware. The relatively large quantities of U-238 are due to the presence of the uranium foil getter that was exposed to the sodium. Based upon the maximum estimates of released U-235 and plutonium, total fuel loss from the rods represents less than 0.03% of the total available fuel.

Reporting on capsule B9D will be continued under Task 6 in future progress reports.

Table 2-6
URANIUM/PLUTONIUM RELEASED FROM DEFECTED B9D FUEL RODS

Element	Sodium Sample ^(a)		
	B	G	L
U-235	0.0011	0.0035	0.0062
U-238 ^(b)	0.0059	0.0117	0.0507
Plutonium	0.0004	0.0022	0.0036

(a) Estimated number of grams released from the defected fuel rods to the sodium coolant.

(b) The source of the U-238 is almost exclusively due to the presence of the uranium foil getter.

2.3 SODIUM FUEL REACTION STUDIES (B11 SERIES)

This series of tests, designated the B11 Series, are to determine the diameter increase of defected mixed-oxide fuel rods as a function of time after defecting. Parameters to be investigated in the series are total oxygen availability (i.e., initial O/M of the fuel and the quantity of fuel present), fuel burnup, operating power, and cladding temperature. Capsules B11A, B11B, and B11C, the first set of capsules of the test

Figure 2-15. Micrographs of Uranium Getter from B9D Capsule

series, are designed to investigate the effect of total oxygen availability on the rate of fuel rod diameter increase. The fuel rod and operating parameters for experiments B11A, B11B, and B11C were summarized in a previous report.²⁻³ The major variables between fuel rods are initial fuel O/M and fuel column length.

The Fabrication Specification and the capsule and fuel rod drawings were approved and issued during this report period. Figure 2-16 shows the capsule design for experiments B11A, B11B, and B11C. Each capsule contains a single fuel rod with a prefabricated notch in the cladding. Differential expansion of the fuel and cladding coupled with the small fuel-cladding gap is used to rupture the cladding in the notched area of each fuel rod at power.^{2-5, 2-6} Reaction between the fuel and sodium (primary containment thermal bonding fluid) may then occur.

The capsule design incorporates a depleted uranium getter in the gap between the fuel rod and the primary containment. The uranium will provide a sink (getter) for oxygen in each test to simulate the large bulk of low oxygen sodium available in an LMFBR. The getter assembly consists of a 0.001-inch thick uranium foil with spacer nubs welded to the surface of the foil. The spacer nubs keep the foil centered in the gap between the fuel rod and the primary containment. Qualification of the welds required to attach the spacer nubs to the foil was initiated and completed. Fabrication of the getter assemblies for the B11A, B11B, and B11C capsules was completed. The assemblies are presently going through inspection.

Fabrication of the fuel rods and capsules was initiated. Fabrication of hardware for each fuel rod was completed and the fuel loaded into the claddings. The final end plug welds were completed and final inspection of the rods is in progress. Machining of capsule hardware is approximately 85% complete.

Inner capsule detailed assembly instructions were completed and submitted to Engineering and Quality Assurance for review and approval. The outer capsule detailed assembly instructions are presently under review.

Design and fabrication of a neutrographic standard to be used with the B11 series capsules was initiated. The drawing and work guide for the neutrographic standard were completed and approved for issue and fabrication initiated. The neutrograph standard design duplicates the cross-sectional configuration of the B11 capsules.

Fuel pellets for the neutrograph standard were pressed from the same blend of power used for the fuel rods B11A, B11B, and B11C. Fabrication of the fuel rod hardware for the standard was completed and the fuel loaded into the cladding. The final end plug weld was completed and inspection of the rod is in progress. Machining of the capsule hardware was nearly completed.

This work will be reported under Task 6 in future progress reports.

2.4 RUN-BEYOND-FAILURE TESTS

The test plan for the Run-Beyond-Failure (RBF) testing program is undergoing review. The preliminary test plan was presented at a meeting on April 6, 1972, between RDT/HEDL/ANL-EAST/ANL-WEST/GE to discuss the interactions of the various organizations in conducting the RBF test program. The results of this meeting have been summarized and distributed to all parties concerned. At this meeting ANL discussed the need for changes in EBR-II Emergency Operating Procedure 2-9 (EP2-9) to allow higher radiation levels in the EBR-II cover gas system and reactor building before reactor shutdown is required. An amendment to change EP2-9 to set an equivalent 1 MPC* in the reactor building as the reactor shutdown limit has been prepared and sent by ANL to the AEC for approval. ANL has also responded to questions by the AEC on an earlier request by ANL for approval-in-principle of a diagnostic test to be conducted in EBR-II.

* MPC = Maximum Permissible Concentration—Equivalent 1 MPC for a mixture of radioactive gases as defined in 10CFR20.

B-11 TRAIL CABLE CAPSULE

Figure 2-16. Drawing of B11A, -B, and -C Capsule Design

This test would expose a short column of mixed-oxide fuel to the reactor sodium with the express purpose of gaining operational experience and data on the radiological behavior of fission-gas releases in EBR-II from a failed mixed-oxide fuel rod.

This work will be reported under Task 6 in future progress reports.

2.5 DATA ANALYSIS AND CORRELATION

Work continued on the development of an engineering correlation for calculations of changes in cladding geometry resulting from failure of a fuel rod and formation of fuel-sodium reaction product. The correlation is based on the following assumptions:

1. The available oxygen in the fuel is assumed to be much more mobile in the fuel than the sodium is in the reaction product. Therefore, oxygen will be relatively plentiful at the fuel-reaction product interface and the availability of the sodium will limit the rate of formation of the reaction product.
2. The diffusion coefficient for sodium in the reaction product is assumed to be constant. This is a simplifying assumption which can be changed when data become available.
3. The reaction takes place instantaneously whenever the proper combination of reactants are present.
4. The density of the reaction product is taken to be 5.57 gm/cc. This value was calculated based on the reaction product being Na_2MO_4 with a cubic cell structure having a lattice parameter of 480 Å.

Two simple geometric cases have been partially analyzed. The first is for a point defect in a plane. The instantaneous concentration distribution of sodium diffusing from the point source is given by the classical equation:

$$C = \frac{M}{8 (\pi Dt)^{3/2}} \cdot e^{-(x^2 + y^2 + z^2)/4Dt}$$

where

C = concentration of sodium

M = total quantity of sodium

D = diffusion coefficient of sodium through reaction product

t = time

x, y, z = position coordinates.

Figure 2-17(a) shows a sketch of the pattern of sodium diffusion from a point defect. The diffusion pattern for the flat plate case can be used to approximate the diffusion pattern from a point defect in a cylindrical element. This is shown in Figure 2-17(a) as the area within the dashed circle. For a given depth of sodium diffusion into the cylindrical element, the affected volume is defined in terms of the radius of the cylinder and the depth of sodium diffusion by

$$V = \frac{2\pi d^3}{3} - \frac{0.3934d^4}{R}$$

(a) SODIUM DIFFUSION PATTERN

(b) REACTED FUEL VOLUME

(c) VOLUME INCREASE DUE TO REACTION

Figure 2-17. Fuel Rod Swelling for Fuel Sodium Reaction from a Point Defect in a Plane

where

V = volume of fuel involved in reaction

d = depth of sodium diffusion

R = radius of cylindrical fuel pellet

Figure 2-17(b) shows a cross section of the affected volume. Assuming that the reaction results in a change in density of fuel at 9.96 gm/cc to reaction product at 5.57 gm/cc and that the swelling is accommodated by an increase in the fuel rod cross section [see Figure 2-17(c)] the new radial dimension is obtained by increasing the original dimension by a factor of 1.2.

The second case considers a slit defect in a plane. The sodium concentration distribution for a slit of width $2h$ and of infinite length is given by

$$C = \frac{C_0}{4\pi Dt} \left(e^{-r^2/4Dt} \right) \left[\operatorname{erf} \left(\frac{x+h}{2\sqrt{Dt}} \right) + \operatorname{erf} \left(\frac{h-x}{2\sqrt{Dt}} \right) \right]$$

where

C = concentration of sodium

C_0 = source strength/unit area

Dt = diffusion coefficient X time

h = slit half-width

x = horizontal width dimension

r = radial depth dimension

The instantaneous sodium concentration for $C_0 = 1$ and a slit of width 0.2 cm is shown in Figures 2-18 and 2-19. The change in geometry of the fuel rod cross section due to the volume change of the fuel/sodium reaction product would be similar to that shown in Figure 2-17(c).

Efforts are continuing to analyze the case for a line defect in a cylindrical geometry. For a continuous line source parallel to the vertical axis through the point $(r', 0)$ on a cylinder of radius a with $c = 0$ at $t = 0$, the concentration is given by²⁻⁷

$$C = \frac{Q}{2\pi K} \left\{ \frac{2kt}{a^2} + \frac{r^2 + r'^2}{2a^2} + \ln \left(\frac{a}{r} \right) - \frac{3}{4} + \sum_{n=1}^{\infty} \frac{r^n (r'^{2n} + a^{2n})}{na^2 n r'^n} \cos(n\theta) \right\}$$

$$- \frac{Q}{\pi K} \sum_{n=0}^{\infty} \epsilon_n \cos(n\theta) \sum_{m=1}^{\infty} e^{-K\alpha_{n,m}^2 t} \frac{J_n(r\alpha_{n,m}) J_n(r'\alpha_{n,m})}{(a^2 \alpha_{n,m}^2 - n^2) J_n^2(a\alpha_{n,m})}$$

Figure 2-18. Instantaneous Concentration of Sodium Diffusing through an Infinite Slit in a Plane

Figure 2-19. Instantaneous Concentration of Sodium Diffusing through an Infinite Slit in a Plane

where

$\alpha_{n,m}$	=	positive roots of $J_n(a\alpha) = 0$
ϵ_n	=	1 for $n = 0$ 2 for $n = 1, 2, \dots$
Q	=	source strength/unit length/unit time
K	=	diffusion coefficient of sodium in reaction product
κ	=	diffusivity of sodium
a	=	radius of cylinder
t	=	time
r	=	radius
r'	=	radius of source
c	=	concentration

Progress on this topic will be reported under Task 2 in future progress reports.

REFERENCES

- 2-1. Shapiro, A. H., *The Dynamics and Thermodynamics of Compressible Fluid Flow*, Parts I and II from Volume I, Ronald Press Company, 1958.
- 2-2. *Flow of Fluids Through Valves, Fittings, and Pipe*, Crane Co., Technical Paper No. 410, 1957.
- 2-3. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Forty-First Quarterly Report*, November 1971—January 1972, February 1972 (GEAP-10028-41).
- 2-4. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Forty-Second Quarterly Report*, February—April 1972, May 1972 (GEAP-10028-42).
- 2-5. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-Sixth Quarterly Report*, February—April 1968, May 1968 (GEAP-10028-26).
- 2-6. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Thirty-Fifth Quarterly Report*, May—July 1970, August 1970 (GEAP-10028-35).
- 2-7. Carslaw, H. S., and Jaeger, J. C., *Conduction of Heat in Solids*, 2nd ed., 1959.

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 blanket lengths and the effect of flowing sodium on both failure thresholds and postfailure events.

3.2 HIGH-BURNUP FUEL EXPERIMENTS

At the direction of AEC-RDT, further work on the partially completed capsule C4M was terminated on June 30, 1972. The present status for the C4M capsule is indicated below.

- Machining of all capsule component parts is complete.
- All quality control and detailed assembly instruction documentations have been completed and approved for open-shop (nonremote) assembly.
- Assembly of capsule components has been initiated.
- Additional effort is required before obtaining approved brazing procedures for mounting the 20-mil diameter thermocouples into the capsule bulkhead.
- All capsule components and control documents are presently stored in a completely traceable form in bonded storage at the Vallecitos Nuclear Center.
- The two HEDL fuel rods (PNL-2-2 and 2-7), intended for use in capsule C4M and C4N (also affected by the AEC-RDT direction), have not been modified to the C4M/N configuration, but remain in a radioactive storage area of Vallecitos Nuclear Center in the as-received condition.

3.3 COMPARATIVE TESTS

3.3.1 Configuration of C6 Flux Filters

Nuclear analysis of the C6 comparative test assembly (three independently encapsulated fuel rods), modified to allow transient irradiation of EBR-II irradiated fuel specimens, has been completed.

A thermal neutron filter (B.C) was designed to minimize interaction between rods. "Power-shaping" collars were added to minimize end peaking in the fuel rods and approximate the axial power distribution experienced by the fuel rods in EBR-II. The capsule and filter configuration is shown in Figures 3-1 and 3-2. An alternate design involving a double-walled Zircaloy-2 heat sink (similar to the SEFOR TOP-1A capsule stainless-steel heat sink²⁻¹) had virtually identical nuclear characteristics. The assumed fuel and cladding specifications are given in Table 3-1. Although the design was accomplished for a 0.250-inch diameter fuel rod, the basic configuration would be applicable to 0.230-inch diameter rods.

* Contributors are listed in Appendix B.

Table 3-1
ASSUMED FUEL AND CLADDING SPECIFICATIONS

Fuel	Material, UO_2 - 25% PuO_2 (fully enriched in U^{235}) Form, pellet, 0.215-inch o.d. 0.250-inch length (nominal) Smeared density, 88% Theoretical Density Column length, 13.5 in.
Blanket Insulator	Material, Natural or Depleted UO_2 Form, pellet 0.215-inch o.d. 0.250-inch length (nominal) Pellet density, 93% Theoretical Density Column length, 4.875/0.375 in.
Cladding	Material, Type-316 SS, 0.250-in. o.d. X 0.015 inch wall

Figure 3-1. C6 Capsule Assembly Cross Section

REGION	COMPOSITION	HEATING RATE (W/cc-W)			TOTAL HEATING RATE (W/cc-W)	% TOTAL HEATING RATE (INSULATED REGION)
		NEUTRON	GAMMA	TOTAL		
1	FUEL PIN	5.201-5*	2.666-7	5.228-5	6.411-6	0.9042
2	INNER CAPSULE	5.409-9	1.086-8	1.627-8	5.017-8	0.0071
3	INSULATION	1.802-9	4.145-10	2.217-9	5.113-9	0.0007
4	SS + B ₄ C	1.352-6	3.878-8	1.391-6	6.242-7	0.0880
5	INSULATION	1.724-9	7.941-10	2.518-9	2.995-9	1.0000
6	CAP + VOID	3.114-9	6.369-8	6.680-8	3.001-8	
7	GRAPHITE	1.699-8	2.395-9	1.939-8	1.683-8	
8	PRESSURE VESSEL	3.829-9	1.742-7	1.780-7	8.978-7	

*5.201-5 $\equiv$ 5.201 $\times 10^{-5}$

Figure 3-2. Regional Heating in C6 Capsule Assembly (8 mil B₄C filter)
(This is a sector of Figure 3-1)

3.3.2 Details of Flux Filter Analysis

To begin the analysis, thermal neutron cross sections were generated using the computer code GETHRM³⁻² on a one-dimensional radial cell model of the capsule and surrounding TREAT elements. These cross sections were used in conjunction with the 24 neutron group library (with gamma production cross sections) from the nuclear analysis of the C4X, C4G and C4H capsules.³⁻³ The computer code SN1D,³⁻⁴ in its diffusion theory mode, utilized this library to compute the amount of TREAT control required to give $k_{\text{eff}} = 1.0$ for a one-dimensional problem. Another SN1D, in the diffusion theory mode, was used to search for the radius of a TREAT cell surrounding the capsule which produces the same k_{eff} .

Using this cell radius, a two-dimensional (r, θ) model of the capsule and associated TREAT elements was specified. The computer code SN2D,³⁻⁵ first in its diffusion theory mode and then in S_4 , was used to compute the neutron flux, gamma sources (in eight gamma groups), and neutron heating distributions for this model. The same data were computed for an 8-mil thick B.C filter. Using the eight-group gamma cross section library, the gamma flux and heating distributions for the 8-mil B.D filter were computed using SN2D in its diffusion theory mode. The results of these computations are shown in Figures 3-2 and 3-3.

To minimize end peaking in the axial fission distribution in the rods and to approximately reproduce, in TREAT, the axial fission distribution history of the rods in EBR-II, the capsule was analyzed in (r, Z) geometry with SN2D diffusion theory. The r, θ neutron and gamma flux distributions were used to obtain homogenized cross sections for all the material inside the B.C filter. Using trial and error methods, several SN2D (r, Z) neutron flux computations were made to determine the amount and placing of the 8-mil and 16-mil B.C filters. The best resulting configuration is shown in Figure 3-4. The axial fission and heating distributions computed are given in Figure 3-5 and Table 3-2. There is some discontinuity in these axial profiles at the interfaces between the 8-mil and 16-mil filter sections. Part of this is due to the different thermal neutron group cross sections in the inner capsule for the 8-mil and 16-mil sections and, hence, is not real since a smooth variation of the cross sections is expected at the interface. However, part of the discontinuity appears to be real and is produced by a difference in the "blackness" of the flux shadow produced by the two filter thicknesses. Significant alleviation of this discontinuity could probably be obtained through use of stepped collars of B.C filters similar to that of the final ANL H-3 Mark-II Loop filter configuration (see Subsection 3.4).

3.4 MODEL DEVELOPMENT

3.4.1 H-3 Round-Robin Analysis

3.4.1.1 Introduction

The General Electric Breeder Reactor Department (GE) participated in round-robin analysis of an upcoming ANL Mark-II integral sodium loop test designated H-3. This loop was flux filtered, and contained a HEDL-supplied central fuel rod that had previously been irradiated in EBR-II to ~3.5% atom burnup, and six surrounding fresh fuel rods of identical makeup (Figures 3-6 and 3-7).

The objective of the H-3 test is to raise the peak temperature of the central, pre-irradiated fuel rod to solidus temperature (~5040°F) during a mild transient overpower (~1\$/sec FFTF ramp) from a simulated steady-state temperature profile typical of FFTF.^{3-6, 3-7, 3-8} The test objectives are intended to be reached through use of the TREAT reactor power-shaping capabilities combining an initial flat-top power terminated by a power pulse (Figure 3-8).

General Electric was directed by AEC-RDT, in late March 1972, to join ANL and HEDL in a round-robin analysis of the H-3 experiment. The objectives of this round-robin analysis were to predict the thermal history of the test section, in particular the central fuel rod, and to predict the resulting cladding deformation and failure probability of this central fuel rod. The following report details the GE analyses leading to these objectives.

Table 3-2
C6C, C6D AXIAL PROFILES IN INNER CAPSULE

	Height (in.)	Height (cm)	Fissions/cc/sec TREAT Watt	Neutron Heating Watts/cc/Watt	Gamma Heating Watts/cc/Watt	Total Heating Watts/cc/Watt
Insulator	-0.375	-0.9525	1.186-8 ^(a)	1.467-8	8.953-9	2.363-8
	0.0	0.0		1.075-6	2.102-8	1.096-6
Fuel Pin	0.9	2.286	1.177-6	1.094-6	2.224-8	1.116-6
	1.8	4.572	1.197-6	1.094-6	2.224-8	1.116-6
	2.7	6.858	1.234-6	1.128-6	2.306-8	1.151-6
	3.6	9.144	1.400-6	1.279-6	2.506-8	1.304-6
	4.5	11.430	1.456-6	1.330-6	2.593-8	1.356-6
	5.4	13.716	1.488-6	1.359-6	2.645-8	1.386-6
	6.3	16.002	1.504-6	1.374-6	2.672-8	1.400-6
	7.2	18.288	1.508-6	1.378-6	2.680-8	1.405-6
	8.1	20.574	1.504-6	1.374-6	2.673-8	1.401-6
	9.0	22.860	1.491-6	1.362-6	2.650-8	1.388-6
	9.9	25.146	1.467-6	1.340-6	2.609-8	1.366-6
	10.8	27.432	1.427-6	1.304-6	2.543-8	1.329-6
	11.7	29.718	1.364-6	1.246-6	2.441-8	1.271-6
	12.6	32.004	1.195-6	1.092-6	2.215-8	1.114-6
	13.5	34.290	1.154-6	1.055-6	2.052-8	1.075-6
Blanket Insulator	15.125	38.4175	1.121-8	1.383-8	7.814-9	2.164-8
	16.75	42.545	1.069-8	1.315-8	7.401-9	2.055-8
	18.375	46.6725	1.032-8	1.263-8	7.332-9	1.997-8

(a) 1.186-8 $\equiv 1.186 \times 10^{-8}$ etc.

Figure 3-3. C6 Capsule Assembly—Relative Total Heating in Fuel Rod (8 mil B₄C Filter)

3.4.1.2 Results

The ANL Mark-II loop planned for use in the H-3 test contains a seven-rod array with six fresh fuel rods (25% PuO₂ and 75% UO₂, 65% U-235 enrichment) surrounding a PNL-17 fuel rod of the same composition but previously irradiated in EBR-II. The test section and loop are filtered by a layer of B₄Si, with local collars of B₄Si located near both ends of the fuel column to suppress end-flux peaking.

