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NOVEMBER 1971

MASTER

SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

FORTIETH QUARTERLY REPORT
AUGUST-OCTOBER 1971

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**

Fortieth Quarterly Report

August-October 1971

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LIST OF ABBREVIATIONS, SYMBOLS, AND ACRONYMS

$\text{\AA}$	=	angstroms
AEC-RDT	=	Atomic Energy Commission—Division of Reactor Development and Technology
ANL	=	Argonne National Laboratory
Ann.	=	annealed
ARC	=	Argonne Reactor Computation (System)
at. %	=	atom percent
bcd	=	binary coded decimal
BEHAVE	=	fuel modeling code
BOL	=	bulk outlet
BUBBLE	=	coolant boiling and channel voiding code, a subroutine of FREADM
cal/gm	=	calories per gram
cm	=	centimeters
co-ppt	=	coprecipitated
C.W.	=	cold worked
CSEWG	=	Cross Section Evaluation Working Group
DEMON	=	diffusion theory flux solution computer code
DFR	=	Dounreay Fast Reactor
DRL	=	Division of Reactor Licensing (AEC)
EBR-II	=	Experimental Breeder Reactor II
ELC	=	Extra Low Carbon
ENDF/B-II	=	Evaluated Nuclear Data File/B, version II
ENDRUN-2	=	computer code for cross section processing version 2
E.o.L.	=	end-of-life
ETOE-MC ²	=	computer codes for cross section processing
FCI	=	fuel-coolant interaction
FFTF	=	Fast Flux Test Facility

FORTRAN	=	standard computer programming language
FREADM	=	Fast Reactor Excursion and Accident Dynamics Model
ft/sec	=	feet per second
FUMBLE	=	fuel management computer code for startup-up cycles
gm/cc	=	grams per cubic centimeter
g-cm	=	grams per centimeter
gm/mole	=	grams per mole
gpm	=	gallons per minute
GETR	=	General Electric Test Reactor
gm	=	grams
GMUG-2	=	Multigroup data file based on modified ENDF/B-II data
h_{gap}	=	fuel cladding gap conductivity
HEDL	=	Hanford Engineering Development Laboratory
i.d.	=	inside diameter
in./in./h	=	inches per inch per hour
I.P.	=	integrated power
kbar	=	thousands of atmospheres
kcal/mole	=	kilocalories per mole
k_{eff}	=	effective neutron multiplication factor
keV	=	kilo electron volts
ksi	=	thousands of pounds per square inch
kW/ft	=	kilowatts per foot
ℓ	=	angular momentum states
lb _f	=	pounds force
lb _m	=	pounds mass
LMFBR	=	Liquid Metal Fast Breeder Reactor
MARS	=	Computer code for core disassembly calculation

MeV	=	million electron volts
MeV-GE	=	General Electric adjusted fluence for EBR-II ⁽¹⁾
mg	=	milligram
msec	=	millisecond
mole	=	gram molecular weight
MWd/Te	=	megawatt-day per ton (metric, tonne, 2205 lb)
nvt	=	time integrated neutron flux
o.d.	=	outside diameter
%Δk	=	percent reactivity addition
%TD	=	percent of theoretical density of fuel
P _L	=	linear power
P _{max}	=	peak power
PISCES	=	Physics International multi-Spatial Code for Engineering and Science
PNL	=	Pacific Northwest Laboratory
PEC	=	Pulsed Eddy Current (analyzer)
ppm	=	parts per million
PPS	=	plant protection system
psi	=	pounds per square inch
psia	=	pounds per square inch (absolute)
psig	=	pounds per square inch (gauge)
PUNT	=	service code for maintaining GMUG data files
R	=	Roentgen, unit of radiation dosage
RABBLE	=	Computer code for special resonance self-shielding homogeneity
SA or S/A	=	subassembly
SEFOR	=	Southwest Experimental Fast Oxide Reactor
SS	=	stainless steel
(ST)	=	solution treated

STP	=	standard temperature and pressure
SUR	=	strong unbroken ring
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
THTD	=	transient heat transfer code (version D)
THTE	=	transient heat transfer code (version E)
TOFF	=	transient overpower fuel failure model
TREAT	=	Transient Reactor Test Facility
TRIAL	=	Computer code for testing accuracy of Three-Dimensional (TRI) (A)xial (L)inking
UTS	=	Ultimate Tensile Strength
VENUS	=	disassembly calculation code
VERA-11	=	critical assembly 11A in VERA facility
VNC	=	Vallecitos Nuclear Center
VOIDUM	=	systems hydraulics model, a subroutine of FREADM
vol %	=	volume percent
W/cc	=	watts per cubic centimeter
wt %	=	weight percent
watts/cc	=	watts per cubic centimeter
XPORT	=	a computer routine to prepare code for shipment to outside users
ZEBRA-3	=	critical assembly 3 in ZEBRA facility
ZPPR-2	=	critical assembly 2 in Zero Power Plutonium Reactor facility
ZPPR-3	=	critical assembly 3 in Zero Power Plutonium Reactor facility
ZPR-6	=	Zero Power Reactor facility 6
μm	=	micrometers

ABSTRACT

This report is the fortieth 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 August through October 1971 is described for tasks related to defected fuel testing, safety experimentation, fuel materials properties, fuel testing in EBR-II, core design analysis, reactor safety analysis, cladding development, physics, and fuel element design analysis.

HIGHLIGHTS

- A second loss-of-flow test in the out-of-pile Mockup 1B facility was successfully completed with all three fuel simulators operating for a significant period of time following initial voiding. The voiding patterns noted in the test were noticeably more extensive than those observed during the earlier Mockup 1A test (operated with one inoperative heater). The pressure pulses did not show the sharp peaks noted in the earlier test. This implies that the voiding in 1B was more strongly vapor dominated and complete collapse of the vapor did not occur.
- The fabrication and sodium filling of capsule B9D was completed ahead of schedule. Irradiation of the capsule is scheduled for early November 1971.
- Fabrication of the second in the series of fuel subassemblies using grid spacers for irradiation in EBR-II is underway. Approval-in-principle for the F9F experiment has been received. The EBR-II Project is reviewing the experiment for feasibility. The clad tubing has been received from HEDL and the hexagonal channel has been ordered.
- Inconel-625 discs irradiated to $5.0 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$ at approximately 630°C (1166°F) have been examined by transmission electron microscopy. No voids were found. Instead, the microstructure showed a very high density of a finely divided precipitate which is probably Ni_3Nb . Incoloy-800, Hastelloy-X, and solution annealed Type 304 irradiated under identical conditions all show extensive void formation.
- An unencapsulated fuel pin irradiated in EBR-II to $\sim 7.3 \text{ at.}\%$ burnup appears to be in excellent condition. The F9C-31 pin, SS316 annealed, operated at $\sim 16 \text{ kW/ft}$ with $\sim 1025^\circ\text{F}$ maximum cladding temperature. Visual examination, gamma scanning, neutrography, scanning with the Pulsed Eddy Current Cladding Analyzer and profilometry are complete.
- The second high-burnup EBR-II specimen was exposed to a TREAT transient and apparently suffered a small failure. The cladding breach was sufficient to permit sodium ingress, yet the fuel pin failed to release any detectable fission gas. Comparative fission gas punctures of the C4G capsule and C4G fuel pin plenum showed complete retention of fission gas within the C4G fuel pin (plenum pressure of $\sim 22 \text{ atmosphere STP}$), yet presence of sodium in the fuel column (confirmed during metallographic examination) demonstrates that fuel pin cladding integrity has been lost.
- The design basis for the first GE EBR-II grid spaced fuel assembly irradiation test, F9E, is reported here. The conceptual design of the second FFTF EBR-II grid assembly experiment F9F was completed. The two 37-pin grid-spaced bundle test sections have been fabricated and a 4000 hour test in flowing sodium at 1000°F has been started. Fabrication of a hydraulic test section has been initiated for a 217-pin FFTF grid-spaced driver fuel assembly.
- Immersion density measurements of irradiation induced swelling in cladding sections from F3B and F5 series fuel pins were completed. A sample cladding with 50% cold-worked Type 316 from pin F5Q showed zero swelling after irradiation at $526 \text{ to } 625^\circ\text{C}$ ($979 \text{ to } 1157^\circ\text{F}$) to fluences of $2.4 \text{ to } 3.1 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. Solution annealed Types 316 and 321 irradiated under similar conditions showed moderate peak swellings, up to $1\% \Delta V/V$.
- A review of Pu-239 nuclear data for ENDF/B-III disclosed no major weaknesses. Initial test calculations with the three-dimensional flux linking code, TRIAL, show promise for the linking procedure. The ETOX-1DX codes have been obtained from WADCO and are being converted to the GE-635 computer. The users manual for the TDOWN code was distributed and codes ENDRUN-1, PUNT, and TDOWN were sent to the ANL Code Center. Preliminary planning of the experimental program in ZPPR-3 defined a three-phase program corresponding to the end-of-cycle, mid-cycle, and beginning-of-cycle conditions in a demonstration reactor. Analysis of ZPR6-7 experiments shows good agreement between computed and measured reactor rate distributions about a simulated control rod with and without BeO .

- The Task C-1 Transient Overpower experiments are being analyzed using the TOFF model. Good agreement between experiment and analysis has been obtained for the C4B test.
- Improvements have been made to the treatment of Doppler feedback during sodium voiding in the FREADM program. The improved model predicts a significantly lower power level at the time of disassembly for both loss-of-flow and reactivity insertion accidents.
- Anomalies at the cladding inside surface of irradiated fuel rods have been successfully identified prior to destructive examination. The metallographic examination of the F5 experiment, irradiated to ~55,000 MWd/Te peak burnup in EBR-II with maximum cladding temperature > 1200°F, revealed excellent correspondence between directly observed anomalies and predictions based upon the scans from the Pulsed Eddy Current cladding analyzer.
- A study was initiated to evaluate the capabilities of the irradiation facilities which may be used for testing failed mixed oxide pins and to determine the consequences of post failure operation under typical LMFBR steady-state operating conditions.
- Burst testing of specimens from six F3A fuel pins at 594°C (1100°F) is complete. Specimens taken from the upper three-inch region of the core exhibited the lowest values for ultimate burst strength.
- The first two series of scoping tests to develop a self-actuating fuel pin defecting mechanism were completed and their examination initiated.
- Excess oxygen in mixed oxide fuel has been shown to have a direct effect on plutonium migration. Mixed oxide fuel pins irradiated under near identical conditions in a fast flux to a burnup of 1800 MWd/Te indicate that hyperstoichiometric pins exhibited plutonium increases from a nominal of 20% to 25% in the center regions. Pins with oxygen to metal ratios of 2.00 and 1.98 did not indicate any increase in plutonium content.

1. SUMMARY

1.1 TASK B—DEFECTED FUEL BEHAVIOR

Fabrication and sodium filling of capsule B9D was completed ahead of schedule. Final preparations for insertion in the reactor early in November 1971 were initiated. This capsule will investigate the performance of mixed oxide fueled pins defected in high temperature (1050°F) flowing sodium after several days of full power operation.

Work was initiated on a study to evaluate available irradiation facilities for testing failed pins and to determine the consequences of post-failure operation of failed mixed oxide fueled pins under typical LMFBR steady-state operating conditions.

The capsule B9B Design Basis Specification was completed and submitted for review and comment. Capsule B9B is to investigate defected fuel behavior at high burnup.

Development work was continued on a heat pipe designed to remove excess heat from a forced convection capsule in case of pump shutdown after removal of the capsule from the reactor.

Two series of scoping tests to develop a self-actuating fuel pin defecting mechanism were completed. Dismantling and metallographic examination of the test specimens were initiated.

Design of a series of tests to determine fuel-sodium reaction rates and the volume changes associated with the reaction was completed. Preparation of a design basis specification for the test was initiated.

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

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

Comparative observations of substructure in the columnar grain growth region indicate that maximums of ~10 vol% of the fuel for C4H and ~30 vol% for C4G reached solidus temperature (defined as melt volume). These figures are below the pre-examination predictions of ~40 vol% and ~50 vol%, respectively. The deviation corresponds to ~5% of the total pin fission energy. Gamma heating, variations in radial power profile (due to high burnup), and irradiated fuel properties are being evaluated to identify the source of discrepancy.

Sodium was found in the interior of the C4G fuel specimen during metallography. The location and form of cladding breach has not been found, but it is significant that there was apparently no fission gas lost from the pin (plenum pressure of ~22 atmosphere at STP).

A topical report (GEAP-13722) covering the C6A and C6B tests has been released for general distribution.

The Task C-1 work scope now includes initiation of transient overpower testing in the SEFOR reactor by the end of GFY-72. The initial tests will involve quarter-inch diameter, mixed-oxide fuel pins in an annular-geometry, stagnant-sodium capsule. Concurrently, the Series 8 Mark II loop tests have been deleted from the current work scope.

Task C-1 presented the functional experimental and analytical relationship between Task C-1 and Task G-10 to a HEDL/AEC review and a AEC-DRL review of the PA-10 program.

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

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

The Mockup 1B out-of-pile loss-of-flow test was successfully completed during this quarter. Voiding appeared to be much more extensive than the earlier Mockup 1A test. Analysis of the voiding behavior is underway using the BUBBLE/VOIDUM model. Work is underway on the Design Basis Specification for the C11A capsule. Work continued on the upgrading of the THTE program. A presentation of Task C-2 tests and results was given to a visiting team from AEC-DRL and ANL.

Table 1-1
TASK B-IRRADIATIONS

		Design Parameters						
		Fuel						
Capsule	Objective	Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in. diam)	Composition			Cladding Material
					Density (Smear) (% TD)	PuO ₂ (%)	O/M	
B8B (2 pins)	Fuel-sodium compatibility. Sodium-bonded fuel central temperature versus exposure time.	0.216	2	0.010	80	25	2.00	347
		0.216	2	0.010	83	25	1.93	347
B8C (2 pins)	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel- sodium-clad compatibility.	0.144	2.2	0.006	86	25	1.97	316
		0.204	2.25	0.017*	84	25	1.97	316
B8D	Fuel-sodium compatibility versus density, stoichiometry and Pu content	0.214	1.75	0.0065	80-88	40	1.97	316
		0.214	1.75	0.0065	80-88	25	1.97	
		0.214	1.75	0.0065	80-88	25	1.94	
B9A (2 pins axially aligned)	Defected fuel pin. Fuel swelling versus stoichiometry in flowing sodium	0.216	5	0.005	89	25	1.99	316
		0.216	5	0.005	89	25	1.95	
E1J-B1	Thermal expansion clad defect mechanism	0.221	5	0.0007	94	25	2.00	316
E1J-B2	Thermal expansion clad defect mechanism	0.221	5	0.0007	92	25	2.00	316
B9C (2 pins)	Swelling behavior of defected fuel pins in flowing sodium	0.216	4.25	0.004	89.6	25	1.97	316
		0.216	4.25	0.004	89.6	25	1.97	20% CW 316 Annealed
B9D (2 pins)	Two defected fuel pins. Fuel swelling in clean sodium. (< 5 ppm Oxygen).	0.215	4.25	0.005	89.5	25	1.97	316 20% CW

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Capsule	Objective	Irradiation Facility	Operating Conditions			Status
			Cladding Surface Temp (°F)	Peak Rating (kW/ft)	Peak Burnup (MWd/Te)	
B8B (2 pins)	Fuel-sodium compatibility. Sodium-bonded fuel central temperature versus exposure time.	V-Raft GETR	800	14	14,500	Geap-13729 completed
B8C (2 pins)	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel-sodium-clad compatibility.	V-Raft GETR	1100	21	113,500 57,000	Examination completed
B8D	Fuel-sodium compatibility versus density, stoichiometry and Pu content	V-Raft GETR	950	17.5	41,000 43,500 41,400	Examination completed
B9A (2 pins axially aligned)	Defected fuel pin. Fuel swelling versus stoichiometry in flowing sodium	Special V-Raft GETR	1019 971	23.4 23.4	8,600 8,600	GEAP-13730 released
E1J-B1	Thermal expansion clad defect mechanism	Trail Cable	835	17.5	50	Examination completed
E1J-B2	Thermal expansion clad defect mechanism	Trail Cable	860	19.0	300	Examination completed
B9C (2 pins)	Swelling behavior of defected fuel pins in flowing sodium	Special V-Raft GETR	1015 1030	20.0 20.0	~7300** ~7300**	Test terminated Test terminated
B9D (2 pins)	Two defected fuel pins. Fuel swelling in clean sodium (< 5 ppm Oxygen).	Special V-Raft GETR	1100 1060	15.0	~5500**	Fabrication in process

* Large gap required to accommodate sampling device

** Planned irradiation.

Table 1-2
TASK C-1-EXPERIMENTAL TESTS

		Design Parameters							
Capsule	Objective	Diameter (inch)	Fuel Length (inch)	Fuel Density (% TD)		Fuel Composition			Cladding Material
				Pellet	Smear	%PuO ₂	%U-235	O/M	
C4A	Transient failure threshold for zero-burnup, mixed-oxide fuel as a function of axial restraint density, and form (pellet versus powder).	0.2159	14.23	93.8	90.4	19.4	0.57	1.983	316SS
C4B		0.2155	14.21	94.5	91.2	19.4	0.57	1.983	316SS
C4C		0.2164	14.19	83.1	80.4	20.9	0.91	1.983	316SS
C4D		0.2154	14.22	83.7	80.1	20.9	0.91	1.983	316SS
C4I		powder	13.48	powder	83.5	20.2	0.89	1.973	316SS
C4J		powder	13.45	powder	83.5	20.2	0.89	1.973	316SS
C4E	Comparison with C4A and C4B test results (high-density pellet fuel)	0.2159	13.46	83.2	80.5	28	92.7	1.974	316SS
C4F		0.2160	13.51	84	80.5	28	92.8	1.975	316SS
C4G	Comparison with C4A and B to determine effect of burnup on failure threshold (transient magnitude will be varied).	0.2154	13.50	94.4	90.5	28	89.3	1.973	316SS
C4H		0.2152	13.45	94.4	90.4	28	89.3	1.976	316SS
C4K	Comparison with C4I and J to determine effect of burnup on failure threshold of powder fuel (transient magnitude will be varied).	0.2205	13.50	Powder	84.1	28	93.0	1.972	316SS
C4L		0.2206	13.47	Powder	84.2	28	93.0	1.972	316SS
C4M	Failure threshold for low-power (~10 kW/ft), high burnup fuel.								
C4N	Comparison to C4M								
C4X	Proof test of redesigned C4 TREAT capsule (remote assembly techniques and TREAT characteristics) to be used with C4E-H and K-M.	0.2186	13.48	96.2	91.0	0	92.8	2.000	316SS
C5A	Mobility of transient melted fuel; investigate potential for fuel relocation under prototypical conditions of length, restraint, and fission gas.	0.214	24	95	91.5	0	12	2.000	316SS
C5B		0.214	24	95	91	0	12	2.000	316SS
C6A-1	Transient performance of defected or sodium bonded fuel; relative performance limits and failure mechanisms of pellet and powder fuel under transient irradiation conditions. Sodium logged powder fuel is to maximize fuel-sodium heat transfer (transient magnitude will be varied).	0.212	24	86	80	0	10.0	2.000	316SS
C6A-2		0.2203	24	powder	80	0	10.0	2.000	316SS
C6A-3		0.212	24	86	80	0	10.0	2.000	316SS
C6B-1		0.212	24	86	80	0	10.0	2.000	316SS
C6B-2		0.2203	24	powder	80	0	10.0	2.000	316SS
C6B-3		0.212	24	86	80	0	10.0	2.000	316SS

NP Not Planned

P Planned

I Initiated

NC Nearly Completed

C Completed

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

† Consists of x-rays, neutron radiographs, de-encapsulation, cladding, diametral measurements, gamma transmission scan

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

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

1.4 TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

Capsule E2R, designed to investigate center hole formation, has been irradiated for a total of six hours. While the results from the most recent two-hour exposure are not yet available, no central holes were observed in 94% T.D. pellets after four hours of irradiation. Central holes were previously noted in 88% T.D. pellets after 26 minutes in pile at a center temperature of 2500°C.

Electron microprobe capability has been extended to carbon and nitrogen by the use of a fusible alloy mounting compound for the sample. Concentrations of these elements have been detected in the stainless steel cladding.

Out-of-pile fueled thermal gradient capsules have indicated that cesium forms compounds with iodine and tellurium. Their identification together in the microprobe analysis of irradiated samples results from these compound formations.

Microprobe examination of fuel pins irradiated to 1800 MWd/Te indicates that excess oxygen in mixed oxide fuel increase the amount of plutonium migration that occurs during irradiation. Plutonium increases up to 5% in the center region of fuel pins with an O/M ratio of 2.03 were observed with no corresponding increase in specimens with 2.00 and 1.98 O/M ratios irradiated under identical conditions.

Small diameter $\text{ZrO}_2\text{-CaO}$ and $\text{ThO}_2\text{-Y}_2\text{O}_3$ tubular electrolytes have been obtained for the construction of an emf probe that will enable the radial variation of oxygen potential of irradiated mixed oxide fuel to be determined. This information will contribute to the study of fuel - fission, product - cladding and fuel-sodium reactions.

Melting point determinations have been made on control and irradiated fuel by the thermal arrest method in a joint GE-LASL endeavor. Unirradiated values are in accord with literature findings and a 30 to 35°C depression occurs at a burnup of 4.3 at. %.

1.5 TASK F—FAST FLUX TESTS AND SUPPORT

Generalized cladding attack to a maximum depth of ~ 3 mils was observed in a longitudinal metallographic section from the peak temperature zone of pin E1H-30. The cladding was 13% cold worked Type 304 stainless steel with an additional Garafalo-type heat treatment. The attack was extensive when compared to that observed in a similar section from pin E1H-29, which had the same cladding except for the special heat treatment. This confirms the earlier comparison based upon transverse sections. Intergranular attack was negligible.

The post-irradiation examination of fuel pin F9C-31 is underway. The unencapsulated fuel pin, irradiated at ~ 13 kW/ft peak with ~ 1035°F maximum cladding temperature to ~ 7.3 at. % burnup, appears to be in excellent condition. Visual examination, gamma scanning, neutrography, scanning with the Pulsed Eddy Current Cladding Analyzer, and profilometry are complete.

The interim metallographic examination of three pins of the F5 experiment, now complete, showed that anomalies at the cladding inside surface could be identified successfully from pulsed eddy current (PEC) scans prior to the destructive examination. The fission gas release from these three pins was nearly 100% based upon measured burnup of ~ 5 at. %. The measured burnup was substantially less than that calculated using EBR-II methods.

Preliminary results of the evaluation of the cladding attack by fission products in 60 fuel pins indicate: (1) no discernible effect of O/M in the range 1.97 to 2.00 or pellet density from 86 to 97% theoretical; (2) high pin power and cladding temperature are necessary but not sufficient to cause attack; and (3) no difference in behavior between annealed Types-316 and -347 stainless steel at cladding temperatures up to 1150°F.

The irradiation of experiment F12P and Q is complete. The F12P and F12Q pins, along with pins F4A, F4E, and F9C-13, are being prepared for post-irradiation examination at LASL. The post irradiation examination of the remaining F4 pins is underway. Topical reports on the results of the post-irradiation examination of the F2 and F3A experiments are in progress. The data package for the reconstitution of the F9C experiment into the second 19-pin Mark E subassembly (F9C-2) is being prepared. Difficulty developed during assembly of the F10A and F10B experiments at EBR-II; the channel supplied by ANL and the triangular flow regulators are being reworked. Fabrication

of the cladding and the fuel pellets for the F20 experiment is nearly complete. The F11A fuel pins and the F9E fuel pins and grid spaced channel have been shipped to EBR-II.