The technique of the physics analysis of the H-3 test was very similar to that described for the C6 capsule assembly flux filter except that due to time restrictions, only a 1D analysis of the H-3 test was performed vs the 2-D ($r-\theta$) analysis of C6. The physics model of the test section internals was taken directly from a smeared annular model based upon the H-3 test geometry (Figures 3-6 and 3-7), and is shown in Figure 3-9. Six radial mesh points were input for both the central fuel rod and the outer annuli of fuel (smeared representation of the outer six fuel rods) to obtain detailed knowledge of the radial power distribution in the rods. The missing TREAT fuel assembly was simulated by an annular ring of void surrounding an annular ring representing the test pump and seven TREAT fuel assemblies surrounding the test section were homogenized into the next annular ring. Beyond this annulus the TREAT reactor and reflectors were also calculated as annular rings. The Hodoscope was not considered in the calculation of heating rates in the capsule.

Computed power generation rates for the central and outer fuel rods are shown in Figures 3-10 and 3-11, respectively. The circumferential power variation for the outer fuel rods was interpolated from the calculations across the corresponding smeared annulus of fuel. The interpolated data (Figure 3-11) was then checked to assure that the results yielded the same total energy generation potential for the six independent fuel rods as calculated for the smeared annulus.

Figure 3-4. C6 Capsule Assembly-Axial B&C Filter Design

Figure 3-5. C6 Capsule Assembly-Altered Axial Flux Profile

Figure 3-6. ANL H-3 Test-MK II Loop Cross Section

Figure 3-7. Radial Thermal-Hydraulic Model of Test Section

Figure 3-8. Treat Power History

Figure 3-9. Homogenized 7-Rod Bundle through Mid-Plane of Test Section

Figure 3-10. Center Rod Radial Power Distribution

Figure 3-11. Outer Rod Radial Power Distribution

Gamma and nonfission neutronic heating of all nonfuel components of the test section (i.e., cladding, sodium, containment, etc.) were also calculated. The total distribution of radial heat generation rates for the smeared test section geometry (Figure 3-9) is given in Table 3-3. These values should be reduced by 0.55 to normalize the calculated generation rates to the values used in the ANL contribution to the H-3 round-robin analysis (e.g., central rod radial average power of 1.34 (watts/gm)/TREAT MW at the core midplane.³⁻⁶ The size of this reduction (0.55 times the physics prediction) is consistent with the semi-empirical power factor change ANL has described as necessary to account for core size change between the measured results of the H-3 calibration test and the planned H-3 test core size (core loading 639). ANL was required to multiply their measured calibration factor by ~0.66 (versus the GE 0.55) to account for core size change.^{3-9, 3-10}

It should be noted that the GE physics calculations match very closely the power split between the central and outer fuel rods experimentally measured during calibration runs applicable to the H-3 test as reported in the H-3 safety analysis.³⁻¹¹

As an additional indication of calculational consistency, the independent GE physics calculations resulted in radial peak-to-average power ratios for the central rod and smeared outer rod annulus of 1.16 and 1.49, respectively, which compares to the respective values of 1.14 and 1.50 from the ANL physics calculations.³⁻⁶

The thermal-hydraulic calculations were performed using the THTD heat transfer code. The basic radial thermal-hydraulic model used in the code is shown in Figure 3-7 and the test section geometry is presented in Figure 3-12. The axial section heated by the fuel column was divided into four equal parts (Figure 3-13). The fuel structure in each rod was divided into ten equal volume nodes as shown in Figure 3-14 and three cladding nodes, at the inner surface, the root mean radius and the outer surface. The relative power distribution

Table 3-3
CALCULATED HEATING RATES BY COMPONENTS
(H-3 TEST)

Region	Mid-Radii (cm)	Neutron (Watts/cc) TREAT Watt	Gamma (Watts/cc) TREAT Watt	Total (Watts/cc) TREAT Watt	Total Watt/cm TREAT Watt	Percent by Zone
Central Fuel Pin	2.02 E-05 ^(a)	2.02 E-05	7.75 E-07	2.08 E-05	0.479716 X 10 ⁻⁵	9.79
	1.09 E-01	2.08 E-05	7.78 E-07	2.16 E-05		
	1.59 E-01	2.17 E-05	7.80 E-07	2.25 E-05		
	1.96 E-01	2.29 E-05	7.82 E-07	2.37 E-05		
	2.26 E-01	2.49 E-05	7.84 E-07	2.57 E-05		
	2.47 E-01	2.76 E-05	7.89 E-07	2.84 E-05		
Clad	2.73 E-01	9.56 E-09	2.36 E-07	2.46 E-07	1.60947 X 10 ⁻⁸	0.03
Sodium	3.20 E-01	4.43 E-09	2.38 E-08	2.83 E-08	1.608929 X 10 ⁻⁸	0.03
	3.76 E-01	4.38 E-09	2.36 E-08	2.79 E-08		
	4.32 E-01	4.39 E-09	2.36 E-08	2.80 E-08		
	4.88 E-01	4.43 E-09	2.38 E-08	2.83 E-08		
Clad	5.45 E-01	9.57 E-09	2.36 E-07	2.46 E-07	4.93533 X 10 ⁻⁸	0.10
Outer Fuel Pin	5.91 E-01	2.36 E-05	7.68 E-07	2.44 E-05	4.16226 X 10 ⁻⁵	84.97
	6.22 E-01	2.41 E-05	7.60 E-07	2.49 E-05		
	6.60 E-01	2.55 E-05	7.67 E-07	2.63 E-05		
	7.11 E-01	2.79 E-05	7.75 E-07	2.87 E-05		
	7.62 E-01	3.28 E-05	7.82 E-07	3.33 E-05		
	8.16 E-01	5.17 E-05	7.79 E-07	5.24 E-05		
Clad	8.64 E-01	9.09 E-09	2.12 E-07	2.21 E-07	4.431076 X 10 ⁻⁸	0.09
Sodium	9.18 E-01	4.01 E-09	1.92 E-08	2.32 E-08	3.7887 X 10 ⁻⁸	0.08
	9.90 E-01	3.75 E-09	1.73 E-08	2.10 E-08		
	1.06 E-00	3.56 E-09	1.60 E-08	1.95 E-08		
	1.13 E-00	3.42 E-09	1.50 E-08	1.84 E-08		

(a) 2.02 E -05 = 2.02 X 10⁻⁵ etc.

Table 3-3 (Continued)

Region	Mid-Radii (cm)	Neutron (Watts/cc) TREAT Watt	Gamma (Watts/cc) TREAT Watt	Total (Watts/cc) TREAT Watt	Total Watt/cm TREAT Watt	Percent by Zone
Liner	1.20 E-00	6.73 E-09	1.44 E-07	1.50 E-07	8.57256 X 10 ⁻⁸	0.15
Holder	1.37 E-00	6.13 E-09	1.24 E-07	1.31 E-07	2.98513 X 10 ⁻⁷	0.61
Sodium	1.56 E-00	2.80 E-09	1.02 E-08	1.30 E-08	1.50595 X 10 ⁻⁸	0.03
Containment	1.73 E-00	5.21 E-09	9.86 E-08	1.03 E-07	9.98175 X 10 ⁻⁷	2.04
	1.96 E-00	4.81 E-09	8.56 E-08	9.04 E-08		
	2.19 E-00	4.49 E-09	7.43 E-08	7.87 E-08		
	2.42 E-00	4.23 E-09	6.29 E-08	6.72 E-08		
B ₆ Si Filter	2.56 E-00	1.22 E-06	1.36 E-09	1.22 E-06	1.006386 X 10 ⁻⁶	2.05
Total					4.898735 X 10 ⁻⁵	99.97 (94.76 in fuel)

Figure 3-12. Radial Thermal-Hydraulic Model Geometry

Figure 3-13. Axial Division of Test Section

- A. 10 EQUAL VOLUME FUEL NODES PER CIRCUMFERENTIAL SLICE
 B. CLADDING NODES (3)
 1. INNER SURFACE
 2. ROOT MEAN RADIUS
 3. OUTER SURFACE

Figure 3-14. Detailed Fuel Nodal Structure

in each fuel volume sector was based on Figures 3-10 and 3-11. The experimentally measured axial power distribution^{3-6, 3-12} in Figure 3-15 was the basis for the relative power for the four axial fuel sections in Figure 3-13. The nonfission power generation rates (primarily gamma heating) for the test section were based on Table 3-3.

The following parameters and boundary conditions were used in these analyses:

- Fuel-cladding interface coefficient— $h = 2000 \text{ Btu/h-ft}^2\text{-F}^\circ$
- Sodium convective film coefficient $20,000 \text{ Btu/h-ft}^2\text{-F}^\circ$
- Sodium mass flowrate³⁻⁷ $2.45 \times 10^6 \text{ lb}_m/\text{h-ft}^2$
- Coolant inlet temperature³⁻⁷

Time (sec)	°F
0	700
8	790
99	790

- Fuel melting temperature (T_m) = 5040°F
- Fuel heat of fusion = 130 Btu/lb
- Fuel density $605 \text{ lb}_m/\text{ft}^3$ (88% TD smeared)

The results of the THTD analysis are presented in Figures 3-16, 3-17 and 3-18. The radial temperature distribution shown in Figure 3-16 was based on calculations at axial node 2 which is 1.69 inches above the midplane of the TREAT core. This is representative of the peak power conditions in the center rod and 0.97 peak power in the peripheral rods (see Figure 3-15). A maximum fuel center temperature of 4620°F (versus 5040°F solidus temperature) is predicted for the center rod at 5.5 seconds into the transient. The GE analysis time base is identical to HEDL analysis,³⁻⁷ and two seconds less than the ANL time base.³⁻⁸ The outer rod maximum fuel temperature of 5560°F (molten) occurred after 5.7 seconds, and resulted in a maximum fuel melt volume (core midplane) of ~25%. The area of melt is not centered on the outer rods but is skewed radially outward away from the central pin (Figure 3-16). A maximum cladding temperature of 1340°F is predicted at the top of the central fuel rod as indicated in Figure 3-17. The thermal response of the cladding and sodium is shown in Figure 3-18 and a temperature maximum is indicated at 5.0 seconds into the transient.

A transient cladding deformation analysis was performed for the central rod using the previously presented thermal analysis for the transient temperatures of the fuel and cladding. This analysis utilized BEHAVE-2, a fuel performance code that calculates the spatial position of cladding, fuel, and interior voidage in an operating fuel rod with unfailed cladding.³⁻¹³ The code is two-dimensional (r,z) with axial power symmetry assumed throughout the program. Fuel and cladding deformation modes include elastic, plastic, and creep processes. Stress, strain, displacement, and temperature fields are also calculated. The influential processes that are simulated include orthotropic cracking of fuel; dilatation of fuel and cladding due to irradiation effects and thermal expansion; fuel densification by both stress and diffusion-controlled processes; transition from axial slippage to locking at the fuel-cladding interface and axial flow of fuel.

The program incorporates these unusual features:

- a. Axially interactive effects are calculated.
- b. Cracks are represented as a continuum, producing zero stress normal to the crack plane. Cracking occurs under any positive stress normal to two possible directions of cracking [(r,θ) and (r,z) planes].
- c. Rapid closure of the fuel-cladding gap is assumed, permitted by immediate cracking of the fuel at the first application of power.
- d. Fuel is divided into two phenomenological zones, the inner zone being made up of two regions. Boundaries between the zones and regions are defined by time-temperature relations and move in space as a function of time. Crack closure is assumed in the inner zone.
- e. The nodal grid is redefined in the radial direction every time step by averaging conditions from old grid. A fine grid is imposed in regions of rapidly changing properties (10 nodes per region).
- f. Imposition of a shear element at the fuel-cladding interface provides a transition from axial slippage to locking for each axial section by continually adjusting its viscosity in response to changes in fuel-cladding normal force.
- g. An interior inviscid fuel zone interacts with the outer, structural fuel zone as a constant pressure boundary condition. Within the inviscid zone axial transfer of fuel by a flow process occurs under certain conditions.
- h. Swelling of fuel and cladding and fission gas release are represented by empirically derived models.

The performance of the center fuel rod in the H-3 experiment was calculated using BEHAVE-2. Input information included the "as fabricated" condition of the fuel rod along with the EBR-II irradiation conditions to calculate the performance during steady-state operating conditions. Data from the THTD analysis were input into the BEHAVE-2 program to predict fuel rod performance during the transient power operation in TREAT.

Figure 3-15. Axial Power Distribution

Figure 3-16. Radial Temperature Distribution in Test Section

Figure 3-17. Axial Cladding Temperature Profiles of Center Rod During Transient

Figure 3-18. Time Dependence of the Cladding and Sodium Temperature at the Top of the Fuel Column

The microstructure at the end of steady-state operation in the EBR-II and after cooldown to room temperature was calculated using BEHAVE-2. As an example, the code predicts a core midplane structure with columnar grain growth extending to ~ 0.043 inch, and the corresponding radial redistribution of void forms a central void of radius ~ 0.013 inch. There are transverse cracks and radial cracks out to ~ 0.06 inch as a result of thermal contraction during the final cooldown. Void is contained in radial cracks from ~ 0.08 inch to the fuel cladding interface as a result of cracks formed during the initial rise to operating temperatures. This outer ring of fuel remains below the solid-state diffusion temperature, therefore, precluding crack healing during subsequent in-pile operation. The maximum porosity region is between ~ 0.045 and 0.075 inch, and is a consequence of the initial radial cracking and the transfer of voidage from the fuel-cladding gap to these cracks. The void volume is converted to porosity within the fuel during the crack healing process involving several mechanisms that are diffusion-rate controlled.

Radial cracks are generated in the fuel as a result of thermal contraction during the final shutdown of the fuel rod in the EBR-II. The closure of these cracks during the initial period of the overpower transient is shown in Figure 3-19. In the startup condition the radial cracks extend across the fuel; therefore, no interaction stress can be generated at the fuel-cladding interface. After ~ 4.7 seconds into the transient, thermal expansion has closed the shutdown cracks and a stress in the cladding is initiated as indicated in Figure 3-20. As the fuel continues to expand the elastic + plastic diametral strain increases as does the hoop stress in the cladding.

After about 5.1 seconds of the transient the outer portion of the fuel is thermally contracting while interior fuel continues to expand. The elastic outer fuel constrains the hot, plastic inner fuel which reduces the cladding stress as shown in Figure 3-20. However, at 5.15 seconds, the exterior zone of fuel, which is in hoop tension, cracks radially removing the constraint on the inner fuel and results in an increase in cladding stress. Finally at 5.3 seconds, a general cooling of the fuel starts a second reduction in the cladding stress. After 5.8 seconds of the transient, fuel cooldown was sufficient to terminate the fuel-cladding interaction. The 0.01% difference in the starting and ending $\Delta D/D$ represents the permanent set in the cladding resulting from this deformation during transient operation.

The small magnitude of transient-induced cladding deformation is strongly tied to the relative compliance (relative plasticity) of the fuel due to high fuel temperatures. The BEHAVE-2 calculations indicated a high degree of plasticity of the hot interior fuel, which had a significant impact on the calculated cladding strain. In comparison with a model not including fuel plasticity or cracking, the inclusion of fuel plasticity reduces the amount of cladding strain calculated while the inclusion of fuel cracking increases it.

The effective plastic cladding strain over the fuel length was calculated for the irradiation history of the fuel rod. The results at the end of the EBR-II test and the TREAT transient are plotted in Figure 3-21. A maximum effective plastic strain of 0.12% was predicted for the fuel rod, of which 0.05% occurred during the transient overpower operation phase. Cladding failure is not expected as a result of this low amount of deformation predicted by BEHAVE-2.

The measurable diametral changes in the cladding over the fuel length were calculated and the results at the end of two irradiation phases are plotted in Figure 3-22. A maximum diameter increase of 0.2% including irradiation-induced cladding swelling was predicted for the EBR-II portion of the irradiation. A diameter increase of only $\sim 0.01\%$ was predicted to occur during the overpower transient in TREAT. The fuel rod cladding is expected to maintain its integrity over these combined irradiation conditions. A comparison of the GE, ANL,³⁻⁶ and HEDL³⁻⁶ calculated cladding deformations appears in Table 3-4 as part of the summary and conclusions.

3.4.1.3 Summary and Conclusions

The results of these GE analyses indicate that the central fuel rod peak conditions will reach maximum fuel and cladding i.d. temperatures of 4620°F and 1340°F, respectively. During the transient, permanent cladding deformation on the order of $\sim 0.01\%$ is predicted and the central rod is not expected to fail.

Table 3-4
COMPARATIVE SUMMARY OF PEAK TRANSIENT CONDITIONS
CALCULATED FOR THE H-3 ROUND-ROBIN ANALYSIS

	GE Analysis	HEDL Analysis (3-7, 3-8)	ANL Analysis (3-6, 3-14)
Central Rod Peak Conditions			
Fuel Temperature ($^{\circ}\text{F}$)	4620 ^(a)	4830	5000 ^(a)
Cladding i.d. Temperature ($^{\circ}\text{F}$)	1340	~1650	1490
Outer Rod Peak Fuel Temperature at Same Time Step ($^{\circ}\text{F}$)	5525 ^(b)	—	~4720 ^(c)
Central Rod Maximum Strain			
Plastic + Elastic (ϵ_{θ})	0.0019 ^(d)	—	0.0080
Plastic Only (ϵ_{θ})	0.00013 ^(d)	0.0055	0.0039
Plastic Effective ($\bar{\epsilon}$)	0.00047	0.012	0.003

(a) Solidus temperature is $\sim 5040^{\circ}\text{F}$ for these analyses.

(b) Molten fuel - If the heat of fusion is interpreted as an equivalent temperature rise then the temperature difference is $\sim 1900^{\circ}\text{F}$ greater than the adjoining ANL calculation.

(c) Estimated from Figure 6 of Reference 3-6.

(d) Top of fuel column (all other strain values at midplane).

Figure 3-19. Closure of Radial Cracks in Fuel During Transient

Figure 3-20. Cladding Response During Transient

Figure 3-21. Effective Plastic Cladding Strain at Peak Radial Node (Not Including Irradiation Creep) versus Fuel Length

Figure 3-22. Pin Diameter Change Relative to Unirradiated Condition versus Fuel Length

The maximum temperature of the outer rods will significantly exceed the melting point of the mixed-oxide fuel, and a maximum of ~25% molten fuel volume is predicted at the core midplane. These fresh fuel rods were fabricated with ~88% theoretical smeared density and are not expected to fail during transient operation.

The basic results of the GE analysis and corresponding results from the HEDL and ANL round-robin analyses^{3-6, 3-7, 3-8} are summarized in Table 3-4 for comparison. All calculations were based on the same center rod peak power of 1.34 watts/gm-TREAT watt and the same TREAT transient power profile.

The peak central rod conditions predicted by GE are lower than that of HEDL and ANL as a consequence of the three-dimensional heat transfer calculations which provide for more prototypic radial heat flow conditions in the seven-rod assembly. This calculation also results in outer rod maximum fuel temperatures which are significantly higher than those predicted by ANL³⁻⁶ because the outer rods are treated as six individual sources as compared with an annular heat source of equivalent fuel volume around the central rod.

GE predicts the least deformation in the central rod during the transient overpower operation in TREAT. This is believed a consequence of several factors. The most important factor is that only BEHAVE-2 has accounted for the variation in fuel compliance (plasticity) with temperature,^{3-6, 3-8} which permits the utilization of internal voidage within the fuel. In addition, the BEHAVE-2 code provides for initial slip between the fuel and cladding and conversion to locking as a function of fuel-cladding interfacial pressure; and the lower predicted central pin temperatures reduce the amount of differential thermal expansion between the fuel and cladding.

3.4.2 ANL-S6 Thermal Analysis

The seven-pin ANL-S-6 transient overpower test has been analyzed with a two-dimensional THTE model which included a representation of the thermocouple and utilized neutron and gamma heat generation rates computed by BRD.

Details of the ANL S-6 experiment are presented in an ANL Monthly Progress Report.³⁻¹⁵ A cross section of the experimental configuration is depicted in Figure 3-23. Seven fuel rods surrounded with stagnant sodium were contained within a piston autoclave. Application of a TREAT power profile of 5374-MW peak and a half-width of about 0.2 second resulted in fragmentation of all the rods. Observed piston displacement and pressure pulses place the time of initial rod failure at 1.64 seconds. ANL³⁻¹⁶ has computed the temperature rise of stagnant sodium at the thermocouple location to be 340°F, using a concentric-ring, one-dimensional model with the ARGUS code.

The THTE thermal-analysis model of the S-6 consists of a 30-degree sector of the experimental cross section, including the primary autoclave modeled with 235 nodes as depicted in Figure 3-24. Ten equimass radial divisions were made in the central fuel pellet and five equimass radial divisions were made in the peripheral pellet. The cladding of the central and outer fuel rods were each divided into four equimass nodes. The calculation was done in three time segments in order to recognize a variable fuel-cladding gap conductance. Power peaking at the ends of the fuel columns (per Reference 3-15) was taken into account.

The results of the GE analysis indicate that, at time of cladding failure, peak fuel temperatures of both the center and peripheral rods are significantly different than the respective ANL analysis while predicting essentially the same sodium temperature rise.

Before definitive conclusions can be drawn about the differences in temperature predictions, it will be necessary to meet with ANL and discuss and compare all aspects of thermal modeling assumptions, such as physical properties and power distributions involved in the two models. But these results, combined with the above H-3 test analysis (Subsection 3.4.1), do indicate sufficient discrepancy to question the use of 1-D radial models to thermally analyze seven-rod tests with complex geometries such as this S-6 test and the previously described H-3 test.

3.4.3 Nonprototypic Testing Conditions

During the same meeting that the above H-3 round-robin analysis was presented, the effects of non-prototypic testing conditions were discussed. The GE discussion was followed up by an additional formal writeup³⁻¹⁸ as requested by RDT.

3.5 LIAISON

3.5.1 Experimental Data Exchange Format

As an additional part of the joint (GE/ANL/HEDL/RDT) meeting on comparative fuel modeling (May 18—19, 1972), Task C-1 presented the revised format for the exchange of experimental data.³⁻¹⁷ This format (previously reported in the PA-10 41st Quarterly Report), with comments incorporated from RDT, ANL, and HEDL, was accepted with minor modifications at the above meeting. The format will form the basis for standardizing the minimum requirements to be met in preparing data packages for transient experiments performed and reported in RDT-sponsored testing programs. These data packages will be sent to requesting organizations who also participate in RDT-sponsored transient test programs for use in developing a broader understanding and characterization of both the experiments and the analytical tools used to interpret them.

3.5.2 Comments on FFTF Safety Testing Program

The closing portion of the joint comparative fuel modeling meeting consisted of discussions of the latest version of the FFTF Safety Testing Program.³⁻¹⁸ The GE discussion was followed up by an additional formal writeup³⁻¹⁹ as requested by RDT. The discussion mainly was concerned with the impact on the FFTF program resulting from test program slippages and deletions (Subsection 3.6.2).

Figure 3-23. ANL-S6 Sodium Autoclave Cross-Sectional Arrangement
(Based on Reference 3-15)

Figure 3-24. ANL-S6 Radial Thermal Analysis Model (THTE Transient Heat Transfer Code)

3.6 ADMINISTRATIVE

3.6.1 Work Plan for Safety Engineering (PA-XX)

Task C-1 has completed draft work plans for the new Safety Engineering Project (PA-XX). The plans are presently in internal review.

The purpose of this project is to establish procedures and implement action to efficiently resolve the specific safety-related questions pertinent to LMFBR development through direction and utilization of national laboratory safety programs.

3.6.2 Termination of Task C-1

As a result of AEC-RDT redirection of the fast-reactor safety-testing program, all in-reactor safety tests will be concentrated at the national laboratories, and the C-1 transient overpower task will be phased out of PA-10 effective July 1, 1972.

The primary impact of this redirection is to stop progressive work in the series 4b High-Burnup Fuel Experiment capsules—particularly C4M and N (Subsection 3.2). In addition, test planning activities have stopped on Series 6 Comparative Test capsules (Subsection 3.3), and Series 7 Failure Mechanisms and Fuel Movement.

Areas of Task C-1 involving predictive analyses (e.g., round-robin analyses) and liaison activities in national laboratory LMFBR safety programs will continue in PA-XX (Subsection 3.6.1) in support of its efforts to direct and utilize these programs.

REFERENCES

- 3-1. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Forty-First Quarterly Report, November 1971—January 1972, February 1972 (GEAP-10028-41) pp 3-3.*
- 3-2. Honeck, H. C., *THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations*, Brookhaven National Library, November 1962 (BNL).
- 3-3. Protsik, R., "Detailed Nuclear Analysis of TREAT Transient Experiments," Summary submitted for presentation at American Nuclear Society 1972 Winter Meeting.
- 3-4. Protsik, R., et al., *SN1D—A One-Dimensional Discrete Ordinates Transport Code with General Anisotropic Scattering*, Rev 1, May 1970 (GEAQ-0064).
- 3-5. Protsik, R., and Leff, E. G., *Users Manual for DOT2DB—A Two-Dimensional Multigroup Discrete Ordinates Transport Diffusion Code with An Isotropic Scattering*, September 1969 (GEAP-13537).
- 3-6. Agrawal, A. K., and Royle, P., *Use of the DEFORM-II Module of the Advanced SAS Code in Analysis of Fuel Failure Experiments in TREAT, I. Pretest H3 Calculations*, Argonne National Laboratory, May 1972 (ANL/RAS 72-13).
- 3-7. Creer, J. M., and Collingham, R. E., *Thermal Analysis of the TREAT H3 Fuel Failure Experiment*, December 1971 (HEDL-TME 71-175).
- 3-8. Bard, E. E., and Dutt, D. E., *PECT-2 Analysis of the H3 Transient Test*, February 1972 (HEDL-TME 72-28).
- 3-9. Letter October 8, 1971, A. B. Rothman, et al., to J. F. Boland, Subject: "H3, First Failure Threshold Experiment on Irradiated FFTF Type Fuel."

- 3-10. Reactor Development Program Progress Report, September 1971, p. 8.6 (ANL 7861).
- 3-11. Letter November 24, 1971, A. B. Rothman to J. F. Boland, Subject: "H3 Safety Analysis."
- 3-12. Reactor Development Program Progress Report, November 1971, p. 8.11 (ANL 7887).
- 3-13. Oldberg, S.. Jr., *BEHAVE-2: Oxide Fuel Performance Code in Two Spatial Dimensions and Time*, February 1972 (GEAP-13788).
- 3-14. Fuel Modeling Meeting, "H-3 Round-Robin Analysis," held at GE-BRD, Sunnyvale, Calif., Dated: May 18—19, 1972, Attended by ANL, HEDL, GE, and RDT.
- 3-15. ANL Monthly Progress Report ANL-7833, June 1971, pp. 8.28—8.33.
- 3-16. Letter Memo, J. H. Field to C. V. Backlund, Subject: "Comments on HEDL Recommendations on Nonprototypic Effects on In-Reactor Tests," Dated June 23, 1972.
- 3-17. Letter Memo, J. H. Field to Manager, AEC-SAN Office, Subject: Revised Format for the Exchange of Experimental Data from RDT-Sponsored Transient Testing Programs, Dated May 3, 1972.
- 3-18. Bailey, W. J., et al., *FFTF Safety Testing Program - Volume III, In-Reactor Safety Testing Program*, Volume III, May 1972 (HEDL-TME-71-150).
- 3-19. Letter Memo, J. H. Field to C. V. Backlund, Subject: Comments on HEDL-TME 71-150, Vol. III, FFTF Safety Testing Program, Dated July 6, 1972.

4. TASK C-2—SAFETY EXPERIMENTATION (TRANSIENT UNDERCOOLING)

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4.1 GENERAL

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

Several major changes have been made in the PA-10 program for GYF-73. The safety related activities have been separated into a new Safety Engineering program (PA-7X at present), and all experimental testing work has been eliminated. Specifically, the Task C-2 Undercooling Transient Test Series C11 and C12 have been deleted from the PA-10 Program for GFY-73.

This is the last progress report for Task C-2.

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 vibration (accelerometers).

4.2.1 Mockup 1A

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

Review of the Mockup 1A final report (GEAP-13851, "Loss-of-Flow Experiment in an Electrically Heated Sodium Cooled Test Facility") has been completed. All figures for the report have been completed in final form.

* Contributors are listed in Appendix B.

The Mockup 1A test established that the instrumentation used was adequate for monitoring the pressure, temperature and flow conditions during the out-of-pile loss-of-flow transient. A comparison of the test results with predictions from the BUBBLE/VOIDUM code shows qualitative agreement during both early void growth and collapse.

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 included the flow transient as well as the initial stages of void-formation.

The final report on Mockup 1B was initiated. The acoustic monitor (accelerometer) data were in the process of being examined to determine if a correlation exists with the transient coolant (temperature, pressure and flow) data. The removal and examination of the heater section was deferred as a result of the program redirection.

4.3 SERIES C11

The C11 series of tests is designed to 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, non-flowing 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.

Work progressed on the C11A test through the month of June. The C11A Engineering Planning Diagram No. 264R167 was completed and reviewed.

Two Engineering Change Notices (NE 12063 and NE 12065) to Design Basis Specification 22A3211, Rev. 0, Capsule C11A, were issued; these ECN's update the capsule and fuel rod requirements. Detailed drawings of the C11A capsule and fuel rod were issued. Mockup tests of the C11A NaK Expulsion System were successfully completed. This system will be used to initiate the simulated local flow blockage transient. The C11A Final Design Review was held and minutes from this meeting have been distributed.

Drafts of the C11A Capsule Fabrication Specification, C11A Fuel Fabrication Work Guide and C11A Fuel Rod Fabrication Specification have been completed. A management release was issued to control fabrication, inspection and storage of 3-cladding tubes, 1-reflector, and 1-bottom end plug for the C11A fuel rod. Work on these parts has been completed and they have been stored with other fuel rod materials received from HEDL.

The two-dimensional THTD model of the capsule is operational for steady-state thermal calculations. Programming was completed on a subroutine that will initiate the flow blockage transient for the model but this subroutine has not been added to the model. A model using the CREEP/PLAST computer code was developed for the fuel rod steady-state and transient stress analysis. This model remains to be checked out.

A copy of the experimental proposal, test C11A Specification No. 22A1846, Rev. 0, Design Basis Specification, Capsule C11A, Specification No. 22A3211, Rev. 0, and the two Engineering Change Notices were sent to HEDL in response to a request from HEDL to PA-10 Program Management.

Work on the C11A test was terminated at the end of June 1972. Documents and hardware have been placed in controlled storage.

4.4 SERIES C12

The C12 series of tests were planned to investigate transient undercooling accidents using flow sodium multi-rod 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 evaluated with regard to the C12 test objectives. Based upon this evaluation, a recommendation was made to run the test in an "R" loop. However, in response to the program redirection, work on the C12 test has been discontinued.

4.5 MODELING

A free standing version of the VOIDUM code has been modified to include flow resistance in the reactor inlet plenum and IHX operation without cover gas. The following additional options, flow blockage, pipe breakage and pump coastdown with individual controllable pump routines are available for the code. The VOIDUM code programming is complete and the checkout is currently being run.

Work was initiated on reviewing and improving BUBBLE for the evaluation of undercooling transient tests but this work has not been completed. However, BUBBLE exists in a form that would permit it to be applied to selected transient undercooling analyses. It is expected that further application of this code will be done under PA-7X.

4.6 NATIONAL SAFETY TESTING PROGRAM LIAISON AND SURVEILLANCE

Task C-2 participated in a joint meeting of RDT/ANL/GE/HEDL/WARD on May 18 and 19, 1972 to discuss the ANL-H3 test round-robin analyses and the safety testing program in support of FFTF.

Comments on HEDL-TME-71-150, Volume III, FFTF Safety Testing Program were forwarded to AEC/SFO, RDT and HEDL. The comments dealt with changes required in the HEDL document as a result of the deletion of Overpower Transient Test Series C4b, C6 and C7 and Undercooling Transient Test Series C11 and C12.

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 utilized to model or predict the behavior of fuel elements under steady-state or failure conditions. The reactions of fuel and cladding with fission products and sodium are investigated 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, and sodium temperature and oxygen levels. The bulk of this work was concluded June 30, 1972. A portion is being continued under the PA-53 program and additional developments will be reported in progress reports issued under that program.

5.2 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR

5.2.1 Center-Hole Formation at Reactor Startup—Capsule E2R

This capsule, containing 88 and 94% TD mixed-oxide pellets, is intended to investigate fuel restructuring (time for center-hole formation) at reactor startup rates of 1 kW/ft/min and center temperatures of 2500°C. The capsule has been inserted eight times for periods of 1, 10, 15, 34, 60, and 120 minutes three times, making a total exposure of 8 hours.

Neutrographs taken after each exposure revealed that the central hole had formed in the 88% TD 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 formed in the 94% TD pellets. The capsule will not receive any additional irradiation, but will be forwarded to LASL for post-irradiation examination. A description of the proposed examination has been submitted to LASL.

5.3 FISSION PRODUCT-CLADDING REACTIONS

5.3.1 Thermal Gradient Capsule Experiments

A summary of the entire test program and findings to date appears in Table 5-1. All capsules contained (U-25 wt % Pu)O₂±x housed in Type-316 stainless steel tubing from heat 20303 and were heat treated 100 hours. Additives included cesia, iodine, tellurium and molybdenum, alone or in combination.

5.3.1.1 Effects of Cesia and Tellurium Additions

Metallographic views of thermal gradient capsules investigating the effects of fuel stoichiometry, and cesium and tellurium ratios and concentrations, on intergranular attack have become available. The results of test series 8, along with results from related capsules investigated in series 5 and 6, are tabulated in Table 5-2.

* Contributors are listed in Appendix B.

Table 5-1
SUMMARY OF THERMAL GRADIENT CAPSULE EXPERIMENTS. ADDITIVES
MARKED WITH AN ASTERISK CONTAINED TRACER.

Series	O/M	Temperature Range (°C)	Container	Additive	Results
1	1.967	660-1550	Mo	Cs ₂ MoO ₄ /U	Cs ₂ O increased O/M redistribution rate Q* = -7 kcal/mole
2-A	1.967	700-1550	Mo	Cs ₂ MoO ₄ /U	Confirmed the results of Series 1.
2-B	1.907	550-1550	Mo	Cs ₂ MoO ₄ /U	Diffusion-controlled reaction along
2-C	1.967	660-1550	Mo	Cs ₂ MoO ₄ /U	cladding surface between 600°C and 750°C.
3-A	1.967	550-1150	Type-316 Stainless Steel	Cs ₂ MoO ₄ /U	Cs caused diffusion-controlled reaction noted above. Cs ₂ O in either form reacted. Stainless capsules suitable for testing.
3-B	1.967	550-1150		Cs ₂ O	
3-C	1.967	550-1150		None	
4-A	1.94	550-1150		Cs ₂ O*	Amount of Cs ₂ O retained at temperature end increased with O/M. Diffusion-controlled reaction observed again. Oxidation of cladding with O/M ≥ 2.00 Excess Cs ₂ O ruptured Capsule E. Severe intergranular attack.
4-B	1.967	550-1150		Cs ₂ O*	
4-C	2.00-	550-1150		Cs ₂ O*	
4-D	2.01	550-1150		Cs ₂ O*	
4-E	2.00	550-1150		Cs ₂ O Excess (500 mg)	
5-A	1.96	550-1150 ^(a)		Cs ₂ O + Te ^{(b)*}	Te reduced migration of Cs ₂ O toward low temperature end. Cs and Te appear to cohabit in fuel phase. No reaction of cladding with I and Cs. I content did not alter Cs ₂ O distribution. Cs and Te appear to cohabit; Te lessens migration of Cs ₂ O to cold end. Intergranular attack near 800°C, ~ 2 mils
5-B	1.96	550-1150 ^(a)		Cs ₂ O* + I ^{(c)*}	
5-C	2.00-	550-1150 ^(a)		Cs ₂ O* + I ^{(c)*}	
5-D	2.00-	550-1150 ^(a)		Cs ₂ O* + Te ^{(b)*}	
5-E	2.00-	550-1150 ^(a)		Cs ₂ O* Excess (100 mg)	Excess of Cs ₂ O produced drastic intergranular attack. Cs ₂ UO ₄ and Cs ₂ U ₂ O ₇ identified at cold end in E and F.
5-F	2.00-	550-1150 ^(a)		Cs ₂ O* Excess (200 mg)	
6-A	2.00-	550-1150		Cs ₂ O* + Te ^(c)	Cs and Te cohabited. Excess Cs deposited along length, migrating to cold end, but inhibited by Te and less migrated than in D. Cs and Te cohabited at about 725°C. Te restricted Cs migration to cold end. Intergranular attack between 750°C and 950°C as in 5-D, ~ 2 mils
6-B	2.00-	550-1150		Cs ₂ O* + Te ^(b)	
6-C	2.00-	550-1150		Te	Intergranular attack between 750°C and 950°C. Cs deposited along length, more concentrated at hot end.
6-D	2.00-	550-1150		Cs ₂ O*	
6-E	2.00-	550-1150		Cs ₂ O* + I ^{(c)*}	Cs + I concentrated at cold end; excess Cs deposited along capsule length. Cs and I concentrated at cold end as CsI.
6-F	2.00-	550-1150		Cs ₂ O + I ^{(b)*}	

Table 5-1 (Continued)

Series	O/M	Temperature Range (°C)	Container	Additive	Results
7-A	2.00-	550-1150	Type-316 Stainless Steel	$\text{Cs}_2\text{O}^* + \text{Mo}^{(b)} + \text{Mo}^*\text{O}_3$	Mo remained at hot end; Cs_2O deposited along length. No oxide film.
7-B	2.00-	550-1150		$\text{Cs}_2\text{O}^* + \text{Mo}^{(d)} + \text{Mo}^*\text{O}_3$	Mo remained at hot end—oxide film on upper temperature in half of cladding. Mo remained at cold end—oxide film on lower temperature in half of cladding. Mo remained at hot end—no oxide film. No significant change in Cs_2O distribution after additional 100 hours of heat treatment.
7-C	2.00-	550-1150		$\text{MoO}^*_3 + \text{Mo}$	
7-D	2.00-	550-1150		$\text{MoO}^*_3 + \text{Mo}^{(c)}$	
7-E	1.96	550-1150		$\text{MoO}^*_3 + \text{Mo}$	
7-F	2.00-	550-1150		Cs_2O^* (Reheat of Capsule 6-D)	
8-A	1.97	550-1100		$\text{Te}^* \text{ (g)}$	No intergranular attack. Te remained at high temperature end.
8-B	2.00-	550-1100		$\text{Cs}_2\text{O}^* + \text{Te}^{*(b)(g)*}$	1 mil intergranular attack.
8-C	2.00-	550-1100		$\text{Cs}_2\text{O}^* + \text{Te}^{*(c)(g)}$	1 mil intergranular attack.
8-D	2.00-	550-1100		$\text{Te}^{*(h)}$	1 mil intergranular attack.
8-E	2.00-	550-1100		$\text{Cs}_2\text{O}^* + \text{Te}^{*(d)(g)}$	1 mil intergranular attack.
8-F	2.00-	550-1100		$\text{Cs}_2\text{O}^* + \text{Te}^{*(g)} + \text{Mo/Mo}^*\text{O}_3 + \text{I}^*$	1 mil intergranular attack.
9-A	2.00	500-750		Cs_2O	No intergranular attack.
9-B	2.00	500-750		$\text{Cs}_2\text{O} + \text{Te}^{(c)(g)}$	Slight intergranular attack.
9-C	2.00	500-750		$\text{Cs}_2\text{O} + \text{Te}^{(b)(h)}$	~ 2 mils intergranular attack; lowest temperature of attack was 850°C
9-D	2.00	500-750		Cs_2O	No intergranular attack.
9-E	2.00	500-750		$\text{Cs}_2\text{O} + \text{Te}^{(c)(g)}$	Slight intergranular attack.
9-F	2.00	500-750		$\text{Cs}_2\text{O} + \text{Te}^{(b)(h)}$	~ 2 mils intergranular attack, lowest temperature of attack ~ 750°C
9-G	2.00	500-750		Cs_2O	Results are not available
9-H	2.00	500-750		$\text{Cs}_2\text{O} + \text{Te}^{(c)(g)}$	
9-I	2.00	500-750		$\text{Cs}_2\text{O} + \text{Te}^{(b)(h)}$	

(a) Temperature drifted gradually during the heat treatment due to deterioration of the heater windings

(b) 1:1 ratio of Cs to Te or Cs to I or Cs to Mo + MoO_3

(c) 4:1 ratio of Cs to Te or Cs to I

(d) 2:1 ratio of Cs to Mo or Cs to Te

(e) Additive at cold end

(f) All capsules heat treated 100 hours except capsules 9D-F (1000 hours) and 9G-I (900 hours)

(g) Tellurium addition was 12-15 mg

(h) Tellurium addition was 48-50 mg

Contained tracer

Table 5-2
CESIUM-TELLURIUM BEHAVIOR IN TYPE 316 CLAD
THERMAL GRADIENT CAPSULES^(a)

Cs/Te				
Specimen ^(b)	Fuel O/M	Atom Ratio	Weight (mg)	Results
8A	1.97	0/1	0/12	No intergranular attack
6C	2.00	0/1	0/12	1 mil intergranular attack
8D	2.00	0/1	0/50	1 mil intergranular attack
5A	1.97	1/1	50/48	No intergranular attack
6B	2.00	1/1	50/48	2 mils intergranular attack
5D	2.00	1/1	50/48	2 mils intergranular attack
8B	2.00	1/1	15/12	1 mil intergranular attack
8E	2.00	2/1	30/15	1 mil intergranular attack
6A	2.00	4/1	50/15	No intergranular attack
8C	2.00	4/1	50/15	1 mil intergranular attack

(a) All testing was conducted at 550°C to 1100°C.