Fabrication of the F9F experiment is underway. The Approval-in-Principle has been received. The EBR-II Project is reviewing the experiment for feasibility. The 0.230 in. diameter, 20% cold worked SS316 clad tubing has been received from HEDL. The 20% cold worked SS316 hex channel has been ordered.

The status of the fast flux irradiations currently in progress under Task F is summarized in Table 1-3.

1.6 TASK G-6—CORE DESIGN ANALYSIS

The group F9E irradiation is the first in-pile test of a grid spaced fuel assembly utilizing the straight wave beam grid spacer concept conceived by General Electric. F9E employs three different variations of the basic wave beam concept to obtain test information on the effect of these design alternatives. The F9E experiment has been shipped to EBR-II where the fuel pins will be loaded for testing.

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

A six month sodium flow test is underway in the General Electric Breeder Test Facility Control Drive Test Loop to evaluate the endurance characteristics of the reference FFTF Grid Spacer Design and to provide engineering information that will support the design bases for the F9E, F9F and reference FFTF grid-spaced fuel assembly. The test assembly consists of two separate 37-pin dummy fuel bundles positioned in series. The test is scheduled for approximately 4000 hours of operation in sodium at 1000°F with a peak velocity of 22.3 ft/sec. The test assembly was installed in the drive test loop on September 23, 1971 and operation began on October 5, 1971.

Flow induced vibration is of concern with any fuel bundle designed for extended periods of operation. In order to determine the flow induced vibration response in the reference FFTF fuel bundles, a two-point test program is in progress:

1. Single pin hydraulic flow tests with simulated grids.
2. 217 pin full scale hydraulic flow tests containing 12 staggered grids and 6 full grids.

1.7 TASK G-10—SAFETY METHODS AND ANALYSIS

The Task C-1 Transient Overpower experiments are being analyzed using the TOFF model. Good agreement between experiment and analysis has been obtained for the C4B test.

The VOIDUM-BUBBLE programs have been integrated. These routines now form a free-standing package to allow the computation of transient primary system hydraulics including slug boiling and condensation.

A preliminary study of the importance of core surveillance instrumentation in limiting the progress of a postulated fuel failure propagation accident was completed. The results indicate that the mean-annual-cost of an excursion is very low and not significantly different for full plant protection system response or the neutron flux only response.

Improvements have been made to the treatment of Doppler feedback during sodium voiding in the FREADM program. The improved model predicts a significantly lower power level at the time of disassembly for both loss of flow and reactivity insertion accidents.

Transient radial temperature distributions near the channel wall of a fuel bundle have been computed to the time of boiling initiation for two flow coastdown rates. The results of this analysis will be used in the formation of a radial condensation model and to provide a basis for relating few pin tests to expected reactor conditions.

Table 1-3
TASK F FAST FLUX TESTING PROGRAM IN EBR-II (a)

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

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Range of Average Pin Powers ^(b) kW/ft	Target at. %	Burnup MWd/Te	Average Calculated Burnup ^(c)		Peak Calculated Fluence ^(d) (nvt X 10 ⁻²³)		Status
			at. %	MWd/Te	Total	E > 0.1 MeV	
7.76	3.91	35000	3.34	30199	0.276	0.199	Post exam complete
5.47– 5.92	8.95	80000	6.88	62247	0.593	0.432	Reconstituted into XG05A
5.44– 6.24	11.18	100000	7.62	68916	0.662	0.490	Reconstituted into XG04A
5.58– 6.23	8.95	80000	7.59	68681	0.659	0.479	In reactor
5.42– 6.19	11.18	100000	8.32	75249	0.729	0.537	In reactor
12.32–14.49	0.22	2000	8.21	1882	0.017	0.015	Reported in GEAP-5578
11.42–13.91	8.39	75000	6.69	60543	0.369	0.488	Post exam complete
8.48–14.83	5.59	50000	5.02	45451	0.422	0.362	Reported in GEAP-13549
10.73–12.86	11.19	100000	11.75	106358	1.034	0.887	Post exam in progress
12.87–14.65	5.62	50000	2.88	25959	0.236	0.200	Reconstituted into X040A
12.53–14.25	5.62	50000	5.17	46620	0.430	0.365	Post exam initiated
11.48–12.80	13.49	120000	8.86	79922	0.754	0.647	Interim exam
13.74–14.53	5.65	50000	3.64	32758	0.298	0.257	Reconstituted into X039
13.41–13.83	9.04	80000	4.85	43633	0.400	0.344	Reconstituted into X074
12.72–13.37	9.04	80000	8.25	74217	0.695	0.597	Interim exam
12.56–13.36	11.23	100000	5.71	51568	0.476	0.408	Removed from reactor
8.59– 9.34	5.59	50000	5.71	51645	0.480	0.391	Post exam complete
12.54–12.65	5.64	50000	5.40	48599	0.456	0.390	Terminated
11.91–12.25	5.63	50000	5.43	48914	0.449	0.384	Terminated
8.53– 9.15	5.62	50000	4.97	44777	0.412	0.347	Post exam complete
8.37– 9.11	11.25	100000	6.15	55398	0.524	0.438	Reconstituted into X061
6.78– 8.46	11.24	100000	8.07	72759	0.693	0.567	In reactor
4.99– 6.39	11.24	100000	4.73	42600	0.401	0.303	Removed from reactor
4.37– 5.58	5.62	50000	4.75	42781	0.406	0.299	In reactor
5.04– 6.28	5.62	50000	5.25	47307	0.447	0.336	In reactor

Table 1-3 (Continued)

Group	Objective	Total Number of Pins in Group	Date In	Projected Date Out	Actual Date Out	Number of Pins in S/A	S/A	Position
F9A	High power, high burnup, unencapsulated fuel pin irradiations	37	2/20/69		5/18/70	37	X043	4D2
			10/27/70	6/ 8/72		37	X043A	6D3
F9B	Medium power, high burnup, unencapsulated fuel pin irradiations	37	5/23/69	11/ 6/72		37	X062	7F3
F9C	High power, high burnup, unencapsulated fuel pin irradiations	37	4/ 3/69		7/16/70	37	X056	5C2
			10/27/70	6/22/72		37	X056A	6B2
F9D	Medium power, high burnup, unencapsulated fuel pin irradiations	37	4/24/69	7/22/73		37	X058	7B6
F9E	High power, high burnup, unencapsulated fuel pin irradiations in a grid spacer subassembly	37						
F9F	High power, high burnup, unencapsulated fuel pin irradiations in a grid spacer subassembly	37						
F10A	Cladding performance at peak cladding temperatures	19						
F10B	Cladding performance at peak cladding temperatures	19						
F10C	Fuel pin performance at peak cladding temperatures	19						
F11A	Cladding performance at peak for cladding temperatures	37						
F20	Irradiation to probe operating limits	19						

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

Range of Average Pin Powers ^(b) kW/ft	Target Burnup at. %	MWd/Te	Average Calculated Burnup ^(c)		Peak Calculated Fluence ^(d) (nvt X 10 ⁻²³)		Status
			at. %	MWd/Te	Total	E > 0.1 MeV	
12.06–13.36	11.25	100000	5.57	50189	0.464	0.398	Reconstituted into X043A
10.64–12.72	11.25	100000	6.62	59670	0.557	0.477	Out of reactor
7.99–10.48	11.24	100000	6.40	57673	0.538	0.451	In reactor
10.50–11.55	11.24	100000	5.67	51157	0.472	0.401	Reconstituted into X056A
10.20–11.71	11.24	100000	6.68	60207	0.561	0.476	Out of reactor; to be divided into 2-Mark E subassemblies
7.83– 9.63	11.24	100000	5.36	48375	0.451	0.378	In reactor

Shipped to EBR-II

Fabrication in progress

Shipped to EBR-II

Shipped to EBR-II

Design in progress

Shipped to EBR-II

Fabrication in progress

Detailed studies were initiated to provide a more explicit coordination of analytical safety efforts and experimental plans.

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

A GE engineer, P. M. Tschamper, has been stationed at the ANL on a full-time basis in a liaison capacity. His principal function is to facilitate the interchange of technical information between ANL and GE-BRD, which is recognized to be beneficial to both organizations.

1.9 TASK L—CLADDING DEVELOPMENT

Post-irradiation nondestructive examination of the three L-6 capsules has been completed and disassembly of these capsules is in progress.

Fabrication of the L-16D capsule inserts is in progress. These capsules will contain both highly irradiated and sodium exposed cladding alloy samples for irradiation in EBR-II.

Inconel-625 discs irradiated in the MT-3 capsule to 5.0×10^{22} n/cm², $E > 0.1$ MeV-GE at approximately 630°C (1166°F) were examined by transmission electron microscopy (TEM). No voids were found. Instead, the microstructure showed a very high density of finely divided precipitate which is probably Ni₃Nb. Incoloy-800, Hastelloy-X, and solution annealed Type 304 irradiated under identical conditions showed extensive swelling.

Immersion density measurements were completed on 18 cladding sections from Task F fuel pin experiments: F3A, F3B, and F5. The results show that essentially no swelling occurred in highly cold worked Type 316 cladding irradiated to 3.4×10^{22} n/cm², $E > 0.1$ MeV-GE at calculated temperatures up to 625°C (1157°F). Swelling measurements on solution annealed Type 316 cladding generally were similar to earlier F2 fuel pin cladding results with the exception of one specimen which was highly strained at 573°C (1063°F). This specimen exhibited 4.69% $\Delta V/V$, which more than double the volume change observed on the F2 pins at this fluence level. Type 321 cladding exhibited slightly more swelling (0.32 to 0.98% $\Delta V/V$) as compared to Type 316 (0.17 to 0.47) under similar irradiation conditions.

Tensile tests were completed on highly irradiated specimens of Incoloy-800, Hastelloy-X, and Inconel-625 at temperatures of 483 to 704°C (900 to 1300°F). Major reductions in ductility were observed at all test temperatures for all alloys as a result of irradiation to peak fluences of 5.9×10^{22} n/cm², $E > 0.1$ MeV-GE. Tensile strength properties show that each alloy was behaving differently during irradiation at elevated temperature.

Burst testing of sections from six F3A fuel pins at 594°C (1100°F) is complete. Specimens taken from the upper three-inch region of the core exhibited the lowest values for ultimate burst strength. It has not yet been determined whether the lower strength values are a direct result of annealing of the radiation damage or an indirect result of intergranular attack caused by the higher irradiation temperature in the upper core region. Burst specimens from Type 304 stainless steel clad fuel pins had significantly lower burst strength values than corresponding Type 316 clad pins.

Cladding samples from an F3A unencapsulated fuel pin clad with Type 316 stainless steel have been analyzed for bulk carbon and oxygen content after cleaning by removal of approximately 0.002 inches of metal from the inside surface. The carbon contents were the same as unirradiated samples of archive tubing. The oxygen content, however, increased during irradiation from 130 ppm to 230 ppm.

Equal quantities (1000 feet) of annealed, 10% and 20% cold worked Type 321H air melted tubing have been received. An order is being placed for the fabrication of Type 321 vacuum melted tubing (annealed, 10% and 20% cold worked).

The current status of the Task L irradiation program is summarized in Table 1-4.

1.10 TASK N—CORE PHYSICS ANALYSIS

The Pu-239 nuclear data for ENDF/B-II were reviewed. While some weaknesses were identified, it is anticipated that it will produce good results for LMFBR computations.