(b) Test specimens 8A, 6C and 8D investigated O/M effects on additions of tellurium only.

Specimens 5A, 6B, 5D and 8B compare cesium and tellurium in a 1/1 ratio.

Specimens 8E, 6A and 8C cover cesium and tellurium at higher molar ratios.

These results show that with mixed oxide of a 1.97 O/M ratio, tellurium does not react intergranularly (8A). With stoichiometric fuel, however, tellurium does cause intergranular attack (6C, 8D) as shown in Figure 5-1. Similarly, cesium and tellurium in a 1-to-1 ratio with hypostoichiometric fuel do not react intergranularly (5A), but attack cladding when loaded into a capsule with stoichiometric fuel (6B, 5D, 8B). Test capsules 8E and 8C show that intergranular attack occurs with 2-to-1 or 4-to-1 cesium-tellurium ratios and stoichiometric fuel. The latter result contrasts with the findings of test capsule 6A as noted in Table 5-2, in which no reaction occurred with stoichiometric fuel and cesium and tellurium in a 4-to-1 ratio.

These findings imply that the oxygen activity within the test capsule, established by the fuel, determines whether cladding attack will occur. At the low oxygen potential of about -155 kcal/mole, equivalent to fuel with an O/M of about 1.97 at 800°C, no cladding attack occurred. With stoichiometric fuel, at an oxygen potential of ~ -95 kcal/mole, cladding attack occurred. Cladding attack might be minimized, therefore, by additives that would maintain the mixed oxide throughout life at the low oxygen potential of hypostoichiometric fuel. Initial experiments to verify this postulation, using a niobium additive, are described in Subsection 5.3.2.

The contrasting experimental findings of test capsules 6A and 8C with cesium and tellurium in a ratio of 4-to-1 may be related to the formation of a cesium-fuel compound by the following reaction: $2\text{Cs} + 3(\text{U,Pu})\text{O}_2 \rightarrow 2\text{Cs}(\text{U,Pu})\text{O}_3 + (\text{U,Pu})\text{O}_{2-x}$. In this explanation, the cesium that is in excess over a cesium/tellurium ratio of 2, which would permit formation of Cs_2Te , formed a compound with the fuel and left the residual fuel at a lower oxygen potential. When the oxygen activity of the reduced fuel was below that required for cladding attack, none occurred, as was the case in capsule 6A. On the other hand, the oxygen activity of the reduced fuel in capsule 8C may still have been high enough that cladding reaction occurred. A critical level of oxygen activity for cladding attack was approached in these tests by reaction of fuel with the excess cesium in the 4-to-1 cesium-tellurium environment; the disparity between 6A and 8C probably occurred because of slight variations around this threshold level.

NEG No. 61803

AS POLISHED

400X

Figure 5-1. Grain Boundary Attack in Type-316 Stainless Steel with Stoichiometric Fuel and Cesium and Tellurium in Ratio of One to One

5.3.1.2 Effects of Cesium, Tellurium, Iodine and Molybdenum Additions

A description of test capsule 8F along with a gamma scan that showed locations of the radioactive tracers, was presented in the previous Quarterly Progress Report.⁵⁻¹ Metallographic examination of the capsule wall showed that intergranular attack had occurred. This is attributed to cesium and tellurium because of the findings described above. Earlier experiments with molybdenum additions or with iodine additions showed that these elements did not produce intergranular attack.

5.3.1.3 Low-Temperature Heat Treatments

A total of nine thermal gradient capsules have completed testing for 100-, 900- and 1000-hour periods in the low temperature range of 500 to 750°C. A triplicate test matrix was followed in which the set of three additives to stoichiometric mixed oxide — Cs_2O alone and $\text{Cs}_2\text{O}+\text{Te}$ in ratios of 1-to-1 and 4-to-1 — were kept constant. All capsules were clad in Type-316 stainless steel. The experimental aim was to develop intergranular attack at low temperatures and to investigate the role of time in this reaction. Metallographic results are available from specimens heat treated 100 and 1000 hours. As shown in Figure 5-1, the specimen containing cesium oxide and tellurium in a ratio of 1-to-1 showed extensive intergranular attack to a depth of about 2 mils. This specimen, heat treated for 100 hours, showed attack at a minimum temperature of 850°C. After 1000 hours this attack was observed at temperatures as low as 750°C. No significant intergranular attack occurred with cesium alone or with cesium and tellurium in the ratio of 4-to-1. This experiment showed that both time and temperature are parameters that affect the extent of intergranular attack.

5.3.1.4 Microprobe Examination of Unirradiated Specimens from Thermal Gradient Test Capsules

Two cladding samples from capsules 6B and 6C of Table 5-1 that were exposed within the temperature range of 885 to 1005°C have been examined by electron microprobe. In addition to stoichiometric fuel these rods contained cesium and tellurium in 1-to-1 or 0-to-1 relationships, respectively; both samples revealed intergranular attack. The microprobe x-ray images showed that cesium, oxygen, and tellurium were present

within the grain boundaries or at the fuel-cladding interface. The behavior of chromium in these two environments was informative. As Figure 5-2 shows, chromium concentrated at the fuel-cladding interface in 6C, with tellurium alone; when both cesium and tellurium were present no chromium concentration took place. Apparently, the addition of tellurium did not significantly reduce the oxygen activity within the capsule and the cladding interior surface oxidized as Cr_2O_3 . In the presence of cesia and tellurium, however, the cesia or the cesium-tellurium compound may have combined with the Cr_2O_3 and caused that oxide to dissolve or disappear as a prerequisite to intergranular attack on the cladding.

5.3.2 Fuel Additive Program

5.3.2.1 Oxygen Control Program

A series of three thermal gradient capsules completed a 100-hour test run within the 550 to 1100°C temperature range. These are identified in Table 5-3 as 2A, 2B, and 2C. All capsules utilized Type-316 stainless steel cladding and stoichiometric mixed-oxide fuel. Cesium and tellurium were added to three of the capsules in the 1-to-1 proportion that had been found, in earlier capsules, to cause intergranular attack. Niobium discs were included in capsule 2B while niobium powder was added to pellets of capsule 2C. The niobium additions were expected to reduce the oxygen activity within these two capsules as discussed in Subsection 5.3.1.1 and thereby lessen cladding attack in comparison to control capsule 2A that did not contain niobium.

Table 5-3
FEATURES OF CAPSULES TESTED IN THE FUEL ADDITIVE PROGRAM

Series	O/M	Temperature Range (°C)	Additive
1-A	2.00	550-1100 C	500 ppm C as $\text{BaCO}_3 + \text{Cs}_2\text{O}$
1-B	2.00	550-1100 C	500 ppm C as BaCO_3
1-C	2.00	550-1100 C	500 ppm N_2 as UN + Cs_2O
1-D	2.00	550-1100 C	500 ppm N_2 as UN
1-E	1.96	550-1100 C	500 ppm N_2 as UN + Cs_2O
1-F	1.96	550-1100 C	500 ppm N_2 as UN
2-A	2.00	550-1100 C	Te + Cs_2O
2-B	2.00	550-1100 C	Te + Cs_2O + Nb (discs)
2-C	2.00	550-1100 C	Te + Cs_2O + Nb (powder)
2-D	2.00	550-1100 C	-
2-E	2.00	550-1100 C	500 ppm C as graphite + Cs_2O
2-F	2.00	550-1100 C	500 ppm C as graphite

Visual examination of the niobium that had been added to capsules 2B and 2C within the 600 to 1000°C zone where intergranular attack had previously occurred, showed a black oxide surface, probably NbO , at the cooler region while at the hotter region the surface coating was white, characteristic of Nb_2O_5 .

Metallographic examination of cladding samples that had operated at ~850°C from all three samples showed that the niobium additions had lessened intergranular attack. This is illustrated in the photomicrographs of Figure 5-3. The control sample (2A) showed ~2.5 mils of extensive intergranular attack. Cladding from sample 2B with niobium discs showed ~1.8 mils of continuous penetration. Cladding from 2C with niobium powder within fuel pellets showed much less reaction and discontinuous intergranular attack for a maximum depth of about 1.7 mils. This initial experiment shows the potential of niobium in lessening fission-product attack of cladding; niobium powder was most effective under these conditions. Refinements of these results and further experimentation will be conducted and reported in the PA-53 Progress Reports.

MICROVIEW

800X

CHROMIUM X-RAY IMAGE

800X

EFFECT OF TELLURIUM ADDITION ON CLADDING SURFACE

MICROVIEW

800X

Cr

800X

Cs

800X

— X-RAY IMAGES —

EFFECT OF CESIA AND TELLURIUM ADDITION ON CLADDING SURFACE

Figure 5-2. Intergranular Attack in Type-316 Stainless Steel Cladding. Chromium Concentrates at Cladding Surface When Tellurium is Added to Stoichiometric Fuel but is Not Present When Cesium and Tellurium Are Added

(a) CONTROL SAMPLE

(b) Nb DISCS

(c) Nb POWDER

NOTE: ALL CAPSULES CONTAINED STOICHIOMETRIC FUEL WITH Cs + Te IN 1/1 RATIO.

Figure 5-3. Samples of Type-316 Cladding that Operated at 850°C in Thermal Gradient Capsules 2A, 2B, and 2C.

5.3.3 Fuel Impurity Program

5.3.3.1 Carbon and Nitrogen Effects

A series of capsules investigating the roles of carbon and nitrogen in cladding attack were heat treated 100 hours in a 550 to 1100°C thermal gradient. Characteristics of the capsules are listed in Table 5-3. Essentially 500 ppm concentrations of carbon as BaCO_3 were added to the hot ends of the capsules with (1A) and without cesia (1B). Approximately 500 ppm of nitrogen as UN were also included in capsules with and without cesia (1C and 1D). The results of the testing showed that slight intergranular attack occurred in capsules of stoichiometric fuel and carbon or nitrogen. No cladding reaction took place when these elements were encapsulated with hypostoichiometric fuel (1E, 1F). Also, no reactions occurred in capsules with stoichiometric fuel and cesium oxide.

In a follow-up test 500 ppm carbon was added to stoichiometric fuel with and without cesium oxide (2E, 2F). As in the earlier case, slight intergranular attack occurred in the sample without the cesia. Under these conditions, carbon actively participates in cladding attack reactions. Samples of these reactions are presented in Figure 5-4.

5.3.4 Cladding Modification Program

The metallographic examination of two thermal gradient capsules of Type-316 stainless steel housing stoichiometric fuel with Cs_2O additions showed that the silicon content of the steel may have an effect on cladding reactions. An intragranular precipitate formed within the cladding at the fuel interface of these samples, and this is believed to be made up of carbide or nitride particles. More precipitate appeared in the cladding with low silicon (0.17%) than in the cladding with a typical silicon concentration (0.6%).

5.4 FUEL-SODIUM REACTION KINETICS

Additional tests of stainless steel clad $\text{UO}_{2.05}$ have been conducted in sodium at 650°C (1202°F). In two cases an open one-half-inch long, 2-mil wide slit exposed the fuel to the sodium in the same manner as the experiment described in the previous quarterly.⁵⁻¹ The initial test lasted 54 hours and a $\Delta D/D$ of 10.7 mils was measured at the elevation of swelling, 90 degrees from the defect. The second test was for 24 hours and an 11.9 mil dimensional increase was measured. In a third 24-hour test, a 30-mil hole was used in the place of the slit. This capsule showed a maximum diametral increase of 6.5 mils, about one-half the increase that developed in the specimen with the slit, indicating that configuration of cladding opening affects the degree of reaction. Dimensional measurements were taken at intervals along the 2-1/2-inch long capsule, and the increases are listed below:

Location (inches from top)	Diametral Increase, mils	
	0 degrees	90 degrees
0.5	6.5	-1
1.0	5.1	2.5
1.25	3.7	N.A.
1.5	4.5	4.4
2.0	1.5	5.3

The location of the 30-mil hole was 1.25 inches from the top. These results show that the sodium completely penetrated the capsule and that nonuniform swelling of the fuel occurred both axially and circumferentially. In all of these tests, the oxygen concentration of the sodium was 60 to 70 ppm.

A fourth test capsule was exposed to sodium at 650°C for 24 hours. The slit geometry of the opening was used, but the test was distinctive because uranium foil was positioned at the opening to lower the oxygen content of the sodium in the vicinity of the fuel. The measured swelling was only 3.8 mils at the maximum location. The uranium foil was expected to reduce the oxygen concentration of the sodium from

(a) CONTROL SAMPLE

(b) GRAPHITE AND CS_2O ADDITION

(c) GRAPHITE ADDITION

Figure 5-4. Carbon Promotion of Intergranular Attack in Fueled Thermal Gradient Capsules at 850°C

about 67 ppm to about 1 ppm, and at this level of oxygen, the fuel would not be expected to swell. The measured swelling is presumed to have occurred during the period when the uranium foil was lowering the oxygen concentration to the low equilibrium figure. A similar use of uranium foil successfully reduced the oxygen concentration of the sodium in the GETR-tested B9D capsule.⁵⁻² The transducer did not record the 3.8 mil diametral increase, possibly because the swelling occurred very quickly, before the transducer had thermally stabilized at 650°C.

5.5 QUANTITATIVE METALLOGRAPHY

Quantitative porosity determinations were made on 500X photomicrographs of two irradiated fuel rods using the Quantimet Image Analyzing Computer. The examined rods were solid pellet fueled F9C-31 at 6.9 at. % burnup and E1H-23, loaded with 96.4% TD annular pellets that reached a burnup of 8.5 at. %. Fabrication and irradiation features of F9C-31, which approximate parameters of FFTF and near-term LMFBRs, and of E1H-23 are listed in Table 5-4. Figure 5-5 illustrates the radial density distribution of F9C-31 at the peak heat flux location. Results of porosity distribution and microstructural boundary measurements for rod E1H-23 are presented in Figure 5-6. This rod underwent a thorough post-irradiation examination which included 13 transverse sections along its length. The various structural boundaries were measured on 100X composite photomicrographs while porosity determinations were made on seven of these sections. Structural boundary data are presented on the left side of Figure 5-6 and corresponding radial density distributions are given on the right side of the figure.

Table 5-4
FABRICATION AND IRRADIATION PARAMETERS OF MIXED-OXIDE FUEL RODS
SUBJECTED TO QUANTITATIVE METALLOGRAPHY

Rod Number	O/M	Fuel Density (% TD)	Smeared Density (% TD)	Fuel-Cladding Diametral Gap (mils)	Calculated Peak Burnup ^(a) (Atom Percent)	Calculated Peak Linear Power ^(a) (kW/ft)
E1H-23 ^(b)	1.97	96.4	88.5	2.9 - 4.9	8.45	15.7
F9C-31	1.97	88.5	84.3	3.8 - 6.0	6.90	13.3

(a) Based on values in the "Experimenters Guide for Irradiation in EBR-II". Revision 4, February, 1971.

(b) Annular fuel pellets with a 0.05-inch diameter centerline hole.

In general, the porosity profiles and structural boundaries indicate up to five radial regions. There is an outer region with densities that range from 70 to 89%. Although a portion of the measured porosity is believed to be the result of pull-out during metallographic preparation, the post-irradiation density of this region is clearly lower than the initial fuel density of the annular pellets of E1H-23. For solid pellet-fueled F9C-31, the density of this region corresponds to the fabricated value. The next inward radial zone is an equiaxed-grain-growth region of fuel with density measurements that are generally greater than the initial density. Radially inward from this high density zone is a region of equiaxed-grained fuel containing a considerable volume of porosity at the grain boundaries with some intragranular porosity. Many of the pores in this region are spherical in shape, suggesting that some may be high-pressure fission gas bubbles. Inward from this region is the typical region of columnar grain growth which measured 99% dense. The overall density of the columnar-grained region including grain boundary porosity is 92 to 95% of theoretical. In many sections, there is an increased amount of porosity near the central void, probably because the pores in gas bubbles migrate more slowly in this region where the temperature gradient is much lower; this constitutes an additional microstructural zone.

The results from these and other fuel rods suggest that the three-region model in its usual form⁵⁻³ does not adequately describe the fuel microstructure and porosity distribution in typical mixed-oxide fuel rods. A multi-region model would appear to provide a more accurate microstructural description.

Figure 5-5. Microstructure and Radial Density Profile at the Peak Flux Region of Fuel Rod F9C-31. Fuel Density was 88.5% TD while Smeared Density was 84.3% TD

Figure 5-6. Structural Boundary Radii and Porosity Distribution Profiles Measured on Fuel Rod E1H-23

REFERENCES

- 5-1. *Sodium Cooled Reactors, Fast Ceramic Reactor Development Program*, Forty-Second Quarterly Report, February—April 1972, May 1972 (GEAP-10028-42).
- 5-2. Bohaboy, P. E., Craig, C. N., and Stimmell, G. L., "Performance of Failed Plutonium-Uranium Oxide Fuel Elements in Flowing Sodium," *Trans. Am. Nucl. Soc.*, **15-1**, p. 196, 1972.
- 5-3. Jankus, V. Z., and Weeks, R. W., *LIFE-I, A Fortran-IV Computer Code for the Prediction of Fast-Reactor Fuel-Element Behavior*, Argonne National Laboratory, November 1970 (ANL-7736).

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 LMFBFR 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.

Future work in this area will be performed under PA-10, Task 2, "Fuel Rod Engineering."

6.2 HIGH BURNUP IRRADIATIONS

F4 Experiment (Medium Density Fuel Performance at High Power)

The objective of the F4 experiment is to determine the performance of fuel rods with medium density fuel at high power. The F4 experiment contains 9 encapsulated fuel rods with 84 to 87% theoretical smeared fuel density, and clad with annealed Types-304 and -316 stainless steel, and Incoloy-800. Fuel rods F4D and F4F were destructively examined at 4 at.% burnup. The remaining 7 fuel rods reached 9 at.% peak burnup at >16 kW/ft peak start-of-life power with 1150°F maximum cladding temperature (nominal**).

The post-irradiation examination of F4A and F4E is in progress at LASL. The results of the Betatron radiography were reported last quarter. There were no indications of fuel rod failure. This was confirmed by analysis of the capsule cover gas which showed neither xenon nor krypton to be present.

* Contributors are listed in Appendix B.

** "Nominal" operating conditions are based on values recommended in the "Guide for Irradiation Experiments in EBR-II," Revision 4, February 1971.

The rods were examined visually after de-encapsulation and appeared to be in good condition. Rod bow was slight. The F4A wire spacer was tight. The F4E wire spacer was slightly displaced from its original position. This could have occurred during de-encapsulation. The F4A rod was clean while F4E was badly tarnished.

The LASL gamma scans were consistent with the GE scans reported last quarter. The isotopic distribution of Rh-106 in F4A shows four locations of significant concentration. The Cs-137 distribution is very nonuniform with some concentration between individual pellets. The Zr-95 and Nb-95 distributions are nearly identical over the fuel region. The distribution of Cs-134 shows spikes at both ends of the fuel column and some regions where the Cs-134 has adsorbed on the UO₂ insulator pellets above the fuel.

In F4E, the Cs-134 has migrated and deposited at the top and at the bottom. The Cs-137 isotope has deposited at the ends of the fuel column and in the inter-pellet gaps. The Rh-106 distribution shows localized axial relocation near the bottom end of the fuel column. The distributions of Zr-95 and Nb-95 show a slight increase at the top of the fuel column, and are nearly identical over the fuel region.

Profilometry of the rods showed an increase in diameter of ~4.5 mils maximum for F4A (1.8% $\Delta D/D$). The diameter increase for F4E was ~6 mils maximum (2.4% $\Delta D/D$).

The rods have been sectioned, and the metallographic examination is in progress.

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

The objective of the F12P and F12Q experiments is to determine the relative restraint of fuel growth provided by cladding with 10 mil and 15 mil wall thickness. The encapsulated rods, with 2-inch fuel column length with 91% theoretical smeared fuel density and clad with annealed Type-316 stainless steel, were irradiated at 15 kW/ft to 100,000 MWd/Te in the GETR at cladding temperatures of 1150°F.

The post-irradiation examination of F12P and F12Q is in progress at LASL. The results of the Betatron radiography were reported last quarter. The central void in F12Q (10 mil wall) is substantially larger than the central void in F12P (15 mil wall); this indicates the F12Q probably ran hotter than F12P for some portion of the irradiation.

Visual examination showed the rods to be in good condition. Both had slight axial distortion, and the surface of both was tarnished.

Gamma scanning is complete. Ruthenium-103 and Rh-106 isotopic distributions for F12P are similar, showing significant axial relocation. The Cs-134 and Cs-137 distributions show that the isotopes have migrated, depositing on the top and bottom UO₂ insulator pellets and the bottom end plug. The Zr-95 and Nb-95 distributions are similar, with spikes at the ends of the fuel column.

The Ru-103 isotope has concentrated near the top of the fuel column in F12Q. The Rh-106 has migrated to the top pellet of the fuel column. The Cs-137 isotope has moved from the fuel and deposited on the UO₂ insulator pellets at the bottom and top of the fuel column. The Zr-95 and Nb-95 isotopic distributions are similar, as in F12P, with spikes at the ends of the fuel column. Cesium-134 has an axial distribution similar to the Cs-137 distribution.