Initial test problems with the three-dimensional flux linking code, TRIAL, show promise for the linking procedure. More stringent tests will be required before a definitive evaluation can be made. These have been started. The ETOX-1DX codes have been obtained from WADCO and are being converted to the GE-635 computer. Arrangements are also being made to run problems associated with ZPPR-3 planning on the IBM 360-75 computer in Idaho.

The user's manual for the TDOWN code was distributed and the rapid cross section adjusting codes ENDRUN-II, PUNT and TDOWN were sent to the ANL Code Center.

Planning of the experimental program for critical experiments in ZPPR-3 was begun. Three basic configurations will be used, corresponding to beginning-of-cycle, mid-cycle and end-of-cycle conditions in a demonstration plant. For the end-of-cycle configuration, control channels will be simulated and the core radius increased over that of ZPPR-2. For the other two configurations, reactivity loss due to B_4C poison insertion will be accommodated by spiked Pu enrichments.

Analyses of ZPR6-7 experiments show good agreement between calculation and experiment. Analyzed experiments include reaction rates about a simulated control rod, both with and without BeO .

1.11 TASK P—FUEL ELEMENT MODELING

The BEHAVE-I modeling of Task F fuel pins F2G, F2H, and F2V suggests that two major mechanisms are active in producing the observed cladding diametral changes: the *in-situ* increase in clad swelling and the containment of released fission gases within the fuel center void. The former phenomenon has been modeled in BEHAVE-I by providing an effective increase in the embryonic void density which is a function of cladding creep strain, in qualitative agreement with the results from the TEM analyses of the L2 specimens and the cladding of pins F2H, F2V, and F2T. The latter phenomenon is modeled by assuming that noble gases and cesium are released to a gas-tight center-void volume and that the resultant pressure acting at the fuel plasticity radius is transmitted through the cooler structural fuel to produce a cladding hoop stress. The inclusion of these phenomena affects qualitative agreement between the calculated and measured fuel pin outside diameters and between the calculated metal swelling and that measured using the immersion density technique.

An analysis of new data from sequential materials test, namely a creep test followed by a tensile test to failure, provides further substantiation of the Cumulative Strain Damage Failure Criterion. The test materials were unirradiated Ti-6Al-4V and Ti-8Al-1Mo-1V.

Table 1-4

SUMMARY OF TASK-L CAPSULES IN EBR-II

Group	Purpose & Major Parameters	Total Pins In Group	Number of Specimens		Date In	Projected ⁽¹⁾ Date Out
			Tubular	Flat Tensile		
L-2'	P.I. Mech. Prop. Eval. of Type-304, -316L, -321, -347, & Incoloy-800.	5	15	80	09/03/65	02/20/67 ⁽²⁾
L-4	P.I. Mech. Prop. Eval. of Type-316.	2	10	20	03/29/66	11/14/66 ⁽²⁾
L-4'	P.I. Mech. Prop. Eval. of Type-304, -321, -347, & Incoloy-800	5	25	50	07/17/66	04/10/67 ⁽²⁾
L-10	P.I. Mech. Prop. Eval. of Type-304, -316, -321, -347, & Incoloy-800	10	70	60	01/13/67	09/ /68 ⁽²⁾
L-14	P.I. Mech. Prop. Eval. of Type-304, -316, -321, -347, & Incoloy-800	19	133	114	10/10/67	12/ /68 ⁽²⁾
L-21	P.I. Mech. Prop. Eval. of Type-321	1	3	16	09/03/65 08/15/68	03/06/68 ⁽²⁾ 12/24/68 ⁽²⁾
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 ⁽²⁾ 05/18/70 ⁽²⁾
MT-3 ⁽⁴⁾	P.I. Mech. Prop. Eval. of Incoloy-800, Inconel-625 & Hastelloy-X	4	None	48	12/13/65	06/26/69 ⁽²⁾
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 ⁽²⁾ 06/26/69 ⁽²⁾ 09/10/70 ⁽²⁾
L-15	P.I. Mech. Prop. Eval. of Type-304, -316, -321, -347 &	4	None	288	08/15/68 08/24/70	12/24/68 ⁽²⁾ 06/ /71
MT-4 ⁽⁴⁾	P.I. Mech. Prop. Eval. of Incoloy-800, Inconel-625, & Hastelloy-X	1	None	48	12/13/65 08/24/70	06/26/69 ⁽²⁾ 1972
L-16A	Meas. γ -heating rates for L-16 Type Capsule design	19			—	—
L-16B	Meas. Swelling in Type-316 & -304 Annealed and C.W.	19	76	None		
L-16C	Meas. Temp. Dependence of Swelling in Clad Alloys (Replacement capsule for L-16B)	7		None	—	—
L-16D	Reencapsulation of previously irradiated specimens	3	36	None	—	—

1. Assuming ERR-II continues to operate at 62.5MW with 50% Load Factor.
2. Actual date out.
3. Fluence calculated as described in GEAP-10028-37, pages 68-69.
4. Capsules from previous AEC program under Contract AT(04-3)-189, PA-29.
5. Fluence prior to reencapsulation.

Table 1-4 (Continued)

Row-S/A	Core Position	Reactor Exposure (MWD)	Target Fluence	S/A Average		Status
			(n/cm ² x 10 ²³) Total	(n/cm ² x 10 ²³) Total	(n/cm ² x 10 ²³) 0.1MeV	
4-XG06	4E2	9,317	0.4	0.33	0.29	Report complete
4-X009	4A2	5,355	0.2	0.19	0.16	Report complete
2-X014	2D1	3,674	0.2	0.16	0.14	Report complete
4-X016	4D3 & 4F3	7,894	0.2	0.28	0.24	Report complete
4-X025	4E2	8,399	0.2	0.29	0.25	Post exam complete
4-X605	4C2	12,632	0.75	0.54	0.46	Post exam in progress
3-X039	3C2	2,532				
4-X605	4C2	12,632	0.75	0.90	0.75	Post exam in progress
4-X050	4C2	11,706				
4-XA08	4F2	19,280	1.0	0.64	0.55	Exam completed
2-X018	2B1	10,687	2.0	0.78	0.67	At FEF, EBR-II
4-X018A	4E2	2,334				
4-X018B	4E2	9,851				
3-X039	3C2	2,532	0.3	0.15	0.12	In reactor
4-X080	6E5	2,525				
4-XA08	4F2	19,280	2.0	0.70	0.60	In reactor
6-X080	6E5	2,525				
4	4C2	—	0.01			Fabrication in progress
4-X098	4C2	—	0.3	0.19	0.16	In reactor
—	—	—				Design in progress
—	—	—		0.90 ⁽⁵⁾	0.75 ⁽⁵⁾	Design in progress

2. TASK B—DEFECTED FUEL BEHAVIOR

Task Leader: C. N. Craig*

2.1 GENERAL

The technical and economic feasibility of fast reactors depends on a thorough knowledge of defected fuel behavior and its consequences to reactor design and safety. Defective fuel elements can be expected in any reactor, no matter how elaborate the preventive measures. Moreover, to reach the full economic potential of the LMFBR, high plant availability must be achieved. It must therefore be experimentally demonstrated that an LMFBR can continue to operate safely with a substantial number of fuel pins having breaches in the cladding.

The specific purpose of Task B is to (1) determine analytically and experimentally the consequences on the operation of the reactor of failure of a fuel pin under normal or near normal LMFBR operating conditions, (2) provide experimental and analytical data to permit selection of fuel pin design parameters and operating conditions which limit the consequences of fuel failure, (3) provide experimental data on failure kinetics to permit modeling and prediction of fuel failure behavior for the projected range of LMFBR operating conditions, and (4) identify the fuel and fission-product species distribution in the reactor coolant and cover gas as a function of the failure mode.

2.2 DEFECTED FUEL TESTING

2.2.1 Forced Convection Capsule B9B

Capsule B9B will investigate the behavior of failed fuel at high burnups. The capsule Design Basis Specification was completed and submitted for review and comment to quality assurance and the organizations responsible for the design, fabrication, testing and examination of the capsule. The capsule will contain a single mixed oxide fueled pin which will be defected at power after an exposure of approximately 25,000 MWd/Te. A pressure transducer will be incorporated in the fuel pin plenum to measure the rate of release of fission gas from the pin after failure occurs.

2.2.2 Forced Convection Capsule B9D

Capsule B9D is designed to investigate the performance of fuel pins defected in high temperature (1050°F) flowing sodium after several days of full power operation. Fabrication and filling of capsule B9D with sodium was completed. Final inspection and acceptance signoffs are presently underway. The completed capsule is shown prior to filling with sodium in Figure 2-1.

To ensure safe handling of the B9D capsule, a dry run was carried out in which the steps to be used to transport the capsule to the reactor and insert and remove the capsule from the reactor were followed. A mockup of the capsule was used for the dry run. Handling of the capsule was carried out in accordance with approved written procedures and was observed by an audit team. Recommendations made by the audit team will be incorporated into the handling procedures prior to transport of the capsule to the reactor. Irradiation of the capsule in GETR is planned for early November 1971.

2.3 TECHNICAL SUPPORT FOR IN-PILE DEFECTED FUEL TESTING

2.3.1 Heat Pipe Development

Developmental testing was continued on a radial heat pipe for use as a "thermal switch" to prevent overheating of the fuel pin in a forced convection capsule in the event of a pump shutdown after removal of a capsule from the reactor. Checkout of the heat pipe test apparatus shown in Figure 2-2 was completed. Two experimental runs were completed which successfully demonstrated the operation of the heat pipe. Based on the successful operation of the prototype a full size heat pipe assembly was designed and fabrication initiated. Two design changes were incorporated into the new design as a result of the experience gained in fabricating the prototype test assembly. These design changes will improve the electrical insulation for the heater lead wires and will incorporate a bellows at the top of the heat pipe to accommodate differential expansion of the inner and outer walls of the heat pipe.

* Contributors are listed in Appendix B.

Figure 2-1. Completed B9D Capsule Prior to Filling with Sodium

Figure 2-2. Heat Pipe Test Assembly

2.3.2 Defecting Mechanism Development

Scoping tests continued for developing a self-actuating fuel pin defecting mechanism for tests to be conducted in-pile. Figure 2-3a shows the defect test capsules being inserted into their test containers. The assembled test container is shown in Figure 2-3b. The first two series of test capsules contained UO_2 and potentially corrosive chemicals (CsOH , CsMoO_4 , and AgF). The two capsule series were exposed to temperatures of 800°F and 900°F respectively for 500 hours. Dismantling of the capsules was completed and preparation of metallographic sections of the capsules initiated. Figure 2-4 shows one of the test capsules being sectioned after its removal from the test container.

2.4 FUEL-SODIUM REACTION STUDIES

Design of the first three tests in a series of experiments to determine rate of reaction between mixed oxide fuel and sodium and the resulting volume change of the fuel and reaction product was completed. A draft of the Design Basis Specification and engineering drawings of the conceptual design were prepared.

2.5 RUN BEYOND FAILURE TESTS

Work was initiated to scope a series of tests to determine the consequences of post-failure operation of mixed oxide fuel pins under typical LMFBR steady-state operating conditions. Contacts were made with the EBR-II Project personnel to determine the feasibility of conducting such tests in the EBR-II. Other reactors will also be evaluated as part of a study to assess the relative capabilities of available facilities which would be used for testing and examining failed mixed oxide fueled pins.

(a) LOADING OF DEFECT TEST CAPSULES INTO TEST CONTAINER

(b) FULLY ASSEMBLED TEST CONTAINER

Figure 2-3. Defect Development Test Capsules and Container

Figure 2-4. Defect Test Capsule After Removing from Test Container and Sectioning

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

Task Leader: G. R. Thomas*

3.1 GENERAL

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

3.2 HIGH-BURNUP FUEL EXPERIMENTS

The transient overpower failure thresholds and subsequent fuel-sodium interactions are being investigated for pre-irradiated fuel specimens of varying fuel smeared density (80% to 90%), fuel form (pellet and powder), and prior irradiation history. Current specimens allocated for these tests have accumulated ~50,000 MWd/Te fast-flux burnup in the EBR-II reactor. Included in this inventory are six GE pins irradiated at 13 to 15 kW/ft, and two HEDL supplied FFTF pins (PNL-2-2 and 2-7) which operated at a lower power level of ~10 kW/ft.

3.2.1 Capsule Fabrication, Assembly and TREAT Irradiation

Fabrication of selected portions of the C4M capsule assembly has been completed (e.g., the flowmeter assembly) and procurement of long-lead items (e.g., the pressure transducer and high-pressure instrumentation connector) has been initiated.

The product quality plan covering capsule assembly has completed first-round vendor (VNC-IPO) iteration and should receive all necessary approvals by early next quarter.

3.2.2 Post-Irradiation Analysis

Primary destructive examinations of both TREAT specimens C4G and C4H (preirradiated in EBR-II to ~50,000 MWd/Te) have been completed. The salient features of the examinations are described below.

The microstructures of C4G and C4H indicate an overprediction of the amount of fuel reaching solidus temperature (defined as melt volume). The original pre-exam calculations predicted ~50 Vol% and ~40 Vol% of the C4G and C4H fuel respectively reaching solidus temperature. Study of comparative microstructures near the midplane regions of both pins shows a loss of substructure in the columnar grain growth region at radii equivalent to ~30 Vol% and ~10 Vol% for C4G and C4H.

Such a structure difference is inconsistent with the fuel restructuring that would occur as a result of the comparative EBR-II power levels of these two pins (the two pins operated within $\pm 2\%$ power level of each other throughout EBR-II irradiation). In addition, there were notable fission gas bubble concentrations (primarily 0.5-1 mil diameter) in the central regions of the fuel areas located axially below the central void region, and portions of the topmost fuel pellets of both C4G and C4H apparently melted (end-flux peaking).

It seems probable therefore that the loss of substructure represents fuel that entered heat of fusion (~30 Vol% for C4G, and ~10 Vol% for C4H).

A sensitivity study is underway to determine the cause of the thermal overpredictions. Areas of investigation include radial power profiles in the fuel, relative fission to gamma heating in the capsule and thermal properties of irradiated fuel.

* Contributors are listed in Appendix B.

It is estimated that the calculational deviation involved in the overpredictions is equivalent to ~5% of the total fuel pin fission energy. Due to the newly-licensed capability for TREAT power shaping, and the experience gained in analysis of the C4G and C4H tests, it will be possible to adequately plan and predict the transient test conditions and eliminate this problem for the remaining Series 4 capsules.

Examination of the C4H cladding cross section in the upper of the two failure regions indicated that the cladding failed in a brittle manner (intergranular cracking), with one complete cladding penetration and several adjacent incipient cracks. There was also a significant amount of intergranular growth of sigma phase in the cladding. Figure 3-1 presents enlarged details of a transverse section through the upper failure area shown in Figure 4-2 of the PA-10 39th Quarterly Report.

Early in the C4G examination it was noted that portions of the lower region of the fuel column were contaminated with an apparently active metal (i.e., cesium, sodium, and/or others). Preliminary wet chemistry analysis of the contamination indicates that sodium is present (concentrations ~10 to 100 times greater than for the expected fission product cesium). It can therefore be concluded that the C4G fuel pin integrity has been lost. Testing and examinations have been initiated to determine the location of failure and to attempt to prove time of failure (i.e., prior to or concurrent with TREAT testing). A very significant point to be noted is that pin integrity was lost sufficiently to allow sodium ingress to the fuel, yet the pin apparently retained its full fission gas inventory (~22 atmosphere STP in the plenum measured during fission gas puncture of the C4G fuel pin), despite facing a low capsule cover gas pressure of ~0.13 atmosphere STP.

It should also be noted that because of the apparently complete retention of C4G fission gas (e.g., none detected in capsule cover gas), loss of cladding integrity was not expected, nor had it been detected by any of the nondestructive examination techniques used prior to fuel pin sectioning. These nondestructive techniques included gas puncture of the capsule wall (no fission gas present), visual examination neutrography, profilometry, gamma scanning and pulsed eddy current signature analysis of the cladding inner and outer surfaces. The pulsed eddy current results indicated only two minor defect regions on the inner surface of the cladding (estimated <0.001 inch in depth), neither of which were found during metallography. Since this technique cannot investigate the weld regions of the pin (nor practically speaking, within ~1 inch of the weld zones because of detector head configuration), it is suspected that the failure is in the bottom weld or closely adjacent to this region. Tests are underway to determine the failure location and form precisely.

3.3 COMPARATIVE TESTS

A topical report (GEAP-13722) discussing the C6A and C6B tests (sodium bonded/logged pins) has been released for general distribution.

3.4 MARK II FLOWING SODIUM TESTS

The series 8 flowing sodium tests have been deleted from the Task C-1 work scope for CP-72 in favor of a SEFOR reactor transient overpower test (see below). The basic work completed under this series will be utilized in the planning and design of Task C-2 transient undercooling tests involving ANL Mark II loops.

3.5 SEFOR TRANSIENT OVERPOWER TESTS

Current work scope has been modified to incorporate the first capsule transient overpower tests to be performed in the SEFOR reactor. The first test series is slated to begin prior to the end of GFY-72, and will involve a relatively simple stagnant sodium capsule design. Present plans require that the fuel pin be surrounded by 0.200 inch of ZrH_2 to provide slight moderation capability for increasing specific test fuel pin power. The fuel will be zero-burnup mixed oxide (fully enriched in U-235), with a column length of ~13.5 inches.

A primary purpose of this first test series is to establish the relative equivalence and worth of previous TREAT transient test series for direct application to fast reactor overpower failure thresholds.

(a) TRANSVERSE SECTION THROUGH FAILURE REGION

(b) INCIPIENT CRACK FORMATION
BRITTLE FRACTURE

(c) RIGHT EDGE OF CLADDING IN FAILED REGION. BRITTLE, INTERGRANULAR FAILURE IS APPARENT

SIGMA PHASE FORMATIONS
FISSION PRODUCT INTERACTION LAYER
(d) INCIPIENT CRACK FORMATION
BRITTLE FRACTURE

Figure 3-1. Task C-1 Series 4 Fuel Specimen C4H—Transverse Section Through Upper Failure Region

3.6 MODEL DEVELOPMENT

Modeling efforts using the BEHAVE-2 fuel performance code (Task P) are continuing. The C4A TREAT test is being used to form a correlative basis for checking the effectiveness of the code predictions because of the well-defined cladding deformation in the C4A case (see GEAP-13562). Once the correlative basis is established, the C4B, C, D, G, and H tests will be studied.

3.7 ADMINISTRATIVE

Task C-1 staff participated in a two-day program review of the experimental-analytical relationship between Task C-1 and G-10 (Safety Methods Development) presented to AEC-RDT/HEDL. The purpose of the meeting was to aid HEDL in their Safety Assurance Diagram development program for FFTF. The same general area was also discussed in a one-day review of the program for AEC-DRL/ANL.

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

Task Leader: H. C. Pfefferlen*

4.1 GENERAL

A major safety-related area that is of concern to both 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 pins, and the time characteristics of signals (such as temperature and flow) are of primary concern in the design of the fuel bundles and the instrumentation system. Firm data in these areas must be available to provide reliable input to accident analyses for licensing purposes.

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

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

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

4.2 MOCKUP 1B

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

Prior to the test itself, Mockup 1B capsule was filled with sodium using the Task B fill loop. Sodium was circulated through the capsule at ~ 800 to 900°F for 30 hours to remove traces of oxide and to generally clean up the loop. Following this an indicated plugging temperature of less than 250°F was obtained from the cleanup loop plugging meter. (This corresponds to sodium oxygen level of less than 5 ppm.) During the final plugging run it was found that plugging would not occur above the sodium freezing temperature indicating very high purity. The sodium capsule was isolated from the sodium fill system and maintained at ~ 500 to 600° through operation on integral heaters provided for this purpose. This condition was maintained for more than 60 hours prior to the installation of the capsule into the test stand (Figure 4-1). After installation, the temperature was maintained at between 400 and 500°F until the actual test sequence was initiated.

A loss-of-flow test was completed using the Mockup 1B facility (Figures 4-2 and 4-3). Prior to actuating the electric heaters (fuel simulators), data were taken under steady state and transient flow conditions to establish loop characteristics. Following this, the fuel simulator (Figure 4-4) power was brought up to ≈ 19 kW each. At this point, the test section, ΔT , was recorded at different flow rates to permit calibration of the flowmeters. The fuel simulator power was then increased to ≈ 24 kW each and was held at this power for ≈ 3.5 minutes before the flow coastdown test was initiated. High speed oscillograph and tape recorders were started ≈ 5 seconds before the sodium pump was scrambled to initiate loss-of-flow test.

* Contributors are listed in Appendix B.

Figure 4-1. Mockup 1B Capsule and Stand

Figure 4-2. Mockup-1B

Figure 4-3. Mockup 1B Schematic

Figure 4-4. Mockup 1B Fuel Simulator Bundle

A preliminary inspection of the test data indicates that boiling started (indicated by flow oscillations) at 2.65 seconds. Data also shows a maximum sodium expulsion rate of ~ 21 gpm (~ 40 ft/sec) and a peak temperature on the test section linear (thermocouples 8 and 9, Figures 4-2 and 4-3) of $\sim 1800^\circ\text{F}$ at the time the final two heaters failed. The following sequence of events occurred during the loss-of-flow test:

T = 0.0 sec	Pump scram
T = 0.31 sec	Flow dropped to 10% of initial flow rate
T = 0.55 sec	Flow rate levels out at low value
T = 2.65 sec	Flow oscillation begins
T = 4.27 sec	First heater failure
T = 5.420 sec	Largest and most rapid expulsion begins
T = 5.432 sec	Second heater failure
T = 5.445 sec	Third heater failure
T = 6.5 sec	Flow and temperature oscillation begin to decrease

In addition to the temperature, flow and pressure data, the capsule was also instrumented with piezoelectric accelerometers mounted in locations 1 through 5 (see Figure 4-3) as was done for the Mockup 1A test. Mockup 1B Instrumentation included, in addition, a piezoelectric pressure transducer at the lower end of the capsule. The signals generated by these piezoelectric transducers were recorded on magnetic tape.

Analysis of the tape-recorded data is in progress. The responses of accelerometers 1 and 2 located just above the water cooling jacket appeared similar to those observed at same location in the previous (Mockup 1A) test. Noise "bursts" on the order of 1G in amplitude appeared to coincide in time with peaks on the EM flowmeter traces. Similar "bursts" appeared in the output of accelerometer number 5. Frequency analysis of these data are planned.

There was no apparent signal detected by accelerometer 3 and 4 located radially opposite the fuel pin simulators. This result is significantly different from the observation of Mockup 1A where acceleration on the order of 10 to 15G was measured at the same location. A possible explanation for this behavior would be the existence of a large vapor volume in the test section interfering with the radial transmission of acoustic signals. The voiding behavior of the Mockup 1B test seems to support this postulate in that the total response seems to be strongly vapor dominated.

A modified version of the BUBBLE/VOIDUM package is available and is being used to analyze the dynamic void behavior of the Mockup 1B test. An instability was found in the original BUBBLE/VOIDUM package but this has been corrected and predictions of void initiation consistent with the Mockup 1A results have been obtained. Work is in progress on analysis of the void growth and collapse behavior in the Mockup 1B geometry.

4.3 CAPSULE C11A

Capsule C11A is an in-pile vehicle for the study of the latter stages of a loss-of-flow accident. This is a small, nonflowing capsule in which boiling will be initiated in a manner similar to that used on the C10X capsule. In this test the upper portion of a fuel pin will be simulated including the fission gas plenum. This test will investigate the voiding process, fuel pin failure mechanism, post-failure fuel movement and fission gas release rate resulting from a local under-cooling condition.