These results, including profilometry, are being evaluated and compared with the earlier non-destructive examination completed at GE.

F8B Experiment (Void Deployment Effect in Low Power Operation)

The objective of the F8B test is to determine the effect of void deployment on the performance of fuel rods at low power. The F8B consists of nineteen encapsulated fuel rods with smeared densities of 83 to 92% theoretical and clad with Type-304 and -316 stainless steel and Incoloy-800, and contains solid

pellets, annular pellets, dished pellets and VIPAC fuel. The rods have operated between 5 and 10 kW/ft to ~7 at.% burnup (nominal*).

Fuel rod F8B-5 was discovered to be damaged during reconstitution of the subassembly. The capsule is intact, but the rod will no longer fit on the T-Bar support plate in the proper manner. The rod cannot be reinserted in-pile. It is currently being stored at EBR-II pending shipment to LASL for destructive examination.

6.3 HIGH CLADDING TEMPERATURE IRRADIATIONS

F11A Experiment (Wire Spaced Subassembly Irradiation with 1300°F Cladding Temperature)

The objective of the F11A test is to determine the performance of unencapsulated fuel rods clad with Types-304, -316, -321 and -347 stainless steel in several structural conditions at cladding temperatures near the peak expected in the near term LMFBR. The fuel rods will be irradiated in a wire spaced subassembly.

The qualification of the end plug welds for the F11A replacement fuel rods with Type-321 stainless steel cladding was successfully completed. Fabrication of the replacement and spare rods is complete and final inspection and documentation is in progress.

The across-the-inside flats dimensions of the J37 channel intended for the F11A experiment were measured at EBR-II with the GE Channel Gage. The dimensions were uniform along the channel and on the high side of nominal.

F5 Experiment (Fuel Rod Performance at High Cladding Temperature)

Immersion density measurements show zero cladding swelling in a cold-worked Type-316 stainless steel clad mixed-oxide fuel rod irradiated to a maximum burnup of 6.2 at.% and a fast fluence ($E > 0.1$ MeV) of 3.4×10^{22} n/cm². Therefore, the maximum cladding diameter change of 0.54% was entirely plastic deformation, presumably the result of fuel-cladding mechanical interaction since fission gas pressures were small. Similar fuel rods clad with solution annealed Type-316 or -321 stainless steel exhibited some cladding swelling and a smaller amount of plastic strain. Figure 6-1 shows the total cladding deformations from profilometry measurements together with the cladding swelling and plastic deformation contributions. Fabrication and irradiation parameters for the three Group F5 fuel rods are given in Table 6-1. The fuel smeared densities of 90 to 91% are higher than for the initial FTR fuel rods and consequently FTR cladding could be expected to show less plastic deformation. However, the analysis of results from a large number of EBR-II irradiated rods clad with solution annealed Type-316 stainless steel revealed essentially no effect of fuel smeared density on cladding plastic deformation at smeared densities less than ~91% TD.^{a-1}

Table 6-1
FABRICATION AND IRRADIATION PARAMETERS OF GROUP F5 FUEL RODS

Rod Number	Cladding	O/M	Fuel Density (% TD)	Smeared Density (% TD)	Peak Burnup ^(a) (Atom Percent)	Peak Linear Power ^(a) (kW/ft)	Peak Cladding Midwall Temp. °F
F5Q	Type-316 SS 50% Cold Worked	1.97	94.3	90.7	6.2	15.7	1157
F5F	Type-316 SS Solution Annealed	1.97	94.0	90.3	6.2	15.7	1157
F5M	Type-321 SS Solution Annealed	1.97	94.1	89.9	6.3	15.9	1160

(a) Based on values in the "Experimenters Guide for Irradiations in EBR-II," Revision 4, February 1971.

* "Nominal" operating conditions are based on values recommended in the "Guide for Irradiation Experiments in EBR-II," Revision 4, February 1971.

Figure 6-1. Diameter Increase Data for Mixed-Oxide Fuel Rods Clad in Various Alloys

6.4 OPERATING LIMITS IRRADIATIONS

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

The objective of the F9C-1 and F9C-2 tests is to provide data on the threshold of failure and the consequences of failure for wire wrapped, unencapsulated fuel rods. Prior to reconstitution into two Mark E19 subassemblies (high risk test vehicles), 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.

The post-irradiation examination of fuel rod F9C-13 is in progress at LASL. This fuel rod, with 87% theoretical smeared fuel density, clad with annealed Type-316 stainless steel, was irradiated to 6.5 at.% peak burnup (4×10^{22} n/cm², $E > 0.1$ MeV) at 12.5 kW/ft peak start-of-life power with 935°F maximum cladding temperature. The results of the Betatron radiography were reported last quarter. The non-destructive phase of the examination is now complete, and the metallographic examination is underway. Cladding attack was negligible, based upon preliminary evaluation of the longitudinal section at the fuel-upper blanket interface.

The maximum increase in cladding diameter was one mil ($0.4 \Delta D/D$). The diameter along the length of the fuel column was measured at one inch intervals using a calibrated micrometer. This fuel rod was bent during the interim examination at EBR-II, and could not be measured on the profilometer.

The gamma scans showed that the Rh-106 distribution was smooth. The Cs-137 isotope had migrated axially, depositing in the interpellet gaps in the fuel column, and on the surface of the bottom end plug. The Cs-134 distribution was similar to that for the Cs-137; it was concentrated in the interpellet gaps and near the top and bottom of the fuel column. The isotopic distributions of Zr-95 and Nb-95 were identical.

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 pre-irradiated fuel.

Fuel pellet preparation is complete. Both fuel and cladding are ready for loading.

Some glass fibers containing small concentrations of neutron monitoring elements have been prepared. Four fissile isotopes and six types of non-fission monitors will be inserted in special filler rods between shroud tubes to provide dosimetry for the 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.

The F9E subassembly was successfully assembled and inspected at EBR-II. Approximately 240 pounds of force was required to seat the inner grid-spaced channel on the lower shield. Radiography revealed that all grids were intact and that there was no axial displacement. Flow testing of the F9E assembly yielded values of 54 gpm at 41 psi pressure drop and 56 gpm at 44 psi. The F9E has been inserted in-pile and began irradiation with Run 56 in July 1972.

* "Nominal" operating conditions are based on values recommended in the "Guide for Irradiation Experiments in EBR-II," Revision 4, February 1971.

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 stainless steel 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 grid spaced channel is complete. Final inspection, which will include a trial fitup of parts at the GE site prior to shipment to EBR-II, is underway. Final documentation and the quality assurance audit are in progress.

Water flow testing of the mockup F9F assembly, including dummy fuel rods, is complete. The results will be used to size the in-reactor F9F assembly orifice.

6.6 IRRADIATIONS EVALUATION AND SUPPORT

A compilation of all the calculated and measured burnup data on General Electric rods irradiated in EBR-II was prepared and forwarded to ANL to be used in the development of BURNUP HISTORY I. Correlations for the axial profile, historical changes, and local perturbations were included. A complete listing of the uranium and plutonium isotopic analyses for the GE experimental fuel was also forwarded to EBR-II for use in this program.

REFERENCES

- 6-1. Rubin, B. F., Black, T. J., Appleby, W. K., Stephen, J. D., and Hilbert, R. F., "Fuel-Cladding Mechanical Interaction in LMFBR Fuel Rods," paper presented at ANS Topical Meeting on Reactor Materials Performance, Richland, Washington (April 1972).

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.

Another purpose of this task is to contribute to the understanding of fuel performance, core design and thermal hydraulic problem areas, and to provide guidance for effective planning and coordination of experimental programs.

The master schedule for grid-spaced driver fuel assembly development remains essentially as before with the main focus on developing grid-spaced FTR driver fuel assemblies which will be available for testing in the first core of the FTR in June 1974. An updated schedule is included in Figure 7-1.

7.2 F9E DIAGNOSTIC GRID SPACER EXPERIMENT IN EBR-II

The orifice design of the F9E group irradiation was modified to take account of the change in heat generation rate in the fuel due to the installation of a stainless steel reflector in EBR-II.

Final assembly of F9E was completed at ANL under the surveillance of the design engineer. On June 25 the assembly was installed into EBR-II core position 6C4 in preparation for reactor run 56. The x-ray technique for the inspection of grid spacer tabs in the channel recesses was used during the assembly of F9E.

7.3 F9F DIAGNOSTIC GRID SPACER EXPERIMENT IN EBR-II

Fabrication of the inner channel assembly was completed under the surveillance of the design engineer.

Two orifice configurations were flow tested and the data evaluated. This will be used to determine the final design configuration when information is supplied by ANL regarding core location and power rating.

7.4 FFTF GRID SPACED DRIVER FUEL ASSEMBLY

The schedule for grid-spaced assemblies is aimed at fabricating two prototypic mock-up assemblies for completion in August 1973 and four reactor assemblies for completion in May 1974.

* Contributors are listed in Appendix B.

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

A formal request for Approval In Principle (AIP) for the FTR grid-spaced driver fuel assemblies was submitted to the AEC in June. Two assemblies in each fuel enrichment zone, designated GSA-1, 2, 3 and 4 have been formally requested.

A preliminary draft of the System Design Description (SDD 22A3345) for the FTR grid-spaced driver fuel assembly was completed during May. This SDD is being reviewed prior to issue on formal distribution.

7.5 SODIUM FLOW TEST

The long-term sodium flow test conducted in the General Electric Breeder Test Facility Control Drive Test Loop, to evaluate the endurance characteristics of the reference FTR Grid Spacer Design, has been completed. Preparations are being made for examining the two 37-rod dummy fuel elements which had been positioned in series in the test loop. A total of 4408 hours of operation at 1000°F with a peak coolant velocity of 22.3 fps was logged.

A Cleaning Specification has been prepared based upon the EBR-II moisturized argon method. Procedures have been written for the disassembly and detailed inspection of the full-grid spaced assembly which was initiated in July.

7.6 HYDRAULIC FLOW TEST

The 217-rod full scale water flow test has been postponed due to the dislodgement of most grids during final assembly of the rod bundle into the channel. The final stage of assembly required that all 217 rods be pushed simultaneously through the grid spacers—a distance of some three inches. During this operation, the grids were dislodged from their locations in the channel wall. Of the 12 staggered grids and 6 full grids in the assembly, only one staggered grid remained in position. The results of prior rod and grid push tests gave no indication that this dislodgement would occur.

A thorough investigation was subsequently carried out to assess the possible causes of the dislodgement and to determine the necessary corrective action to be taken in the design and fabrication as appropriate. A specific series of mechanical tests to verify the adequacy of the recommended design and fabrication procedures will be part of the corrective actions taken.

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 coolants.

An analysis of pre-failure axial fuel motion, or relocation, using the TOFF, THT and FREADM codes was performed as a test case to illustrate TOFF-II operational status. In particular, the axial fuel motion model in TOFF was checked out.

The fuel motion model, described in the 42nd Quarterly Report, provides means of relating fuel motion to melt fraction, or cladding failure as the case may be. At any time after the transient starts the normalized axial fuel displacement is computed from the power and temperature distributions as they determine fuel melt fractions in each core zone. The axial fuel motion as a function of time was computed for each radial core zone for a Demonstration Plant design.

A modified reference Demonstration Plant core design was used to illustrate fuel motion effects. A 30-inch fuel column length and 36-inch plenum length were assumed. A uniform burnup of 50,000 MWd/Te was assumed in all the fuel. The gap between fuel and cladding was assumed to be completely closed, and the cladding fuel frictional force was assumed equal to the radial force resulting from thermal expansion mis-match between fuel and cladding. (However, for the assumed burnup, the plenum gas pressure was high enough to provide the dominant axial resistive force.) The failure criteria for the cladding was assumed to be 4% strain or an average fuel temperature of 6000°F, at which melt-through or fuel vapor formation occurs.

The most significant force was found to be the gas pressure in the plenum acting downward on the upper axial blanket. Thus, to allow for axial relocation for the assumed failure threshold, the plenum pressure must remain low (less than 325 psi). This pressure limit will be met for end-of-life conditions in a demonstration plant only if either a 70-inch plenum length or vented fuel is included in the design. This calculation has shown that axial fuel relocation is very sensitive to the plenum pressure and, therefore, to the plenum geometry and fuel burnup.

In this study, friction between the upper axial blanket and its cladding were ignored or assumed negligible, it being assumed that such friction could be minimized by design. While the concept of friction between the fuel and cladding is valid, correctly estimating the normal force is difficult because the precise state of the fuel is not known at the transient start. In particular, the fuel-to-cladding gap size is still disputed. (These uncertainties would be lessened if the molten fuel and fission gas were allowed to axially relocate through a central hole in the upper part of the fuel and blanket, a relocation concept which has been demonstrated experimentally,⁸⁻¹¹ but which was not used in this analysis.)

* Contributors are listed in Appendix B.

The effects calculated here have also been shown to be very sensitive to cladding failure mechanisms and thresholds. While the criteria used here are deemed reasonable, very little is known about failure threshold for irradiated cladding. A review of existing experimental data and modification of future experiments to supplement these data have been proposed to increase our knowledge in this area.

Figure 8-1 illustrates the fuel motion (normal expansion and relocation) following a $50 \text{ } \mu\text{C}/\text{second}$ reactivity ramp insertion in the assumed core design. The axial expansion is rather uniform until 2 seconds into the transient, the small increase in length being attributed to normal thermal expansion. However, as the transient continues, the peak rod melts and axial relocation occurs as a result of the increase in pressure of fission gas in the central void overcoming wall friction and the plenum gas pressure (the central void and plenum are assumed to be in communication). This results in the non-uniformity in the high-power zones of the core as shown in Figure 8-1. Neglecting negative reactivity feedback, the transient continues until failure of the cladding occurs in the outer zone at about 2.82 seconds into the transient.

Note that the majority of the core has experienced some fuel relocation after 2.5 seconds. However, the melt fractions computed were small (~ 1 to 4%) until 2.8 seconds. Subsequently large melt fractions ($\sim 40\%$) were computed for peak power positions in the core. In reality, before very extensive melting occurs, the negative reactivity feedback from such large dimensional increases should be capable of limiting the reactivity insertion.

This calculation has illustrated a special case in which axial fuel motion occurs prior to cladding failure and results in reactivity feedback to limit an excursion. This is not a general result and is very dependent on design and operating conditions. Nevertheless, the calculation illustrates the potential for designing LMFBR fuel which has some measure of inherent safety.

Figure 8-1. Axial Fuel Motion by Core Zone following a $50 \text{ } \mu\text{C}/\text{sec}$ Reactivity Ramp Insertion

8.2.2 FREAMM Update

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

Specifications were written for the following modifications to FREAMM:

- Additional heat transfer calculations for axial blanket and gas plenum.
- Calculation of film thickness for heat transfer with sodium boiling (BUB2).
- Calculation of bubble pressure when voiding is initiated (BUB3).

These modifications will complete the addition to the FREAMM code of boiling heat transfer calculations with film evaporation and vapor condensation (i.e., coupling of BUBBLE to FREAMM).

REFERENCES

- 8-1. Hikido, T., and Field, J. H., *Molten Fuel Movement in Transient Overpower Tests on Irradiated Oxide Fuel*, General Electric Company, September 1969 (GEAP-13543).

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 development planning in support of the Demonstration Plant in safety and related areas.

9.2 LIAISON

The resident GE liaison engineer at ANL has continued to provide both organizations with information and development results of mutual interest. His communications were concentrated on the advances in the LMFBR safety analysis and the safety in-pile testing programs. However, he also exchanged and relayed information pertaining to core design, instrumentation and other LMFBR component-related research.

During this period GE has obtained access to the REXCO containment analysis code system and the SAS-2A safety analysis code, both designed at ANL. The resident engineer has assisted in securing user guidance during the early application phase of both models.

The commentary subjects chosen for the periodic reports were either in response to particular requests received from GE or were intended to call GE-BRD's attention to planned ANL development work which could possibly be directly related or adapted to GE design needs or requirements.

The subjects covered the following classifications: TREAT MARK II Loop Test Program, Fuel Element Failure Propagation Project, Accident Analysis, Containment Analysis, Post Accident Heat Removal, Coolant Dynamics, TREAT Converter, Inherently Safe Fuel Development, Core Restraining Methods, Flow Induced Vibration Testing, Sodium Leak Detection, Acoustic Monitoring, Control Rod Material Testing, and Computer Code Maintenance.

9.2.1 Participation in ANL Activities

The liaison engineer spent about one-third of his time on engineering support calculations, pre-test analysis and safety evaluation for the TREAT R-series test vehicle and initial flow transient experiments.

* 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; feasibility depends especially on the survival of the cladding at temperatures and specific fuel rod powers of practical interest.

The objectives of the cladding materials development program are:

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 capsules for irradiating clad alloy specimens in EBR-II and 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 1-4.

10.2.1 Series L-15 and MT-4 Materials Capsules

The L-15 and MT-4 capsules are designed to irradiate precut tensile specimens for post-irradiation mechanical testing. The four L-15 and the MT-4 materials capsules (Mark-A19 type) have completed irradiation in S/A X148. The total fluence accumulated to date for these capsules is 3.5×10^{22} and 9.1×10^{22} n/cm² for the L-15 and the MT-4 capsules, respectively. The L-15 were discharged from EBR-II and shipped to VNC for post-irradiation examination. The MT-4 capsule is scheduled for reconstitution into S/A X148A for continued irradiation. Further work on the series L-15 and MT-4 capsules will be performed in FY-73 under PA-10, Task 2 Fuel Rod Engineering.

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 clad alloys including Types 304, 316, and 321 as a function of temperature of irradiation, metallurgical condition and fluence. The L-16 program plan is approved 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 clad alloy swelling. The initial subassembly in the L-16 program is X098. All future S/As 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. Only ten (10) of the initial 19 capsules were discharged; one less than originally planned. The ten discharged capsules are at the Fuel Evaluation Facility (FEF) at EBR-II, Idaho and will be examined there in FY-73.

* Contributors are listed in Appendix B.

10.2.2.2 S/A X098A Capsules

The ten new L-16 capsules (instead of the originally planned 11) were incorporated into S/A X098A. These capsules have been selected to provide a broader base for the L-16 program. Nine (9) of these capsules contain a prime stabilized grade stainless steel Type-321H which was double vacuum melted material having been processed to optimize the creep properties for Type 321. Air melted Type 321H will also be included in these L-16B capsules as a minor variable. The remaining capsule was one of the original three L-16D capsules; it was a cold capsule containing sodium-exposed ring samples. The two hot capsules containing highly irradiated ring samples from previously irradiated cladding were not available for S/A X098A and will be incorporated into a later EBR-II MARK A 19 Subassembly. The identification and operating characteristics for the capsules in S/A X098A are shown in Table 10-1.

Table 10-1
CAPSULE IDENTIFICATION AND OPERATION CHARACTERISTICS
FOR L16 CAPSULES IN S/A 3 (X098A)

Capsule Identity	Drawing Identity	Operating Temperature		Rod Location ⁽¹⁾ In Subassembly	Condition ⁽⁴⁾	Alloy
		(°C)	(°F)			
LP2HA(3)	286E240-G3	525	977	15 (0.97)	Cold	321H
LP2AA(2)	286E240-G4	450	842	16 (0.96)	Cold	321H
LP2C (2)	286E240-G4	450	842	18 (0.97)	Hot	316H
LP2A (2)	286E240-G4	450	842	19 (0.95)	Hot	304H
LP2E (2)	286E240-G3	525	977	11 (0.98)	Hot	304H
LP2BA(2)	286E240-G3	525	977	12 (0.97)	Cold	321H
LP2G (2)	286E240-G3	525	977	14 (0.99)	Hot	316H
LP2GA(3)	286E240-G4	450	842	17 (0.98)	Cold	321H
LP2I (2)	286E240-G2	600	1112	3 (1.01)	Hot	304H
LP2CA(2)	286E240-G2	600	1112	6 (1.01)	Cold	321H
LP2K (2)	286E240-G2	600	1112	7 (0.99)	Hot	316H
LP2JA(3)	286E240-G1	675	1250	9 (1.01)	Cold	321H
LP2EA(3)	286E240-G2	600	1112	10 (1.00)	Cold	321H
LP2L(3)	286E240-G2	600	1112	13 (1.00)	Hot	(5)
LP2FA(3)	286E240-G2	600	1112	5 (1.02)	Cold	321H
LP2P (2)	286E240-G1	675	1250	1 (1.05)	Hot	304H
LP2DA(2)	286E240-G1	675	1250	2 (1.03)	Cold	321H
LP2R (2)	286E240-G1	675	1250	4 (1.04)	Hot	316H
LP4C (2)	816E123-G2	520	968	8 (1.03)	Cold	(6)

(1) Based on core position 4E2, number in parentheses is rod fission rate normalized to S/A average.