The Experimental Proposal for the C11A capsule has been issued. Work is now in progress on the Design Basis Specification for this capsule.

4.4 MODEL DEVELOPMENT

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

Work continued on the incorporation of the gas conductivity and thermal expansion options into the THTE program. The routines required for second round data processing have been debugged. The computational link routines have been modified and checkout procedures are in progress.

4.5 LIAISON

Plans and results of this program are interchanged with those of other organizations concerned with safety. A presentation describing current Task C-2 tests and results was prepared and given for visiting teams from AEC/DRL, ANL, and HEDL.

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 with an electron probe microanalyzer, 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 is being studied in irradiated samples.

5.2 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR IN GETR

5.2.1 Central-Hole Formation at Reactor Startup — Capsule E2R

This capsule, containing 88% and 94% T.D. 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 seven times for periods of 1, 10, 15, 34, 60 and 120 (twice) minutes making a total exposure of 6 hours. Neutrographs taken after each exposure revealed central hole formation in the 88% TD pellets between 11 and 26 minutes.

After the fifth irradiation cycle that lasted 1 hour (accumulated exposure of 2 hours), some expansion of the center voids in the 88% T.D. fuel pellets was noted. No center void was observed in the 94% T.D. fuel. The sixth irradiation cycle lasted 2 hours (accumulated exposure of 4 hours) and produced little void growth in the 88% T.D. fuel. Again no center void was observed in the 94% T.D. fuel. The seventh irradiation cycle of 2 hours increased the accumulated exposure to 6 hours. Neutrographic results of this irradiation period are being processed.

5.3 ELECTRON MICROPROBE EXAMINATION

5.3.1 Examinations Related to Fuel - Fission Product - Cladding Reactions

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

The results of the examination of this longitudinal sample were reported last quarter.⁵⁻¹ An additional region in this sample that showed a modest amount of general attack was examined this quarter. Iodine, cesium, tellurium, molybdenum and oxygen were identified in the reaction phase along with the cladding alloy elements iron, chromium and nickel. With the exception of iodine the same fission products were identified in the specimens reported previously.

5.3.1.2 Examination of Unirradiated Sample of Type 304 ELC

The capability to detect carbon and nitrogen was established for the first time on a sample of extra low carbon (80 ppm) Type-304 stainless steel sheet. This material had been heat treated in a 550 to 1150°C thermal gradient for 100 hours in a fueled capsule containing Cs₂O. The sample had been mounted in a low melting nonferrous alloy to minimize the carbon fluorescence that occurs in conventional organic mounting compounds. Cesium, nitrogen, carbon, silicon and cladding alloy elements were detected during the initial examination, in which the sample was prepared in the usual manner with diamond paste. Because the diamond paste could have provided the detected carbon in this low carbon alloy, the surface was ground, then repolished with an alumina compound. Re-examination revealed no carbon was present and it is concluded that the carbon detected in the first examination had been a contaminant. The silicon was also identified and this element was found in a lower concentration at the surface than elsewhere. Where it was present at the surface it was concentrated in discrete locations.

Figure 5-1 illustrates the features of the stainless steel sheet after the initial polish with diamond paste. A 12 μm nitride layer was present on the surface of the sheet adjacent to the fuel which was also in the capsule. The nitrogen probably originated in the fuel and reacted with the cladding to form a surface layer of Cr₂N. Cesium was also found in the nitride layer.

* Contributors are listed in Appendix B.

Figure 5-1. Negative Specimen Current Image and Nitrogen, Chromium and Cesium X-Ray Images of a Disc of Extra Low Carbon 304 Exposed to Cesium in a Temperature Gradient for 100 Hours at 595°C

After repolishing with alumina, the nitride layer remained, as shown in Figure 5-2. The nitride layer is completely devoid of carbon. The occasional silicon concentrations may also be seen in the figure. The scanning electron microscope view shows the different textures of the nitride layer, the silica particles and the unreacted stainless surface.

5.3.2 Examinations Related to Plutonium Redistribution

5.3.2.1 Examination of Irradiated Pins F1B, F1C, F1D, and F1E

Mid-axial sections of four pins irradiated in EBR-II to a burnup of 1800 MWd/Te were examined in the electron microprobe to establish the effect of as-fabricated stoichiometry on plutonium redistribution.⁵⁻² The effect of stoichiometry on plutonium migration is noted in Table 5-1 and Figures 5-3 through 5-5. Under near identical operating conditions it is apparent that the hyperstoichiometric sample (F1C), with O/M = 2.03, exhibited a significant amount of plutonium migration in comparison to the stoichiometric (F1B) and hypostoichiometric (F1D) samples with O/M = 1.98. Pin F1E which was also hypostoichiometric exhibited a slight shift in plutonium but its lower density may have resulted in a different operation temperature and rate of fuel migration.

Table 5-1
EFFECT OF STOICHIOMETRY ON PLUTONIUM MIGRATION

Pin	O/M	Linear Power (kW/ft)	Smeared Density (% T.D.)	Pu Content in Center Region	
				Pre-Irradiation (%)	Post-Irradiation (%)
F1D	1.98	16.2	95.3	20	20
F1B	2.00	16.4	96.2	20	20
F1C	2.03	16.8	95.6	20	25
F1E	1.98	14.8	89.2	20	~ 22

5.4 FUEL – FISSION PRODUCT – CLADDING REACTIONS

5.4.1 Fission Product Migration and Reactions

The modes of combination of cesium, tellurium and iodine as well as the effects of the Cs/Te and Cs/I ratios on intergranular attack are being investigated. These fission products were loaded into the first inch of a set of capsules which were filled with (U-25 % Pu)O₂ and then heat treated for 100 hours in a thermal gradient of 550 to 1100°C (with the initial location of the fission products being the hot end). Gamma scans before and after the heat treatment established the migration. Features of the complete set of tests, some of which were reported last quarter, are summarized in Table 5-2.

The tests discussed below are the Series VI and VII. The two capsules containing iodine were heat treated as noted above and subsequently annealed at 200°C for 6-3/4 hours. The results are as follows:

1. The capsule containing cesium alone (in the form of Cs₂¹³⁷O) exhibited a reasonably uniform distribution of the cesium along the 6-inch length, but with peaks at the hot and cold ends. An additional 100-hour test under the same conditions showed a slight increase in cesium migration. Figure 5-6 displays the results.
2. In the capsule containing cesium and (nonradioactive) tellurium in the ratio of 1/1, the cesium migrated to a position 1-1/2 inches from the cold end, corresponding to a temperature of about 725°C, as shown in Figure 5-7. Based on previous tests, it is assumed that the tellurium combined with the cesium probably as Cs₂Te or Cs₂TeO₃, and migrated with it. Since in this test the tellurium was nonradioactive, it is not possible to determine its final location.

Figure 5-2. Negative Specimen Current Image, Silicon and Nitrogen X-Ray Images and Scanning Electron Microscope Image of a Disc of Extra Low Carbon 304 Exposed to Cesium in a Temperature Gradient for 100 Hours at 595°C

Figure 5-3. Photomicrographs of Specimens F1B and F1D and Corresponding Radial Distributions of $Pu/(Pu+U)$

Figure 5-4. Photomicrographs of Specimen F1C and Corresponding Radial Distributions of $Pu/(Pu+U)$

Figure 5-5. Photomicrographs of Specimen F1E and Corresponding Radial Distributions of $Pu/(Pu+U)$

Figure 5-6. Migration of Cesium Loaded at Hot End of a (U-25 w/o Pu)O_{2.00}-Fueled Capsule. Heat Treated in 550-1100°C or 400-750°C Thermal Gradients

Figure 5-7. Migration of Cesium and (Cesium + Tellurium) in (U-25 w/o Pu)O_{2.00}-Fueled Stainless Steel Capsule Heated 100 Hours in 1100-550°C Gradient

Table 5-2
SUMMARY OF FISSION PRODUCT THERMAL GRADIENT CAPSULE EXPERIMENTS^(a)

Test Series	O/M	Temperature (°C)	Capsule	Additive	Results
I	1.967	660-1550	Mo	Cs ₂ MoO ₄ /U	Cs ₂ O increased oxygen redistribution rate
II-A	1.967	700-1550	Mo	Cs ₂ MoO ₄ /U	Confirmed the results of Series I. Surface reaction between fuel and cladding in 600-750°C range.
II-B	1.967	550-1550	Mo	Cs ₂ MoO ₄ /U	
II-C	1.967	660-1550	Mo	Cs ₂ MoO ₄ /U	
III-A	1.967	550-1150	316 SS	Cs ₂ MoO ₄ /U	Cs participated in surface reaction noted above. Cs ₂ O reacted in either form.
III-B	1.967	550-1150	316 SS	Cs ₂ O	
III-C	1.967	550-1150	316 SS	None	
IV-A	1.94	550-1150	316 SS	Cs ₂ O	Amount of CsO retained at high temperature end increased with O/M. Intergranular attack with O/M > 2.00.
IV-B	1.967	550-1150	316 SS	Cs ₂ O	
IV-C	2.00	550-1150	316 SS	Cs ₂ O	
IV-D	2.01	550-1150	316 SS	Cs ₂ O	
IV-E	2.00	550-1150	316 SS	Cs ₂ O Excess	Excess Cs ₂ O ruptured Capsule E.
V-A	1.97	550-1150 ^(b)	316 SS	Cs ₂ O + Te ^(c)	Te reduced migration of Cs ₂ O toward high temperature end. Cs and Te appear to cohabit, possibly as Cs ₂ TeO ₃ . No reaction with iodine. I did not affect Cs ₂ O redistribution. Cs and Te appear to cohabit; Te lessens migration of Cs ₂ O to cold end.
V-B	1.97	550-1150 ^(b)	316 SS	Cs ₂ O + I ^(d)	
V-C	2.00	550-1150 ^(b)	316 SS	Cs ₂ O + I ^(d)	
V-D	2.00	550-1150 ^(b)	316 SS	Cs ₂ O + Te ^(c)	
V-E	2.00	550-1150 ^(b)	316 SS	Cs ₂ O Excess	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.
V-F	2.00	550-1150 ^(b)	316 SS	Cs ₂ O Excess	
VI-A	2.00	550-1150	316 SS	Cs ₂ O + Te ^(d)	Cs and Te cohabit. Excess Cs deposited along length, migrating to cold end, but was inhibited by Te and less migrated than in D. Cs and Te cohabit at about 725°C. Te restricted Cs migration to cold end.
VI-B	2.00	550-1150	316 SS	Cs ₂ O + Te ^(c)	
VI-C	2.00	550-1150	316 SS	Te	

Table 5-2 (Continued)

Test Series	O/M	Temperature (°C)	Capsule	Additive	Results
VI-D	2.00	550-1150	316 SS	Cs ₂ O	Cs deposited along length, more concentrated at cold end than when Cs + I concentrated at cold end in VI-A.
VI-E	2.00	550-1150	316 SS	Cs ₂ O + I ^(d)	Cs + I concentrated at cold end; excess Cs deposited along capsule length. Slight I migration after 200°C soak.
VI-F	2.00	550-1150	316 SS	Cs ₂ O + I ^(c)	Cs and I concentrated at cold end as CsI. No migration after 200°C soak.
VII-A	2.00	550-1150	316 SS	Cs ₂ O + Mo ^(c)	} Mo remained at hot end; Cs ₂ O deposited along length.
VII-B	2.00	550-1150	316 SS	Cs ₂ O + Mo ^(e)	
VII-C	2.00	550-1150	316 SS	MoO ₃ + Mo	
VII-D	2.00	550-1150	316 SS	MoO ₃ + Mo ^(f)	
VII-E	1.96	550-1150	316 SS	MoO ₃ + Mo	
VII-F	2.00	550-1150	316 SS	Cs ₂ O (Reheat of Capsule VI-D)	