(2) Capsules (LP2A, LP2C, LP2AA), (LP2E, LP2G, LP2BA), (LP2I, LP2K, LP2CA), (LP2P, LP2R, LP2DA), and (LP4C) are five groups of capsules designed for nondestructive examination. These capsules will continue irradiation to maximum fluences of 3×10^{23} n/cm² or until saturation of clad swelling is established.

(3) These capsules are to be destructively examined and replaced by new capsules in later subassembly loading.

(4) Cold refers to new unirradiated capsules. Hot refers to irradiated capsules from X098.

(5) LP2L contains specimens from Types-347 and 321H and Incoloy-800 tubing which were irradiated in the earlier L-2 capsules to $\sim 7 \times 10^{22}$ n/cm².

(6) LP4C contains tube specimens exposed to 1300°F sodium for 15,000 hours.

Type 321H (L-16B) Capsules

Fabrication of twelve L-16B capsules containing Type-321H vacuum melted tubing was completed in time for insertion in S/A X098A as described above. The design of these capsules is identical to the earlier L-16B capsules.¹⁰⁻¹

L-16D Capsules

Encapsulation of only one of the original three L-16D capsules was completed in time for reconstitution into S/A X098A as described above. The remaining two hot inserts were shipped to FEF in Idaho for encapsulation, hopefully by the end of August 1972, in time for reconstitution of S/A X148A.

The series L-16 capsules experiment will continue in FY-73 under PA-10, Task 2, Fuel Rod Engineering.

10.3 CLAD ALLOY SWELLING

The objective of this program is to measure clad 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 clad swelling are liquid immersion density measurements, dimensional measurements (profilometry), and transmission electron microscopy (TEM).

10.3.1 Void Formation in Fuel Rod Claddings

Examination of void formation in experimental fuel rod claddings irradiated in EBR-II using transmission electron microscopy (TEM) has been completed. A tabulation of the TEM data obtained from cold-worked AISI Type 316, cold-worked and cold-worked plus aged Type 304 and solution annealed Type-321 stainless steel is presented along with data obtained from liquid immersion density and profilometry in Table 10-2. The chemical composition of each of these claddings is given in Table 10-3.

The data for the Type 304 and Type 316 claddings have been previously discussed.^{10-2,3} In summary, the cold-worked plus aged Type 304 exhibited void formation and consequent swelling comparable to annealed Type 304.¹⁰⁻⁴ TEM revealed that the aging subsequent to cold-work resulted in significant recovery with copious $M_{23}C_6$ precipitation on original slip bands. The regions between the bands of $M_{23}C_6$ exhibited a greater degree of recovery and also a greater propensity for void formation.

The 13% cold-worked cladding, E1H-29, exhibited moderate swelling. The void sizes are comparable to annealed Type 304, the difference in swelling accounted for by the reduced void number density. Void formation was heterogeneous and was present only in regions that appeared to have undergone some recovery.

The 50% cold-worked Type-316 cladding, F5Q, underwent a slight densification during irradiation indicating that recovery had occurred. TEM post-irradiation evaluation of this cladding revealed a substructure that had in fact undergone considerable recovery, but a high dislocation density was still retained. No voids were observed in the section irradiated at 625°C (1158°F) and only 10 voids were found in an entire TEM foil from the section irradiated at 526°C (480°F).

TEM micrographs typical of AISI Type-321 stainless steel irradiated to approximately 3×10^{22} n/cm² ($E > 0.1$ MeV-GE) at 527, 595 and 627°C (982, 1102, 1160°F) are presented in Figures 10-1 to 10-3 respectively. An attempt is made to present the void distribution in the (a) portion and the dislocation substructure in the (b) portion of each figure. There is a strong indication that Type 321 has a lower stacking fault energy than either Types 304 or 316 as faulted dislocation loops persist to at least 627°C (1160°F), Figure 10.3. Whereas, in Types 304 and 316 dislocation loops have unfaulted and only a dislocation network exists at similar temperatures.^{10-2,3} A bimodal void size distribution is indicated in the Type 321 cladding sections. The average void size is similar to that reported for annealed Type 316; however, a wider void size distribution and few voids are present.¹⁰⁻⁴ The voids found in section F5M-F were extremely irregular in shape (Figure 10-2) such that an estimate of the void volume fraction would be very inaccurate.

Table 10-2
RESULTS OF SWELLING MEASUREMENTS IN IRRADIATED COLD WORKED
AUSTENITIC STAINLESS STEEL FUEL ROD CLADDINGS

Specification Number	Material	Irradiation History			Density Change $\frac{\Delta\rho}{\rho}$, Percent
		Distance from Bottom of Core (in.)	Fluence 10^{22} n/cm ² E > 0.1 MeV	Temperature (°C)	
E1H-29-B	13% Cold Worked Type-304 SS	1.8 -2.8	4.7 ⁽⁵⁾	500	-0.85
E1H-29-H	13% Cold Worked Type-304 SS	10.9 -11.8	4.2	593	-0.21
E1H-30-B	13% Cold Worked + Heat Treated ⁽³⁾	1.9 -2.9	4.7	503	-4.32
E1H-30-H	13% Cold Worked + Heat Treated ⁽³⁾	10.9 -11.9	4.1	593	-0.51
F5Q-B ⁽⁶⁾	50% Cold Worked Type-316 SS	2.5 -3.5	3.1 ⁽⁶⁾	526	+0.07
F5Q-E	50% Cold Worked Type-316 SS	6.8 -7.8	3.3	592	+0.03
F5Q-I	50% Cold Worked Type-316 SS	11.8 -12.8	2.4	625	+0.07
F5M-B ⁽⁶⁾	Annealed Type-321 SS	2.8-3.8	3.1	527	-0.52
F5M-F	Annealed Type-321 SS	7.5 -8.5	3.2	597	-0.98
F5M-J	Annealed Type-321 SS	12.0 -13.0	2.3	627	-0.32

(1) Thickness based on stereo pairs.

(2) NE = Not evaluated.

(3) Heat Treated = 482°C for 24 hours, plus 704°C for 216 hours.

(4) VL = Very low; too low to obtain quantitative values.

(5) E1H rods were irradiated in Position 4B3 for 3313 MWd at 50 mw and 15,280 MWd at 62.5 MW.

(6) F5Q and F5M rods were irradiated in Position 4C3 for 12,099 MWd at 62.5 MW.

(7) Profilometry measurements.

Table 10-2 (Continued)

Specification Number	Voids				Dimensional Change $\frac{\Delta D^{(7)}}{D}$ (%)
	Volume ⁽¹⁾ Fraction (%)	Diameter (A°)		Number Density $\times 10^{-14} \text{ (cm}^{-3}\text{)}$	
		Mean	Range		
E1H-29-B	NE	—	—	—	1.2
E1H-29-H	0.15	365	100-750	0.5	0.8
E1H-30-B	2.9	350	80-900	11	2.60
E1H-30-H	0.45	350	< 64-1050	0.7	1.83
F5Q-B	VL ⁽⁴⁾	200	—	—	0.44
F5Q-E	NE	—	—	—	0.52
F5Q-I	0	0	0	0	0.24
F5M-B	0.30	260	65-600	2.3	0.52
F5M-F	NE	NE	~ 100-2000	NE	0.64
F5M-J	0.77	605	150-1120	0.49	0.30

Table 10-3
CHEMICAL COMPOSITION (wt %) OF
FUEL ROD CLADDING
FROM F5M, F5Q, E1H29 AND E1H30

Fuel Rod	F5M	F5Q	E1H29 & 30
Material	321	316	304
Heat Number	28967	85014	01973
Element			
C	0.06	0.04	0.066
Mn	1.83	1.90	1.62
P	0.019	0.021	0.024
S	0.011	0.006	0.018
Si	0.55	0.52	0.44
Ni	11.15	13.16	9.91
Cr	17.80	17.63	18.83
Mo	ND ⁽¹⁾	2.81	0.28
Cu	ND	0.15	0.13
Ti	0.49	ND	ND

(1) ND - Not Determined

A fine needle-like precipitate is present in this cladding as shown in Figure 10-3(a). The precipitates have not been identified but they do not appear to influence void formation. The grain boundaries are comparatively free of precipitates with only an occasional $M_{23}C_6$ particle present, Figure 10-2(b). The large particles present are believed to be TiC. Only an occasional precipitate particle in Type 321 was found to be associated with a void; whereas, in a sister fuel rod, E1H-26 (annealed 316), almost every carbide had a void associated with it.¹⁰⁻⁴

10.3.2 Elastic Moduli of Irradiated Stainless Steels

The effects of irradiation induced void formation on the elastic moduli of stainless steels have been evaluated using a dynamic resonance technique. Rings, approximately 0.25 inch high were cut from solution annealed Types 316 and 347 fuel rod claddings irradiated in the GE-F2 fuel rod test in EBR-II. These samples had volume increases of up to 10% as measured by the immersion density method. The results indicate that 10% swelling in Type 316 reduces the room temperature Young's Modulus by more than 20%.

Dynamic resonance techniques have proved to be extremely useful for the evaluation of the elastic and anelastic behavior of materials. By these techniques^{10-5, 6} the specimen, in the form of a simple geometric shape, is excited into vibration and the resonant frequencies for the different modes of vibration are used to calculate the various elastic properties of interest. The absolute accuracy of these techniques for measurement of elastic properties is unexcelled (to 0.1%), and changes with temperature can be measured to one part in 10^4 or better.¹⁰⁻⁵

Evaluation of the effects of neutron fluence and irradiation temperature on the elastic behavior of the materials requires that the sample volume be minimized, and preferably obtained from a position of uniform flux and temperature. For these reasons, ring-shaped specimens were cut from fuel cladding. A very local measurement of the Young's modulus of the irradiated material could then be obtained in as nearly uniform flux and temperature field as possible.

0.5 μm

0.5 μm

Figure 10-1. AISI TYPE-321 Stainless Steel Fuel Cladding Section F5M-B Irradiated to $3.1 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV-GE}$) at 527°C

0.5 μm

0.5 μm

Figure 10-2. AISI TYPE-321 Stainless Steel Fuel Cladding Section F5M-F Irradiated to $3.2 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV-GE}$) at 595°C

0.5 μm

1 μm

Figure 10-3. AISI TYPE-321 Stainless Steel Fuel Cladding Section F5M-J Irradiated to $2.3 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV-GE}$) at 627°C

10.3.3 Vibration of the Ring

For a circular ring-shaped specimen of rectangular cross section, the lowest resonant frequency is for a flexural mode of vibration with the flexure in the plane of the ring. The motion for the fundamental ring-flexure mode of vibration is schematically depicted in Figure 10-4. For that mode of vibration and its overtones, the Young's Modulus of the specimen is related to the resonant frequency and the physical parameters of the specimen as follows:

$$E = \frac{4\pi^2 \rho A r (1 + n^2) f^2}{\ln^2 (n^2 - 1)^2} \quad (10-1)$$

where

- E = Young's Modulus,
- ρ = the specimen density,
- A = the area of the ring cross section,
- r = the radius of the ring,
- f = the resonant frequency,
- I = the moment of inertia of the ring cross section, and
- $2n$ = the number of nodes for the harmonic of interest; $n = 2$ for the fundamental mode, $n = 3$ for the first overtone, etc.

10.3.4 Experimental Equipment and Technique

The experimental equipment for determination of the resonant frequency of the ring specimens is comprised of essentially two parts: (1) a set of electronic equipment including a signal generator, amplifiers, sensitive frequency counter, voltmeter and oscilloscope, and (2) a driver and pickup that convert the electrical signal (sine-wave) from the signal generator to a mechanical vibration and reconvert the mechanical vibration of the specimen to an electrical signal (see Figure 10-5).

In operation, the ring-specimen is suspended from the driver and pickup. The driver is oriented so that the vibration direction of the beam supporting the ring on the driver is in the radial direction of the ring and at 45 degrees to the horizon. The pickup is similarly oriented, but with the direction to which the pickup beam is sensitive to motion—at 90 degrees to the vibration direction of the driver beam. See Figure 10-5. The resonant frequency of the ring is determined by sweeping the appropriate frequency range with the signal generator. Only when the driving frequency is at or very near the resonant frequency of the ring is vibration of the ring excited. Resonance is detected as the frequency at which the ring vibrates with maximum amplitude. That frequency is determined to one cycle per second or better by counting with the frequency counter (1 in 30,000).

Figure 10-4. Fundamental Flexural Vibration Mode for a Ring

Figure 10-5. Schematic Diagram of Experimental Elasticity Equipment

10.3.5 Results

To demonstrate the proposed technique, three ring-shaped specimens were first investigated. The characteristics of those specimens are summarized in Table 10-4. Also listed are their resonant frequencies for the fundamental ring-flexure mode and the Young's Modulus calculated using Equation (10-1) for that mode ($n = 2$).

The measured values of the Young's Modulus are also compared with values reflected in the data for Table 10-4. The measured values for the specimens are slightly on the low side (to ~10%) as would be expected since the equation of motion used here has no correction for rotary inertia or extension of the centerline.¹⁰⁻⁷ These corrections would be expected to make a few percent increase in the predicted modulus. Also, the measured densities are somewhat low compared to the accepted values, which would also produce slightly low values of the absolute modulus. These corrections were determined for the actual cladding specimens by measurement of specimens of duplicate geometry and known moduli. This initial work also revealed that the accuracy of the modulus measurement would be critically dependent on the uniformity of all thickness of the specimen. Thus, the cladding rings were machined to obtain such uniformity and unirradiated control specimens, taken from archive material, were machined to wall thicknesses in the same range as the irradiated specimens (7 to 10 mils).

Table 10-4
CHARACTERIZATION OF SPECIMENS USED TO DEMONSTRATE TECHNIQUE

Sample Number	Material	o.d. (in.)	i.d. (in.)	Thickness (in.)	Weight (gm)	Density ^(a) (gm/cm ³)	Resonant Frequency (Hz)	Young's Modulus (10 ⁶ psi)	
								Measured	Literature Value
1	Type-304 SS	0.251	0.210	0.024	0.0447	7.655	35,863	25.40	28 ¹⁰⁻¹⁰
2	Type-304 SS	1.000	0.900	0.120	2.2379	7.625 ¹⁰⁻¹⁴	5,155	25.35	28 ¹⁰⁻¹⁰
3	Zircaloy-2	0.568	0.491	0.079	0.5175	6.241	9,875	12.39	12.5 - 13.7 ^{10-11, 10-12}

(a). Calculated from weight and dimensions.

Resonant frequency measurements for the ring flexural mode and dimensional measurements were made on all specimens and the Young's Modulus were calculated. The results are listed in Table 10-5 and shown in Figures 10-6 and 10-7.

For the Type-316 stainless steel, a least squares analysis of the effect of swelling on the Young's modulus gave the relationship:

$$E_{316} = 33.417 \times 10^6 [1 \pm 0.086 - (2.749 \pm 1.686) S] \text{ psi} \quad (10-2)$$

at the 95% confidence level for the coefficients where S is the volume fraction swelling. A similar analysis for the Type 347 specimens gave the relationship:

$$E_{347} = 34.988 \times 10^6 [1 \pm 0.145 - (1.575 \pm 2.745) S] \text{ psi} \quad (10-3)$$

Individual moduli values are shown in Figures 10-6 and 10-7 along with the uncertainty in those moduli based on the uncertainty in specimen wall thicknesses calculated from 6 to 10 measurements of those thicknesses.

Table 10-5
CHARACTERIZATION OF SAMPLES AND RESULTS

Samples ⁺⁺	o.d. (in.)	Wall Thickness (mils)	Density (gm/cc)	Swelling (%)	Resonant Frequency (cps)	Young's Modulus 10 ⁶ psi	Alloy*	$\sigma^{**}E$ (%)
F2V-C1	0.2516	7.7 ± 1.1 [†]	7.6354	3.95	12905.3	31.029	316	28
F2V-E1	0.2534	9.7 ± 1.0	7.3834	7.48	14813.9	25.236	316	20
F2V-F1	0.2555	9.2 ± 1.2	7.3651	7.76	13654.8	24.689	316	26
F2V-J1	0.2581	9.6 ± 0.6	7.3031	8.68	15198.4	28.935	316	12
F2X-D1	0.2503	6.7 ± 1.1	7.7159	2.86	11163.5	30.591	316	32
F2X-J1	0.2534	9.1 ± 0.7	7.4964	5.86	14409.6	27.679	316	14
F2X-M1	0.2517	9.0 ± 0.9	7.6770	3.37	14335.5	27.928	316	20
F2G-D1	0.2568	9.4 ± 0.6	7.4886	5.80	15392.4	31.144	347	12
F2G-H1	0.2593	8.3 ± 0.2	7.2007	10.02	13391.8	30.507	347	4
F2G-M1	0.2570	8.8 ± 0.6	7.5114	5.49	14662.5	32.601	347	12
F2H-7A2	0.2512	8.7 ± 0.6	7.8197	1.38	14384.8	30.480	316	12
F2T-9B1	0.2535	11.3 ± 0.7	7.7488	1.99	18087.5	28.768	347	12
F1	0.2482	6.9 ± 0.3	7.9014 ⁺	0.00	12818.0	37.571	347	8
F12	0.2481	9.7 ± 0.2	7.9014 ⁺	0.00	17979.0	36.494	347	4
F52	0.2481	7.1 ± 0.6	7.9288 ⁺	0.00	13154.0	37.376	316	17
F55	0.2485	10.3 ± 0.6	7.9288 ⁺	0.00	17900.0	32.247	316	11

* All specimens irradiated in solution annealed condition, control specimens solution annealed.

† Wall thickness uncertainties are for one standard deviation based on 6 to 10 measurements.

** Standard deviation of moduli in percent, based on uncertainty in wall thickness.

+ Unirradiated.

++ Detailed results from the post-irradiation examination of these specimens were presented in previous quarterly reports in this series, GEAP-10028-36, GEAP-10028-37, and GEAP-10028-38.

The data show a decrease in elastic modulus with swelling of the steels. The control specimen data revealed an apparent bias in the calculated moduli toward higher values. The absolute values of the moduli reported are therefore suspect; the difference in the true and calculated moduli is probably the result of neglecting rotary and lateral inertia in the calculation of the moduli from the resonant frequencies. However, the swelling dependence is thought to be faithfully represented by the data since the physical characteristics of the specimens were all very similar. That is, the first coefficients on the right hand side of equations (10-2) and (10-3) are *not* accurate but about 25% high, but the dependence on swelling is thought to be as accurate as the data allow, i.e.,

for Type-316 stainless steel

$$\left(E/E_0 \right)_{316} = [1 \pm 0.086 - (2.749 \pm 1.686) S]$$

and for Type-346 stainless steel

$$\left(E/E_0 \right)_{347} = [1 \pm 0.145 - (1.575 \pm 2.745) S]$$

at the 95% confidence level for the coefficients where the E_0 's are the moduli of the fully dense materials.

Figure 10-6. Effect of Swelling on Elastic Modulus of Solution Annealed Type-316 Stainless Steel.

Figure 10-7. Effect of Swelling on Elastic Modulus of Solution Annealed Type-347 Stainless Steel

The ratios of the Young's modulus of the materials after swelling to those of the fully dense materials are shown in Figure 10-8 with their 95% confidence intervals.

Analysis of the results shows that the major uncertainty arises from the wall thickness variation in the ring specimens. In order to better determine the effect of swelling on the elastic moduli, specimens with better dimensional control are required.

10.4 CLADDING ALLOY MECHANICAL PROPERTIES

The objective of this subtask is to evaluate the post-irradiation mechanical properties of available cladding alloy specimens, including precut tensile specimens 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 High Pressure Stress-Rupture Rig

The high pressure stress-rupture rig was developed to provide a facility for testing irradiated and unirradiated cladding tubing at higher pressure and temperatures closer to the actual irradiation temperature of Task F fuel rods. This facility was to provide stress-rupture data more representative of the as-irradiated fuel cladding by minimizing the thermal effects on cladding microstructure which occur when post-irradiation tests are performed at temperatures higher than the temperature of irradiation. Assembly and proof testing was completed.

Since all mechanical testing of irradiated fuel cladding has been terminated, this facility will be utilized in the stress-rupture testing of the Type 321H tubing being irradiated in the L-16 capsules. This work will continue under PA-10, Task 2, Fuel Rod Engineering.

Figure 10-8. Effect of Swelling on Elastic Modulus — Comparison of Types-316 and 347

10.5 MATERIALS MANAGEMENT

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

10.5.1 Type 321H Air-Melted Tubing

Three thousand feet of Type 321H air-melted tubing, including 2 lots of annealed and one lot each of 10% and 20% cold-worked tubing have completed incoming inspection and characterization. The four lots were from the same heat: 1065°C (1950°F) anneal, 1150°C (2100°F) anneal, 1065°C (1950°F) anneal plus 10% cold-worked and 1065°C (1950°F) plus 20% cold-worked. A few of the longer term stress-rupture tests will continue in FY-73. A report is in preparation which will present all the results on these tubing lots.

10.5.2 Vacuum Melted Type-321 Stainless Steel Tubing

Difficulty in ultrasonic inspection of the 1500 feet of vacuum melted AISI Type 321 tubing has been encountered. A new test standard has been manufactured from this material. It is believed that the use of this standard will reduce the background noise of the standard set-up, and will facilitate inspection.

10.5.3 Ti Modified 316 Stainless Steel Tubing

The order for 1000 feet of 0.250 inch o.d. tubing has been aborted due to a change in design for both fuel rod and materials experiments. The 0.230 inch o.d. tubing will be available through the LMFBR inventory maintained by HEDL.

10.5.4 Stabilized Grade of Austenitic Stainless Steel

The effect of various thermo-mechanical processes on 1300°F creep strength of AISI Type 321 stainless steels is being evaluated. Preliminary results indicate that the best treatment involves a combination of cold work age cold work.

REFERENCES

- 10-1. *Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, Thirty-Fifth Quarterly Progress Report, May-July 1970* (GEAP-10028-35).
- 10-2. Brager, H. R., Straalsund, J. L., Holmes, J. J., and Bates, J. F., *Met. Trans.* (1971), **2**, 1893.
- 10-3. Sandusky, D. W., Spalaris, C. N., and Wolf, U. E., *J. Nucl. Mat.* (1972), **42**, 133.
- 10-4. *Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, Forty-First Quarterly Progress Report, November 1971-January 1972* (GEAP-10028-41).
- 10-5. Wachtman, J. B., Jr., *Elastic Deformation of Ceramics and Other Refractory Materials*, Proc. of the Symposium on Mechanical and Thermal Properties of Ceramics, Gaithersburg, Md., 1968.
- 10-6. Marlowe, M. O., *Elastic Properties of Three Grades of Fine Grained Graphite to 2000°C*, 1970 (NASA CR-66933).
- 10-7. Flugge, W., *Handbook of Engineering Mechanics*, 61-19, McGraw-Hill, New York, N.Y., 1962.
- 10-8. *Sodium Cooled Reactors Fast Ceramic Reactor Development Program, Fortieth Quarterly Progress Report, August-October 1971* (GEAP-10028-40).

- 10-9. Marlowe, M. O., and Wilder, D. R., *Elasticity and Internal Friction Through the Kilocycle Range. Review and Annotated Bibliography*, 1964 (IS-925).
- 10-10. *Metals Handbook*, "Properties and Selection of Metals," Vol. 1, 422-423, 1961, ASM, Metals Park, Ohio.
- 10-11. Ibid, pp. 1228-1229.
- 10-12. Douglass, D. L., "Physical Metallurgy of Zirconium," *Atomic Energy Review*, 1, No. 4, IAEA, (1963).
- 10-13. Marlowe, M. O., and Kaznoff, A. I., "Elastic Behavior of Uranium Dioxide," *Ceramic Nuclear Fuels, Proc. of the International Symposium*, Washington, D.C. 1969.
- 10-14. Nienart, L. F., and Hasselman, D. P. H., "Measurement of Young's Modulus of Elasticity from Resonant Frequencies of Circular Ring Specimens," *Bull. Am. Cer. Soc.*, 49, pp. 999-1000 (1970).

11. TASK P—FUEL ELEMENT MODELING

Task Leader: F. E. Gelhaus*

11.1 GENERAL

The objective of Task P is the development of models 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:

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

11.2 PERFORMANCE CODE DEVELOPMENT

The documentation of the BEHAVE-2 fuel rod performance code (GEAP-13788) is approved for distribution, and the documentation of the BEHAVE-1 code (GEAP-13741) is receiving the final review for printing. The LIFE-2 performance code is in a production status on the H6000 system, the sample problem has been run, and linking to cut running time and cost is underway. The evolution of the LIFE-2 code at GE will reside in the Fuel Engineering Task of PA-10 in FY-73.

11.3 MODELING OF FUEL ROD E1H-26 USING THE BEHAVE-2 CODE

11.3.1 Power History

Fuel rod E1H-26 was irradiated in the EBR-II at core location 4B3 for 305.6 full power days between runs 26C and 42A inclusive. At an average linear power from 13.91 to 12.43 kW/ft, the fuel rod reached a peak burnup of 9.61 at. %, equivalent to 86,719 MWd/Te. The cladding was annealed Type-316 stainless steel. The fuel configuration was solid pellet with a smear density of 86.9% TD.

Initial modeling, using the steel swelling equation of Reference 11-1 and only a slowly varying power history, underpredicted the two main components of diametral increase—cladding swelling and plastic strain. Improvements were sought in the diametral increase predictions by investigating various mechanisms which would increase plastic strain and cladding swelling. Previous studies have indicated¹¹⁻² that power cycling, which includes a period on the order of weeks or months at intermediate power levels (~ 8 to 11 kW/ft) followed by a power increase, may cause plastic strain of the cladding under certain conditions. Accordingly, the EBR-II power history was searched for cycling behavior. Although the power history shows considerable cycling, there was none of significant duration at intermediate powers. The power history did indicate a 21-day period at reduced power which occurred 90 days after startup. The reduced power period had the correct parameters (8.75 kW/ft for 21 days) to produce plastic strain due to this effect.

11.3.2 Enhanced Cladding Swelling

The initial modeling had shown good agreement with measured cladding swelling at a point 2.6 inches from the fuel bottom but poor agreement at a point 8 inches above the fuel bottom. At the latter point, the measured $\Delta V/V$ was a factor of 2.35 larger than the result calculated by the computer code using the equation of Reference 11-1 for solution-treated Type-316 stainless steel. Large enhancements in swelling

Contributors are listed in Appendix B

of this alloy have been reported previously, most notably in the cladding of some rods irradiated in Dounreay Fast Reactor (DFR).¹¹⁻³ This enhancement of swelling has been commonly attributed to hydrostatic stress, but without convincing experimental or theoretical confirmation. It was suspected, and later confirmed, that the subject rod's cladding would see little fuel-cladding interaction in the simulation; over most of its lifetime the only stresses would be due to differential swelling and thermal expansion within the cladding itself. Since these latter stresses are present in all rods, they do not serve to differentiate rods having normal from those having enhanced swelling. Therefore, it was not reasonable to adopt the stress hypothesis in this case either.

As an alternative explanation, plastic strain has been involved in a past study¹¹⁻⁴ as the parameter against which the enhancement in swelling was correlated. For the purposes of the present study, a set consisting of data from 15 immersion density measurements from GE fuel rods clad in solution-treated Type-316 stainless steel was used to investigate this hypothesis. However, no useful correlation was found. The divergence in conclusions of this study and the previous study¹¹⁻⁴ regarding plastic strain enhancement is explained by the difference in the equations used to calculate $(\Delta V/V)_{\text{unenhanced}}$. The previous study used the HEDL " θ " equation while the equation used in the present study is based in part on data from pin F2V, the very pin modeled in the previous study. By using the data from F2V in deriving the correlation used¹¹⁻¹ to obtain $(\Delta V/V)_{\text{unenhanced}}$, the degree of enhancement predicted for that pin was naturally considerably reduced below that using the " θ " equation, and the remaining "enhancement" failed to correlate against strain. Furthermore, the high-temperature point from E1H-26 shows much more enhancement than any other sample while also showing less apparent plastic strain than most samples. Thus, while plastic strain can be used to correlate swelling enhancement for certain rods (when the θ equation is employed to define normal swelling), the concept cannot be used to explain the sample from E1H-26.

In view of the apparent inability of either stress or plastic strain to correlate the swelling enhancement at the high temperature end of the rod, other factors were investigated which might influence swelling. It is noted that stainless steel samples irradiated in the chemically isolated environment of a materials capsule apparently fail to show the effect. Fuel rod claddings, exposed inside and out to a variety of chemical agents *sometimes* show the effect. Therefore, it is logical to look for elements of the chemical environment which occur in some fuel rod experiments and not others.

Aitken and Bailly* have discussed the mechanism by which carbon, present as a fuel impurity in concentrations of 10 to 500 ppm, can be transferred to the cladding in the form of CO/CO₂. According to the authors, practically all of the carbon initially present in the fuel should, in theory, be transferred to the cladding inside surface and deposited in elemental form. Transfer of 100 ppm carbon impurity to the cladding is sufficient to increase carbon concentration in the latter material from 0.06 to 0.15%. If the added carbon were to diffuse throughout the cladding it would promote intragranular carbide precipitation. Incoherent precipitates are believed to enhance swelling.

Direct evidence of dispersal of the fuel impurity carbon throughout the cladding is not available. However, TEM examination of the high-temperature (573°C) specimen from E1H-26 has shown heavy precipitation of particles which have been identified¹¹⁻⁵ as M₂₃C₆. Furthermore, almost every precipitate particle is associated with a void.¹¹⁻⁵ The carbon content of the E1H-26 fuel is not available but it may be noted that this rod had considerably more cladding attack than its sister rod E1H-23 (4.6 vs 2.6 mils penetration), and E1H-26 also swelled more. If attack is associated with carbon content, the greater attack may indicate a higher carbon content in the higher swelling rod.

It has not been possible to develop a quantitative theory of swelling enhancement resulting from carbon as an impurity in the fuel. However, since the rate of carbide precipitation is probably very temperature dependent, the theory did supply a rationalization for tying the effect strongly to temperature in accord with the DFR data. Therefore, a simple temperature-dependent procedure was used to force agreement of the quantity calculated by the code with the immersion density results. It was noted in the profilometry

* Preliminary data, E. A. Aitken and W. E. Bailly, editors.

that, proceeding upward from the cold bottom end of the cladding, a rather sharp upturn in $\Delta D/D$ could be discerned 4.5 inches from the bottom of the fuel. The cladding midwall temperature at this point is $\sim 540^\circ\text{C}$. The model of cladding swelling was then modified so that for

$$T < 540^\circ\text{C} \\ (\Delta V/V)_{\text{modified}} = (\Delta V/V)_{\text{reference 11-1}}$$

$$T \geq 540^\circ\text{C} \\ (\Delta V/V)_{\text{modified}} = 2.34 (\Delta V/V)_{\text{reference 11-1}}$$

Use of a force fit to the observed cladding swelling then permitted consideration of other phenomena occurring in the fuel and cladding with an accurate portrayal of the cladding's swelling behavior preordained.

Figure 11-1 shows the forced agreement between calculated cladding swelling and the two measured points. The fact that the profilometry is above the calculated cladding swelling can be interpreted as an indication of fuel-induced plastic cladding strain. No significant source of this apparent plastic strain emerged from the present study. The power increase following the 21-day period of steady operation produced only a negligible amount (0.006%) of plastic cladding strain. A possible cause for the apparent plastic strain is the gas bottle effect which has been advanced^{11-4, 11-6} as a possible source of apparent plastic strain in rod F2G. F2G and E1H-26 were removed from the reactor at the same time and saw the same 22-day period of steady full power shortly before removal. According to the gas bottle hypothesis, fission gas is released from the center void of a high linear power rod to the plenum only when cracks form into the fuel interior during power changes. During periods of steady operation cracks heal up in the fuel interior. No escape pattern exists for fission gas and pressure may build up to a high enough degree to strain the cladding. The gas bottle effect was not modeled in the present study since it is not available in the current version of BEHAVE-2.

Figure 11-1. BEHAVE-2 Predictions and Measured Values of Diametral Change versus Axial Position—Model 1 on E1H-26

11.4 GAP CONDUCTANCE

Figure 11-2 compares predictions of radial crack fraction* distributions and fuel and cladding length increases to measured quantities. In this first comparison of measured crack fraction distribution to the code's predictions, agreement is very good in the inner 80% of the radius of metallographic Section F. In the outer 20% of the radius, measured crack fraction is much larger than the predicted values. The discrepancy may result from the difficulty of distinguishing cracks from fabricated porosity in the unrestructured region of this fuel with low fabricated density. In metallographic Section K, measured values are substantially greater than predictions. The two most likely explanations for the lack of agreement are that either metallographic preparation increased the apparent crack width over the true crack width by a factor of ~ 2 or that fuel temperature was much higher than calculated due to a low gap conductance. The 3-mil radial thickness of fission products which appears in the fuel-cladding gap in this section may be very porous and hence of low thermal conductivity which would yield a lower gap conductance than the 1500 Btu/h-ft²-°F employed in the BEHAVE-2 run. This latter interpretation is given strong support in a comparison of predicted and measured columnar grain growth radii. In Section F, where predicted and measured crack fractions are in good agreement, the predicted columnar grain growth radius divided by the outer fuel radius is 0.68, in agreement with the measured value of 0.75.

In Section K the predicted normalized columnar grain growth radius is 0.45, much lower than the measured value of 0.76. This tends to confirm the conclusion from the crack fraction measurements that the fuel temperature was much higher than calculated on the basis of a gap conductance of 1500.

The degradation of gap conductance at high burnup could explain the approach to 100% gas release which has been experienced in rods going to >8 to 10 at. % burnup. Mechanisms thus far proposed for gas release involve diffusion-controlled processes which are of insufficient magnitude below ~ 1300 to 1600°C for significant gas release. At a gap conductance of 1500, the fuel surface temperature is in the neighborhood of 800°C , leaving a wide ring of fuel seemingly too cool for gas release. If a gap conductance low enough to raise the surface temperature into the credible gas release zone can be demonstrated, the dilemma will be resolved. A large number of rods should be studied to investigate the potentially important effect of gap conductance degradation at high burnup.

11.5 AXIAL LENGTH CHANGES

Predicted cladding length increase is in reasonable agreement with the measured quantity, but fuel column length increase is significantly overpredicted. The predicted fuel column length increase of 2.1% is in much better agreement with the measured length increase of sister pin E1H-23 which had a 1.45% length increase than with the 0.32% measured for E1H-26. The most probable explanation is that the $\sim 90\%$ dense fuel column rapidly sintered in-pile to a shorter length while the $\sim 96\%$ dense fuel column of E1H-23 did not. French experiments have shown¹¹⁻⁷ a reduction in fuel length in a low density fuel that is not present in high density fuel. BEHAVE-2 does not currently treat fuel shrinkage due to sintering, except in the columnar grain growth region.

An additional BEHAVE-2 run was made to investigate both the underprediction of strain and the overprediction of fuel length change. Increased plastic strain might be predicted if the fuel length increase were reduced. The diminished fuel column length would cause a significantly smaller central void during power increases. The code's provision for simulating a mechanical fuel column restraint was employed for this purpose; the startup period was artificially extended to permit fuel creep in order to minimize axial cladding strain at startup due to the fuel pushing on the axial restraint. A reduced central void was produced by the axial restraint, and the desired matching of predicted-to-observed fuel column length increase was achieved on this run. However, the amount of plastic strain produced was negligible, as it had been previously.

* Radial crack fraction is defined as the volume fraction of a ring of fuel at a particular radius in the form of radially oriented cracks (cracks in the r,z plane).

Figure 11-2. BEHAVE-2 Prediction and Measured Value of Radial Crack Fraction versus Radius-Model 2 on E1H-26

11.6 CLADDING BEHAVIOR LIFETIME CODE DEVELOPMENT

11.6.1 Introduction

A plan has been formulated to develop a structural design code which will be capable of predicting the failure of fuel rod cladding. The main emphasis of the plan described is to evaluate the form required of such a design code. Various modeling approaches will be utilized initially and based on the results of this evaluation; the optimum modeling approach will be chosen to form the basis for the final design code. The code developed will be capable of being used with the Structural Design Criteria for Core Components.

The structural design code plan development and partial plan implementation to date have indicated that:

- The code should be capable of evaluating the following clad loading effects: a pressure at the inner boundary due to fuel swelling or fission gas; a temperature gradient through the wall; a temperature gradient across the rod; the effect of a localized hot spot; the effect of loading due to spacers; and the effect of a reduction in the cladding thickness due to wastages.
- The important material effects to be included are elasticity, thermal expansion, thermal creep, irradiation creep, swelling and plasticity.
- The required analytical techniques are well established and, consequently, the problem is to determine the optimum combination of techniques to form the basis of the code.
- Improved damage or failure criteria are an urgent requirement.

11.6.2 Clad Loading Mechanisms

Table 11-1 contains a list of possible clad loading mechanisms due to both steady state and transient operation together with their potential importance relative to cladding rupture. Included in the table as loads are all the mechanisms which cause a stress or a stress increase in the cladding. For the purposes of the following discussions all such effects are referred to as loads.

The structural design code to be developed will be capable of analyzing the interaction between the loading mechanisms designated "important." These loading mechanisms will include both primary and secondary stresses and, bearing in mind the low rupture strains which are anticipated, it is not evident that the inelastic strains which will occur in response to the secondary stress systems are unimportant. Typical estimates of the magnitude of the stresses which result under steady state conditions due to these loading mechanisms for a LMFBR hot rod are:

	ΔT (°F)	Elastic Thermal Stress (psi)
Across rod	35	~ 4500
Through wall	45	~ 7500
Hot spot	30	~ 7500

One of the purposes of the planned structural code will be to evaluate the potential of these stresses to interact with the primary stresses to give a ratcheting mechanism under non-steady state loading conditions.

The load effects in Table 11-1 categorized as "not known" contain those loading mechanisms which are not presently well-defined. Two of these effects (rod to wire and rod to grid loads) are amenable to design fixes. However, some analysis will be required to evaluate these effects. The other three loading cases categorized as "not-known" are special loading cases associated with the fuel/cladding interaction.

Table 11-1
LOADING MECHANISMS FOR FUEL ROD CLADDING LISTED
ACCORDING TO ESTIMATE OF POTENTIAL TO CAUSE CLADDING FAILURE

	Important	Not Known	Not Important
Pressure at inner boundary (fission gas or fuel/clad interaction loads)	X		
ΔT through wall ΔT across rod ΔT hot spot Axial temperature gradient Circumferential temp. gradient	cause stresses due to differential thermal expansion and swelling	X X X	X X
Clad thickness wastages (sodium corrosion, fretting, wear, and fuel/cladding intergranular attack)	X		
Rod-to-wire loads (including helical banding)		X	
Rod-to-grid loads		X	
Asymmetric fuel loads:			
Pellet discontinuity (ridging)		X	
Cracking		X	
Axial Loading of Clad by fuel		X	
End plug effects			X
Vibration			X

It is proposed that such effects should not be a requirement for the structural design code to be developed at this time, based upon the fact that there is insufficient experimental evidence to prove that these effects are major contributors to mixed oxide LMFBR fuel rod failures. In addition, even if such effects could be structurally modeled, the large uncertainties associated with fuel behavior would render any conclusion not supported by experimental evidence highly speculative; such modeling is the major objective of the LIFE code development.

The loading effects in Table 11-1 categorized "not important" will not at this time form any requirement on the structural design code. Of the four loads in this category, two (axial and circumferential temperature gradients) give quite small stresses which may be neglected and the other two are amenable to design fixes.

11.6.3 Material Effects to be Considered

The material behavior effects which are important in determining the behavior of the cladding are:

- elasticity.
- thermal expansion,
- thermal creep,
- irradiation creep,
- swelling, and
- plasticity.

All these material effects will be included. However, in particular cases it may be possible to neglect certain material effects, e.g., plasticity where the stresses remain below yield and thermal creep where the temperature is sufficiently low.

It is also desirable that the code developed should treat plasticity according to the rules of kinematic hardening. This is particularly important when on-off power cycles are considered and it is suspected that yielding may occur due to the reversal of stresses on going off-power. With regard to thermal creep, it is desirable that the assumptions on creep hardening can be varied to determine the sensitivity of the results of this effect.

11.6.4 Types of Structural Model

Structural analyses techniques are well developed and analytical capability is not considered to be a limitation on the development of the structural code. The problem is singularly that of development of an analytical procedure which best meets all the requirements placed on the code including those of accuracy and cost.

Table 11-2 summarizes the relative capability of four structural models which might be used to predict fuel rod cladding behavior. The four codes considered reflect three basic levels of sophistication in modeling. The simplest type of code is typified by a code such as STATUS which was earlier developed by Task P. STATUS accepts as input the average strains and temperatures which are predicted to exist in the cladding using a fuel behavior code such as LIFE. STATUS then superimposes on these strains the strains which would result from other locally applied loads. Obviously the super imposition of non-linear effects is an approximate procedure. The errors introduced in this way are unknown. One of the aims of the plan is to evaluate the magnitude of such errors by comparing the predictions made by STATUS with those made by more rigorously based analytical codes.