- (a) Fueled capsules heated 100 hours in thermal gradient
 (b) Temperature drifted gradually during the heat treatment due to deterioration of the heater windings
 (c) 1/1 ratio of Cs/Te or Cs/I or Cs/Mo + MoO₃
 (d) 4/1 ratio of Cs/Te or Cs/I
 (e) 2/1 ratio of Cs/Mo
 (f) Additive positioned at cold end

3. In the capsule containing cesium and tellurium in a 4/1 ratio, cesium migrated to the cold end, depositing en route. Part of the cesium combined with the tellurium as in the above reaction and the excess proceeded along the thermal gradient. Figure 5-7 illustrates cesium migration with these two ratios of tellurium. A graph of cesium migration alone is superimposed to show the similarity between its movement and that of the mixture with the excess cesium. Since the normal fission yields of these two isotopes is $\sim 4/1$, the experiment is a plausible explanation of the finding in irradiated fuel pins that cesium is always found in the presence of tellurium, but tellurium is not always present where cesium is detected.
4. In the capsule containing cesium and iodine in a 1/1 ratio, both elements were concentrated at the hot and cold ends. No cesium or iodine were observed between these extremities. An additional 200°C isothermal anneal (which is above the boiling point of I, 185°C) caused no significant change in the iodine locations implying that it is combined with cesium as CsI. A graph of the axial activity of these isotopes is presented as Figure 5-8.
5. The capsule containing cesium and iodine in the 4/1 ratio showed a similar concentration of cesium and iodine at the hot and cold extremities. The excess cesium was distributed along the capsule length in the manner of items 1 or 3 above. No significant iodine movement occurred after the 200°C isothermal anneal (which is above the boiling point of iodine but below the boiling point of cesium iodide). An axial activity graph of these isotopes in the 4/1 ratio is shown in Figure 5-9.
6. The results of Pin VI-C which contained Te alone are not yet available.

The behavior of Mo/MoO₃ in a fueled thermal gradient environment was investigated in an additional 100-hour test of the same type. Six Type 316 stainless steel capsules were loaded with radioactive cesium or molybdenum, alone or combined. Gamma scans yielded the following results:

1. In all cases molybdenum remained at its original location.
2. In capsules containing both cesium and molybdenum, the molybdenum behaved as above, while the cesium concentrated at the hot end and also distributed itself along the length of the capsule. This latter behavior was the same as exhibited by cesium above or in excess with other isotopes (Figures 5-6, 5-7, 5-9). Metallographic results are not yet available to indicate whether the two elements interacted at the hot end. Table 5-2 summarizes the testing on this program.

5.4.2 Cladding Modification Testing

In this program, capsules prepared with cladding alloys in varied thermomechanical states have been reacted in a 550 to 1100°C thermal gradient with Cs₂O in a fuel environment. Initial findings were reported last quarter. Samples of fuel used in this test program were analyzed for cesium, carbon and nitrogen. Results show that the fuel contains 3 ppm C, 10 ppm N₂ and < 3 ppm Cs.

In addition to the previous tests at 550 to 1100°C , tests were instituted to investigate cladding reactions at temperatures closer to those normally experienced in an LMFBR. Hence tests were made at lower temperatures and in a less steep thermal gradient. An initial pair of fueled rods with Cs₂O additions were tested for 100 hours within a 400 to 750°C thermal gradient. The cesia was radioactive and gamma counting after the test showed that a substantial portion of the Cs₂O had migrated down the thermal gradient. The Cs₂O concentration along the tube length is shown in Figure 5-6. This shows similar plots of Cs₂O migration after 100 and 200 hours in a 500 to 1100°C thermal gradient. Longer time exposures would be expected to enhance the Cs₂O movement in this lower temperature regime. Additional capsules will be treated within the 500 to 750°C and 500 to 850°C temperature ranges to more closely reproduce cladding temperatures of commercial LMFBRs. They will also be extended to 1000 hour exposures.

5.5 FUEL-SODIUM REACTION KINETICS

An instrumented stainless steel vessel* is being constructed to investigate reactions between sodium and mixed oxide. Mixed oxide fuel clad in stainless steel will be exposed to high temperature sodium in this vessel. The vessel is equipped with a transducer for continuously monitoring the cladding dimensional changes in the region of the exposed fuel, so that swelling rate data can be obtained. Fuel of several stoichiometries and plutonium compositions will be tested in sodium at several levels of oxygen purity. Testing will begin during the next quarter.

5.6 MEASUREMENT OF FUEL OXYGEN POTENTIAL

An electro-chemical probe is being constructed to measure the oxygen potential ($\Delta\bar{G}_{O_2}$) as a function of radius across irradiated fuel. This parameter, the partial molar free energy of oxygen, is a thermodynamic characteristic and may be related to: the stoichiometry of unirradiated fuel, the oxidation state of solid fission products, fission product-cladding reactions and fuel reactions with sodium. A high temperature galvanic cell utilizing solid electrolytes reversible to oxygen ions only will be the basic test unit. It will be cycled over the range of about 650 to 800°C and the resultant emfs measured. By using reference electrodes of known oxygen potentials such as Cu^+ , Fe^{++} or Ni^{++} , the oxygen potential of the mixed oxide will be determined. Because the oxygen potential of mixed oxide varies so drastically with stoichiometry ($\Delta\bar{G}_{O_2}$ for $MO_{2\pm x}$ ranges from about -55 to -165 kcal/mole) a less commonly used electrolyte, $ThO_2-Y_2O_3$, that covers this wide range will be employed; Ni/NiO_2 will serve as the electrode. ZrO_2-CaO electrolytes are satisfactory for measurements of $\Delta\bar{G}_{O_2} \geq 85$ kcal/mole and will be used for trial measurements with UO_{2+x} . Both electrolytes have been obtained as small diameter pointed tubes. These are being loaded with the Ni/NiO reference electrodes and will then be assembled to serve as probes. $UO_{2.02}$ and $UO_{2.03}$ pellets have been manufactured for initial testing of high $\Delta\bar{G}_{O_2}$ values ($(UO_2-Ce)O_{2\pm x}$ pellets are being prepared to cover the wide range of values typical of mixed oxides).

5.7 MELTING POINTS OF IRRADIATED MIXED OXIDES

A cooperative program has been developed with Los Alamos Scientific Laboratory to measure the melting points of mixed-oxide fuels irradiated in EBR-II. A differential thermal analysis (DTA) furnace, installed in the hot cells at LASL, is being used to determine the thermal arrests at solidus and liquidus temperatures. Irradiated fuel extracted from the cladding and loaded into sealed tungsten capsules is heated into the melting range. Emerging light from the incandescent sample is reflected through a shielding window to a micro-optical pyrometer for the measurement of sample temperatures, and to a pair of photosensitive detectors that yield absolute and differential temperature values on a continuous recorder.

The melting points of unirradiated controls and seven samples with a burnup range of 4.3 to 10.6 at.% are being determined. Their identities are listed in Table 5-3 along with first measurements. The results on unirradiated $U-25$ wt.% Pu $O_{1.96}$ fuel show a solidus temperature of 2790°C and a liquidus temperature of 2840°C. Similar measurements on irradiated (U-25 wt.% Pu) $O_{1.98}$ from F3B-7 at 4.3 at.% burnup are 30 to 35°C lower. This is keeping with the findings of Krankota and Craig⁵⁻³ who determined melting points on irradiated and unirradiated mixed oxides by the V-filament method and found lower melting points in irradiated fuel. While these values represent single readings (measurement precision has not yet been established), they do show a close correspondence to unirradiated values reported by Aitken and Evans.⁽⁵⁻⁴⁾ Those experimenters showed liquidus-solidus values of 2780 and 2845°C for (U-20 wt.% Pu) $O_{1.96}$. The composition measured by Aitken and Evans has a melting point that is estimated to be about 25°C above that measured in this program, so that the LASL values are within about 15°C of the anticipated temperatures.

* 2 inches in diameter by 8 inches long.

Figure 5-8. Migration of Cesium and Iodine Loaded at the Hot End of a Fueled Capsule: Heat Treated 100 Hours in a 550-1100°C Thermal Gradient, Then 6 Hours at 200°C (Cs/I Ratio = 1/1)

Figure 5-9. Migration of Cesium and Iodine Loaded at the Hot End of a Fueled Capsule: Heat Treated 100 Hours in a 550-1100°C Thermal Gradient, Then 6 Hours at 200°C (Cs/I Ratio = 4/1)

Table 5-3
MELTING POINT DETERMINATION

Sample	Burnup (at. %)		Wt % PuO ₂	O/M	Fuel Batches	Measured Melting Points (°C)	
	Peak	Average				Solidus	Liquidus
1. Irradiated Samples for Melting Point Determination							
F0K-H	5.6	5.4	20.1	2.00	B44, B47, B48, B50		
F2X-Powder	10.2	9.7	20	1.99	B101-1, B101-2, B101-3		
F2G	10.8	10.0	20	1.98	B26, B47, B49, B53, B54		
	10.8	10.6	20	1.98	B26, B47, B49, B53, B54		
F3B-7	4.85	4.3	24.9	1.98	B237	2755	2810
E1H-29-C	9.2	9.0	25.2	1.97	B237		
E1H-29-A	9.2	7.6	25.2	1.97	B237		
2. Unirradiated Controls for Melting Point Determination							
			25	2.00			
			25	1.96		2790	2840

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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 are determined under conditions simulating as closely as possible those expected in a fast ceramic reactor.

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

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

A summary of the fast flux irradiation tests in the EBR-II is given in Table 1-3. Calculated values for total fluence, fluence greater than 0.1 MeV, and fuel burnup values in MWd/Te and atom percent (taking into account fuel depletion) are incorporated into the summary table. The fluence, power, and burnup values given in Table 1-3 were calculated using the fission rates provided in the latest revision of the "Experimenters' Guide for Irradiations in EBR-II" as a basis.

6.2 SCOPING AND STATISTICAL CONFIDENCE IRRADIATIONS

6.2.1 F2 Experiment (Scoping Irradiations to High Burnup)

A topical report evaluating the irradiation of seven unfailed, encapsulated pins which operated at ~16 kW/ft peak with ~1075°F maximum cladding temperature to ~7 at.% burnup (F2C, H, R, T) and to ~11 at.% (F2G, V, X) is being prepared.

6.2.2 F3A Experiment (Unencapsulated Fuel Pin Irradiation)

The 16 pin F3A experiment operated at >16 kW/ft peak with ~1150°F maximum cladding temperature to ~6 at.% burnup. A topical report evaluating the results of the post irradiation examination is being prepared.

6.2.3 F3B Experiment (Fuel Void Deployment and Clad Alloy Studies)

Additional metallography has been conducted on two of the six F3B pins which are being examined. This experiment operated at ~16 kW/ft and ~1150°F maximum cladding temperature to ~10 at.% burnup. The interim examination is now complete. Table 6-1 summarizes the fabrication parameters and the irradiation conditions for these pins.

* Contributors are listed in Appendix B.

Table 6-1

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

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

* Initial diameter of central hole was 0.048-inch

** Solution annealed

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

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

An unusual two-phase fuel structure was noted at the central void of the transverse sample from the peak flux region of pin E1H-30. The two phases appeared similar to the material found earlier in pin F2G. The lighter of the two had a dendritic structure as shown in Figure 6-1. Both phases reacted with the fuel etchant and were completely removed.

The central void in each of the two samples from pin E1H-23, which was loaded with annular fuel, was irregular in shape, and one section near the top of the fuel had fingers of fuel extending into the void, as shown in Figure 6-2. The fabricated central hole in the fuel pellets was 0.048 inch in diameter. The distance across the void near the center of the fuel column was approximately 0.055 inch, and near the top of the fuel column it varied from 0.050 to 0.062 inch.