Table 11-2
RELATIVE CAPABILITY OF DIFFERENT STRUCTURAL MODELS

	STATUS	2 DIMENSIONAL	2 DIMENSIONAL	3 DIMENSIONAL
	Simplified Superposition of Effects	Closed Form Generalized Plane Strain	Finite Element Generalized Plane Strain or Axisymmetric	Finite Element Generalized Plane Strain or Axisymmetric
Pressure at Inner Boundary	1*	1	1	1
ΔT through Wall	2	1	1	1
ΔT across Rod	2	1	1	1
ΔT Hot Spot	2	1	1	1
Axial Temperature Gradient	2		1	1
Circumferential Temperature Gradient	2	1	1	1
Clad Thickness Wastage	2	1	2	2
Rod-To-Wire Loads			1	1
Rod-To-Grid Loads			1	1
Asymmetric Fuel Loads				2
Axial Loading of Clad by Fuel	2	2	2	2
End Plug Effects			1	1
Vibration				
Elasticity	2	1	1	1
Thermal Expansion	2	1	1	1
Thermal Creep	2	1	1	1
Irradiation Creep	2	1	1	
Swelling	2	1	1	
Plasticity	2		1	1
Relative Cost	\$	\$\$	\$\$\$	\$\$\$\$

* 1 — correctly modelled

2 — approximately modelled

The middle two columns in Table 11-2 describe existing two-dimensional codes. The first 2-DIM code utilizes what is essentially a closed form solution of the generalized plane strain problem. Such a code is capable of analyzing the "important" loading effects and includes all the main material phenomena except plasticity. Its technical advantage over other codes is its ability to correctly model the typical wastage which occurs in cladding thickness. It is also attractive due to its relative low running costs. The second two-dimensional code described is CREEP-PLAST which utilizes a finite element approach and is, therefore, somewhat more versatile. The total capability of CREEP-PLAST as defined in Table 11-2 is the result of using the code both in the generalized plane strain and the axisymmetric form. Not all the load cases identified in Table 11-2 can be considered at once. The net result is that, by using both modeling alternatives and then superimposing the results, CREEP-PLAST can approximate the full range of three dimensional loads applied to the cladding. This range of capability makes CREEP-PLAST an attractive tool for evaluating a variety of loading effects.

Where axial loading effects and their interaction with those occurring due to circumferential variations in load become important, then a truly three-dimensional capability becomes necessary. The fourth column in Table 11-2 describes the attributes of a typical three-dimensional, finite element code such as MARC IV. No irradiation creep or swelling capability is indicated, but the inclusions of such effects poses no real technical problem. The modeling of the loads which result due to the fuel axial movement or asymmetry is shown as being approximate, reflecting the difficulties in defining such material effects for the fuel rather than any shortcoming of such a code with known loads. One drawback is the inability to accept changes in the cladding thickness continuously with time. With a finite element code, such changes in geometry can only be approximated.

11.6.5 Fuel-Cladding Interface

Present experimental experience and analytical predictions of typical LMFBR fuel rods irradiated in the EBR-II would seem to indicate that some fuel-cladding interaction will occur in the region of 5 to 7 at. %. Above 7 at. %, it appears that the cladding swells at a rate in excess of the fuel and a fuel-cladding gap opens up. It is desirable that the structural code be able to account for this potentially important interface. One approach is to use directly the output of LIFE as input to the clad structural code. Such output could either be in the form of the operational history of pressure or of radial displacement at the inner boundary of the cladding. The use of such a pressure-time history would be successful where the additional loads applied to the cladding in the structural code did not change the average radial displacement. The radial displacement-time history would be expected to work best in cases where the fuel is significantly stiffer than the cladding. If no fuel-cladding interaction occurs, then the pressure at the inner boundary of the cladding is simply that of the fission gases.

11.6.6 Failure Criteria

Typical failure or damage criteria presently used by fuel rod designers are:

1. An overall limit on inelastic strains;
2. A summed strain fraction for inelastic strains of the form

$$\sum \frac{\epsilon}{\epsilon_R} = 1,$$

where ϵ_R is the rupture strain for the conditions prevailing during the time increment of interest;

3. A summed life fraction based on stress-rupture data

$$\sum \frac{t}{t_R} = 1,$$

where t_R is the time to rupture at the stress level associated with t ; and

4. A fatigue damage fraction of the form

$$\sum \frac{N}{N_f} = 1,$$

where N_f is the number of cycles to failure associated with the conditions at which N cycles occur.

These criteria will be included in the structural code. However, before any code can be judged fully acceptable, the failure criteria must be supported by statistically meaningful experimental data. The code will be a basic tool to be used in the application and evaluation of Design Criteria.

11.6.7 Plan

The plan to proceed is described very generally by the block diagram shown here.

Under the heading of "evaluation of structural models," a selection of sample cases will be run (Table 11-3) using STATUS, CREEP-PLAST and the other two-dimensional code defined in Table 11-2. It is also intended that some cases will be run using a three-dimensional code such as the CDC MARC IV program. By comparing results it should be possible to determine the adequacy of the various codes to correctly analyze a complex problem.

These codes will also be used in evaluating different loading effects, material properties, assumptions, and the problems associated with the fuel-cladding interface. Table 11-3 outlines a number of test cases which could be run with the two dimensional codes for the purpose of evaluating the various effects noted in the table. The extension of these sample cases would be influenced by the knowledge gained from the early results.

Table 11-3
MATRIX OF SAMPLE CASES TO BE RUN WITH 2 DIMENSIONAL CODES
USING HOT SPOT OF HOT ROD IN LMFBR

Condition at Inner Boundary of Cladding	Wall	Rod	Hot Spot	Wastage	Point Load	Power	Plane Strain Mode	Axisymmetric	Purpose
Fission Gas Pressure	x			x		100% Steady State	x		Compare results with status prediction
Fission Gas Pressure	x	x	x	x		100% Steady State	x		
Fission Gas Pressure	x	x		x		70-110% Cycles	x		Evaluate thermal ratcheting under normal power variations
Fission Gas Pressure	x	x	x	x		70-110% Cycles	x		
Fission Gas Pressure	x	x		x		0-100% Cycles	x		Evaluate reserved plasticity on unloading
Fission Gas Pressure	x	x	x	x		0-100% Cycles	x		
Fission Gas Pressure	x				x	Steady & Variable	x		Superimpose results and compare with 3-dimensional predictions to evaluate importance of this effect
Fission Gas Pressure	x				x			x	
Fission Gas Pressure	x	x	x	x		100%	x		Compare results of both 2-dimensional codes to evaluate importance of the correct modelling of a continued varying wastage allowance
Fission Gas Pressure	x	x	x	x		100%	x		
Pressure/Time History From Fuel Code For Fuel/Clad Interaction Case	x	x	x	x		100% Power	x		Compare the radial cladding displacements in the two codes and check for discrepancies to determine if this interfacing procedure is adequate
Pressure/Time History From Fuel Code for Fuel/Clad Interaction Case	x	x	x	x		100% Power & Transients	x		

11.6.8 Progress to Date

The STATUS code was developed as part of Task P and is available for modeling cladding strains. It is intended to further evaluate on a quantitative basis the usefulness of this tool and the principle of superposition.

CREEP-PLAST, a finite element, elastic, plastic, creep code with a generalized plane strain and axisymmetric modeling capability, was identified as a suitable tool. This code has already been modified to include both swelling and irradiation creep effects and sample cases have been run to test the correctness of the new subroutines. Although some success has been achieved, it appears that there are still some minor problems associated with the running of the modified code for certain types of problems.

In addition to CREEP-PLAST, another two-dimensional, generalized plane strain code has been developed. This code has the capability indicated in column 2 of Table 11-2. The solution procedure is essentially closed form, although numerical integration is necessary to evaluate certain integrals. The code is in general more limited than CREEP-PLAST but, as a tool specifically designed for analyzing cladding behavior, it has some distinct advantages. In particular, the resolution of individual strain components and the treatment of cladding wastage allowances are both superior to that which may presently be achieved using CREEP-PLAST. Also the code is more efficient in its handling of loading cycles. However, it should be noted that in its present form it does not include plasticity effects. Two sample problems have been successfully run using this code, but additional work is required to fully evaluate its capabilities.

The CDC MARK IV and the NASTRAN-general purpose finite element codes have been reviewed relative to their ability model three-dimensional cladding performance. It was concluded that the CDC MARC IV code was more suited than the NASTRAN code for the inelastic analysis of fuel rods. Other general purpose finite element codes, such as ANSYS, will be included in further reviews to determine which code is best suited to fuel rod analysis.

REFERENCES

- 11-1. Bates, J. F., and Straalsund, J. L., *A Compilation of Data and Empirical Representations of Irradiation-Induced Swelling of Solution Treated Types-304 and 316 Stainless Steel*, WADCO Corporation, September 1971 (HEDL-TME 71-139).
- 11-2. *Sodium Cooled Reactors Fast Ceramic Reactor Development Program, Fortieth Quarterly Report, August-October 1971*, November 1971 (GEAP-10028-40), p. 12-1.
- 11-3. FR/N (70)6, U.K. Fast Reactor Newsletter, December 1970.
- 11-4. *Sodium Cooled Reactors Fast Ceramic Reactor Development Program, Fortieth Quarterly Report, August-October 1971*, November 1971 (GEAP-10028-40), pp. 12-3 to 12-5.
- 11-5. Busboom, H. J., "Task L Monthly Report—February 1972."
- 11-6. *Sodium Cooled Reactors Fast Ceramic Reactor Development Program, Thirty-Seventh Quarterly Report, November 1970-January 1971*, March 1971 (GEAP-10028-37), p 87.
- 11-7. Mikailoff, H., et al., "Irradiation Results of Mixed UO_2 -PU O_2 Test Pieces for Fast Reactor Fuel Elements," EURFNR-707 Report CEA.R.3878, EURATOM-C.E.A. Association Fast Neutrons (October 1969).

12. QUALITY ASSURANCE

12.1 SUMMARY OF QA ACTIVITIES

The PA-10 Plan was revised in response to RDT comments and distributed for internal BRD review June 30, 1972. Issue is expected during September, 1972. Additional procedures are in preparation for implementation of requirements identified in the plan where it is determined that existing procedures are inadequate.

A presentation of the scope and planning for PA-10 FY-73 activities was made by BRD to RDT headquarters. This presentation included the status of the QA program and planned activities for FY-73.

A series of introductory codes and standards seminars was begun. The intent of this program is to provide engineers training and orientation to the RDT standards system. Also, a series of seminars was initiated to acquaint managers and engineers with the classification system for functional use and for design characteristics. The classification system will provide a means of integrating program management planning, engineering planning, and quality assurance planning for establishing the quality level of effort.

A detailed investigation of L-16 capsules, including hardware, documents, and records, was performed as a result of shipping problems (described in the Problems, Nonconformance, and Failures section of this report). Satisfactory resolution of quality-related problems was accomplished, and the capsules were released for shipment to EBR-II.

Planned Quality Engineering review of engineering documents and preparation of surveillance planning, as well as Quality Surveillance of inspection and verification of fabrication and test activities, occurred on active PA-10 tasks during this period.

12.2 PROBLEMS, NONCONFORMANCES, AND FAILURES

Capsules L16B and L16D were shipped to EBR-II without the required QA release. Unresolved discrepancies and incomplete quality records had prevented QA release prior to shipment. However, the QA organization did initiate action to retrieve the shipment before reaching EBR-II. The capsules were returned to BRD for management and quality review to determine their disposition. Inadequate shipping control procedures were the cause of the occurrence. Interim procedures were established as corrective action with comprehensive procedures to be prepared to prevent similar recurrence.

Nonconforming item activity during this period included seventeen discrepancy reports.

One report covering capsule B11C required first level acceptance of blanket insulator pellets whose impurity levels exceeded specification limits. Engineering investigation revealed that the impurity levels would not affect the type of testing planned for the capsule.

Four reports covering capsule F11A required rework of discrepant dimensions, and one report required Material Review Board (MRB) acceptance of a non-mating, out-of-tolerance dimension. Three of these reports covering capsule F11A required first level acceptance of miscellaneous fuel pellet discrepancies (cracks exceeding specification, impurity levels exceeding specification, and dimensions differing from specification). Engineering investigation showed that these discrepancies would have no effect on the tests planned for the capsule.

Two reports covering capsule L16B required rework of discrepant dimensions; one report required MRB evaluation of the acceptability of flux wires which had traceable inadequate materials, and one report required MRB decision to accept the bottom cap after MRB investigation determined that it conformed with the latest ANL drawing.

Two reports covering the G-6 water-flow fuel assembly required the MRB to determine disposition of fabrication discrepancies (grids dislodged during assembly). Final decision was to scrap the assembly and salvage usable components.

One report covering F9E bottom end plug required first level disposition to accept disorientation of the identification flat. The flat is nonfunctional in assembly or operation. Another report covering F9E inner channel assembly required MRB disposition to determine adequate criteria for spacer retention to judge acceptance of the assembly.

12.3 PROGRAM AUDITS AND MANAGEMENT REVIEWS

There were no scheduled internal audits during this period.

Informal engineering reviews of L16 capsules were scheduled between QA and task principals as a result of the investigation described in the Summary Section of this report.

An audit was performed at EBR-II of the F9E assembly operation preparatory to loading in the reactor in order to provide BRD release for insertion.

A design review was held with BRD Engineering and Quality Assurance participation, on the C11A capsule at the GE-Irradiation Processing Operation, which is responsible for the design and fabrication of the capsule under contract to BRD.

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 June 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

GEAP-10028-42
Forty-Second Quarterly Report
February–April 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
for Analyzing the Reactor Response
to Small Reactivity Oscillations;
P. Greebler, B. A. Hutchins, and M. D. Kelley
(March 22, 1965)

GEAP-4630

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

GEAP-4675

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

GEAP-4770

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

GEAP-4787

An Analysis of Fast Reactor Transient Response and Safety in Selected Accidents;

W. Meinhardt, Jr.

(August 1966)

GEAP-4804

Experimental Studies of Transient Effects in Fast Reactor Fuels, Series II;

J. E. Hanson, J. H. Field, and S. A. Rabin

(June 1965)

GEAP-4809

Fast Reactor Meltdown Accidents Using Bethe-Tait Analysis;

R. Meyer and B. Wolfe

(January 1957)

GEAP-4819

Experimental Studies of Transient Effects in Fast Reactor Fuels, Series III, Pre-Irradiated Mixed Oxide ($\text{PuO}_2\text{-UO}_2$) Irradiations, Interim Report, Steady State Irradiations;

J. E. Hanson and S. A. Rabin

(September 1965)

GEAP-4822

Studies of Alternate Core Design Concepts for Fast Sodium Cooled Reactors;

C. L. Cowan

(November 1966)

GEAP-4870

Some Design and Irradiation Experience with Capsules for Long Ceramic Fuel Elements;

F. W. Knight

(July 1966)

GEAP-4877

Diffusion of Plutonium and Uranium in Oxide Fuel at 2400°C ;

J. Davies and P. Novak

(August 1966)

GEAP-4878

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

W. L. Lyon and W. E. Baily

(May 1966)

GEAP-4879

Thermal Conductivity of Uranium and Uranium Plutonium Oxides;

F. J. Hestzler, T. E. Lannin, K. J. Perry and E. L. Zebroski

(January 1968)

GEAP-4897

Irradiation Behavior of Hyper- and Hypo-Stoichiometric Plutonia-Urania Fuel at $70,000\text{ MWd/Te}$;

E. L. Zebroski, W. E. Baily, and D. P. Hines

(March 1965)

GEAP-4922

BISYN—

A Two Dimensional Synthesis Program;

M. D. Kelley

(December 1966)

GEAP-4939

Safety and Economic Characteristics of a 1000 MWe Fast Sodium Cooled Reactor Design;

K. P. Cohen and G. L. O'Neill

(April 1967)

GEAP-5271

Implications of Recent Fast Critical Experiments on Basic Fast Reactor Design;

P. Greebler

(March 1968)

GEAP-5272

Evaluation and Computation of Pu-239 Cross Section Data for the ENDF/B Files;

P. Greebler, P. Aline, B. Hutchins

(January 1967)

GEAP-5273

FORE II, A Computational Program for the Analysis of Steady-State and Transient Reactor Performance;

B. E. Lawler, et al.

(September 1967)

GEAP-5340

Neutron Spectrum Measurements in a Fast Plutonium Assembly ZPR-III Assembly 48;

P. S. Brown and A. Weitzberg

(April 1967)

GEAP-5344

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

GEAP-5355A

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

GEAP-5411

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

GEAP-5493

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

GEAP-5494

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

GEAP-5556

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

GEAP-5583

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

GEAP-5633

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

GEAP-5635

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

GEAP-5683

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

GEAP-5744

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

GEAP-5748

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

GEAP-5761

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

GEAP-5570

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

GEAP-10046

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

GEAP-10052

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

GEAP-10054

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

GEAP-10062

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

GEAP-10066

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

GEAP-10385

High Burnup Performance of Powder & Pellet Mixed-Oxide Fuel;
K. J. Perry, W. E. Bailly
(July 1971)

GEAP-10403

Axial Expansion of Mixed-Oxide Fuel Measured In-Pile;
D. A. Cantley, G. R. Hull, C. N. Craig
(July 1971)

GEAP-13515

The Melting Point of Plutonia-Urania Mixed Oxides Irradiated to High Burnup;
J. L. Krankota, C. N. Craig
(July 1969)

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
(September 1969)

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;
T. A. Lauritzen, A. Montes, A. Conti
(February 1970)

GEAP-13551

Effects of Nuclear Methods Approximations on LMFBR Analysis;
R. Protsik
(August 1969)

GEAP-13553

An Evaluation of the Quality of Seamless Austenitic Stainless Steel Tubing;
T. A. Lauritzen
(December 1969)

GEAP-13556

BICYCL - A Computer Code for Two-Dimensional Analysis of Equilibrium Fuel Cycles;
P. C. Vaughn
(December 1969)

GEAP-13562

Transient Overpower Irradiation of Axially-Restrained, Zero-Burnup Fast Reactor Fuel Specimens;
G. R. Thomas
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
(November 1970)

GEAP-13582

Summary of (U,Pu)O₂ Properties and Fabrication Methods (November 1970);
B. F. Rubin
(February 1971)

GEAP-13591

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

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

GEAP-13592

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

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

GEAP-13599

FUMBLE: An Approach to Fast Power Reactor
Fuel Management and Burnup Calculations;

P. Greebler
C. L. Cowan
(November 1970)

GEAP-13603

Center Temperature Measurements of Mixed Oxide
Fuel Zero to 3,000 MWd/Te;

R. R. Asamoto, P. E. Bohaboy,
D. W. Sandusky, A. E. Conti
(March 1971)

GEAP-13608

THE FREADM CODE: A Fast Reactor
Excursion and Accident Dynamic Model;

D. Freeman, et al.
(September 1970)

GEAP-13620

Failed Fuel Performance in Naturally Convecting
Liquid Metal Coolant;

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

GEAP-13637

Steady State and Defected Fuel Irradiation
Testing Needs;

W. G. Meinhardt
(March 1971)

GEAP-13639-1, -2

Core Design Development Needs in Relation to
Fuel Failure Propagation, Sodium Boiling and
Clad/Fuel-Sodium Thermal Interaction;

Compiled by L. M. McWethy, P. R. Pluta, D. B. Sherer
(October 1970)

GEAP-13672

SYN--Operational Aspects of the Combined
BISYN, PERT, and BICYL Synthesis Codes;

M. D. Kelley
(April 1971)

GEAP-13686

Irradiation Induced Swelling Rates of $\text{PuO}_2\text{-UO}_2$
Fuel with Strong Radial Restraint;

R. C. Nelson, et al.
(March 1971)

GEAP-13703

Effective Thermal Conductivity of $(\text{Pu,U})\text{O}_2$ at
High Burnups (EIE/F);

C. N. Craig
(March 1971)

GEAP-13704

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

B. A. Hutchins, C. L. Cowan, M. D. Kelley, J. E. Turner
(March 1971)

GEAP-13722

Performance Tests of Sodium-Filled Powder &
Pellet-Oxide Fuel Under Transient Overpower;

T. Hikido, J. H. Field
(July 1971)

GEAP-13729

$(\text{Pu,U})\text{O}_2$ --Sodium Reactions: Effect of Fuel Parameters;

D. E. Plumlee and P. E. Novak
(September 1971)

GEAP-13730

Effect of Stoichiometry on the Behavior of Mixed-Oxide
Fuel During Extended Operation in Failed Pins;

R. E. Murata, C. N. Craig, H. C. Pfefferlen, P. E. Novak
(October 1971)

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).

GEAP-13788

BEHAVE-2: Oxide Fuel Performance Code
in Two Spatial Dimensions and Time;
S. Oldberg
(February 1972)

Appendix B LIST OF CONTRIBUTORS

Task B—Defected Fuel Behavior

Task Leader: C. N. Craig

Contributors:

P. E. Bohaboy	K. E. Gregoire	D. E. Plumlee
C. W. Burger	R. E. Herbst	J. J. Regimbal
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J. H. Cherb	C. R. McMillan	M. L. Thompson
J. R. Crepeau	R. E. Murata	J. D. Wilson
W. L. Conant	P. E. Novak	D. L. Zimmerman
W. K. Eggert	G. E. Payne	

Task C-1—Safety Experimentation (Transient Overpower)

Task Leader: G. R. Thomas

Contributors:

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R. T. Fernandez	P. E. Novak	G. L. Stimmell
R. E. Herbst	G. E. Payne	R. G. Stuart
T. Hikido	H. C. Pfefferlen	D. M. Switick
M. M. Idota	M. C. Plummer	R. C. Thompson
R. P. Johnson		

Task C-2—Safety Experimentation (Loss-of-Flow)

Task Leader: H. C. Pfefferlen

Contributors:

F. L. Bamford	T. L. Gregory	J. L. Scofield
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