The transverse samples from the three pins clad in Type 316 stainless steel, E1H-21, 23, and 26, showed reaction layers about one-half mil thick. Intergranular attack to a maximum depth of one mil was found only in pin E1H-21 which contained powder fuel. The maximum operating temperatures indicated in Figure 6-3 are estimated from the measured burnup. The longitudinal samples revealed two areas of intergranular attack on pin E1H-21, four areas on E1H-23, and one area on E1H-26. The depth of penetration was approximately two mils on both E1H-21 and 23 but extended to about four mils on E1H-26, as shown in Figure 6-3.

It is usually difficult to distinguish the gray zones in the fuel-cladding gap from the mounting material. Finely divided alumina added to the mounting material of the specimen, when viewed under polarized light, appeared very bright. Figure 6-4 shows photographs of the same area under bright field and polarized light illumination indicating how the gray phase of fission products can be distinguished from the mounting compound with polarized light.

Slight localized intergranular attack (< 1 mil) was found only on the longitudinal specimen from pin E1H-29, which was clad with 13% cold-worked Type 304 stainless steel. General reaction was observed on both the transverse and longitudinal specimens and occurred to a maximum depth of about one mil near the top of the fuel column.

Extensive general cladding reaction occurred in fuel pin E1H-30. This cladding was from the same heat as the 13% cold-worked Type 304 stainless steel used for fuel pin E1H-29, but it received a Garofalo-type heat treatment after the cold work. The heat treatment of 24 hours at 900°F followed by 216 hours at 1300°F and rapid air cooling was intended to improve creep strength. The extent of the reaction was greatest near the top of the fuel where cladding temperature was higher, Figure 6-5. In some areas the cladding attack appeared to progress intergranularly to a depth of about one mil. The longitudinal specimen near the top of the fuel column showed no intergranular attack, but general attack extended almost to the very top of the fuel column. The central void at the top of the fuel of this pin is nearly filled with metallic fission product ingots as seen in Figure 6-6.

Two transverse samples were examined from the failed fuel pin E1H-33 which was clad in annealed Type 304 stainless steel. One was from a grossly failed region near the center of the fuel column, and the other was from a region containing longitudinal splits near the top of the fuel column. Both cross sections as illustrated in Figure 6-7 show sodium-fuel reaction products completely around the periphery of the fuel, and the sample from the grossly failed area shows a section of cladding that melted. No extensive cladding reaction or corrosion was observed, but numerous regions of both general and intergranular attack to depths of about one mil were found on the sample near the top of the fuel column.

The high cladding temperature sections from each of the four different types of cladding were etched with Vilella's Reagent to reveal sigma phase. All of the Type 316 stainless steel cladding used was from the same heat. Significant amounts of sigma phase were found only in pin E1H-30 clad with 13% cold-worked Type 304 stainless steel with the Garafalo-type heat treatment. Sigma phase was present in this alloy before irradiation.

All of the transverse specimens were electroetched in 10% oxalic acid for 2 seconds to reveal cladding structure. All showed grain boundary precipitation across the cladding, a denuded zone at the fuel surface, and a heavy matrix precipitation adjacent to the denuded zone. In addition, the cold-worked cladding showed substantial precipitation along slip planes. Figure 6-8 shows the typical structure for Type 316 stainless steel from the sections of pin E1H-23 at regions of both peak flux and peak cladding temperature.

100X

100X

Figure 6-1. Two-Phase Structure in the Central Void of Fuel Pin E1H-30 Near the Peak Flux Region

Figure 6-2. Cross Section of the Top of the Annular Fuel Column of Fuel Pin E1H-23

6.2.4 F4 Experiment (Low Density Fuel Performance)

The post-irradiation examination of the seven F4 encapsulated pins, with fuel densities of 84 to 87% theoretical and irradiated at >16 kW/ft with maximum cladding temperature of $\sim 1100^{\circ}\text{F}$ to ~ 9 at.% burnup, is underway. Complete metallographic examination of two of the pins (F4A, F4E) is planned to provide control information for two similar F4 pins (F4G, F4J) to be used in the F20 experiment. The F20 experiment will investigate power-to-melting parameters in both fresh and irradiated fuel. Pins F4A and F4E are scheduled for destructive examination at the LASL facility.

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

The post-irradiation examination of fuel pin F9C-31 is underway. The unencapsulated fuel pin, irradiated at ~ 13 kW/ft peak with $\sim 1035^{\circ}\text{F}$ maximum cladding temperature to ~ 7.3 at.% burnup, appears to be in excellent condition. Visual examination, gamma scanning, neutrography, scanning with the Pulsed Eddy Current Cladding Analyzer, and profilometry are complete. Neither the neutron radiographs nor the gamma scans showed any abnormalities. The pulsed eddy current examination indicated a very slight irregularity on the inside diameter 1.6 inches above the bottom of the fuel. The maximum radial extent of the anomaly was 1.6 mils or 0.64 percent. The pin is currently being sectioned for metallographic examination, for burnup and cladding density samples, and for burst testing of the cladding.

The post-irradiation examination of pin F9C-13 will be conducted at the LASL facility.

The data package for reconstitution of the second F9C Mark E subassembly (F9C-2) is being prepared.

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

The irradiation of the F12P and F12Q pins with cladding temperature $\sim 1150^{\circ}\text{F}$ in GETR to $>100,000$ MWd/Te is complete. The pins are being prepared for post-irradiation examination at the LASL facility.

250X

E1H-21 — 1050°F MAXIMUM CLADDING INSIDE SURFACE TEMPERATURE*

250X

E1H-23 — 1085°F MAXIMUM CLADDING INSIDE SURFACE TEMPERATURE*

250X

E1H-26 — 1095°F MAXIMUM CLADDING INSIDE SURFACE TEMPERATURE*

*BASED ON MEASURED BURNUP

Figure 6-3. Regions of Intergranular Attack at the Top of the Fuel Column of Pins Clad in Annealed Type-316 Stainless Steel

Figure 6-4. Fuel-Cladding Interface Showing How Fission Products Can be Distinguished from Mounting Material

6.3 ELEVATED CLADDING TEMPERATURE IRRADIATIONS

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

The post-irradiation examination of three of the nineteen encapsulated F5 pins, which operated at ~ 16 kW/ft peak with $\sim 1200^\circ\text{F}$ maximum cladding temperature to ~ 6 at.% burnup, is continuing. The transverse section in the peak flux zone of pin F5Q (50% cold-worked Type 316 stainless steel) has been mounted for microprobe examination of the detached rim of unidentified material between the fuel and cladding.

Analyses of fission product neodymium were made on F5F, F5M, and F5Q to determine the actual burnup achieved by these pins. The results are shown in Table 6-2. The $\sim 20\%$ discrepancy between the measured and calculated burnup is larger than the discrepancy observed in previous irradiations. The fission gas release data for pins F5F, F5M, and F5Q are shown in Table 6-3. The near 100% release of Xe and Kr for these pins is somewhat higher than generally observed at similar power and burnup. The fission gas analyses for the three fuel pins are shown in Table 6-4. The Xe and Kr concentrations found by gas chromatography vary from ~ 84 to 86% for the three fuel pins, while the mass spectrometry results for the F5M pin show a concentration of $\sim 91\%$. The gas chromatography results are believed to be the more reliable.

Anomalies at the cladding inside surface were successfully identified from pulsed eddy current (PEC) scans prior to the destructive examination. The two major anomalous indications from the PEC scans were: (1) near the fuel-upper blanket interface of F5M, and (2) the peak flux region of F5Q. The metallographic examination revealed ~ 1 mil (maximum) localized intergranular attack at the fuel upper blanket interface of F5M, and a detached rim of material between the fuel and the cladding in the peak flux region of F5Q. Other areas where possible cladding attack was indicated included the peak flux region and the plenum of all three pins. The peak flux region indications were all correlated with irregular fuel-cladding reaction zones, either adherent to the cladding or detached. The plenum indications were correlated with fabrication flaws.

100X

E1H-30 - 1070°F MAXIMUM CLADDING INSIDE SURFACE TEMPERATURE*

500X

E1H-30 - 985°F CLADDING INSIDE SURFACE TEMPERATURE*

*BASED ON MEASURED BURNUP

Figure 6-5. Cross Sections of Fuel Pin E1H-30 Showing General Reaction (~3 mils Reduction of Cladding Thickness) and Intergranular Attack (<1 mil Penetration)

Figure 6-6. Longitudinal Section of the Top of the Fuel Column of Fuel Pin E1H-30

NEAR TOP OF FUEL COLUMN

NEAR CENTER OF FUEL COLUMN

Figure 6-7. Transverse Cross Sections of Failed Fuel Pin E1H-33

250X

or,

1085°F

11. 11. 1972

Table 6-3
FISSION GAS RELEASE - GROUP F5

Pin	Total (Xe+Kr) Released ^(a) cm ³ at STP	Gas Release Based on Measured Burnup (%)
F5F	77.0	95.8
F5M	80.0	~ 100
F5Q	74.3	91.7

(a) Based on gas chromatographic results

Table 6-4
FISSION GAS COMPOSITION - GROUP F5

Component	Percent of Total (Gas Chromatography)		
	F5F	F5M	F5Q
Xenon	74.05	74.81 (80.5) ^(a)	72.46
Krypton	11.30	11.18 (10.8) ^(a)	11.37
Helium	13.82	13.20 (8.2) ^(a)	15.51
Hydrogen	0.16	0.23 (0.3) ^(a)	0.17
Nitrogen	ND	0.06 (0.2) ^(a)	ND
Carbon Monoxide	0.06	0.09	0.03
Carbon Dioxide	0.03	0.06 (0.01) ^(a)	ND
Argon	ND	ND (0.004) ^(a)	ND
Neon	ND	ND -	ND
Oxygen	ND	ND -	ND
Methane	ND	ND -	ND
Total	99.41	99.62 (100.014) ^(a)	99.54

ND Not detected

(a) (0-00) Mass Spectroscopy

6.3.2 F10A, F10B Experiments (Cladding Performance at Peak LMFBR Cladding Temperatures)

These two 19 encapsulated pin experiments to evaluate performance of several clad alloys will operate at ~ 16 kW/ft peak with $\sim 1270^\circ\text{F}$ maximum cladding temperature to > 11 at.% target burnup. Rework of the triangular flow regulators to overcome the tolerance buildup problem that prevented assembly of the fuel pin bundle into the channel at EBR-II is complete. It now appears that the hexagonal channel that was supplied by ANL will also need reworking. The channel has been returned to the EBR-II shop at their request.

6.3.3 F11A Experiment (37 Unencapsulated Pin, Wire Spaced Subassembly at Peak LMFBR Cladding Temperature)

The F11A will operate at ~ 15.5 kW/ft peak with $\sim 1280^\circ\text{F}$ maximum cladding temperature to > 11 at.% target burnup. The fuel pins have been shipped to EBR-II for assembly of the experiment.

6.4 IRRADIATIONS TO INVESTIGATE OPERATING LIMITS

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

The F20 Experiment (with 19 encapsulated fuel pins to be irradiated at > 16 kW/ft for several days) will determine power-to-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.

Batches of coprecipitated mixed-oxide powder have been blended into one batch sufficient for all pins. Density control of the sintered pellets was more difficult than usual. A reduction of the sintering temperature from the usual 1650°C to 1430°C was required to produce the 90% theoretical density pellets.

The possibility of using melt wires of Mo, Ir, Pt, and Pd inserted into several pellets to determine local fuel temperature is being investigated. Arrangements are being made to impregnate plenum tubes and spacing wire with Kr-85 at high temperatures and pressures to evaluate cladding temperatures. Post-irradiation analysis of these plenum tubes should provide an accurate measure of outlet sodium temperatures. These measurements will aid in determining the actual operating conditions of the experiment.

6.5 SUBASSEMBLY SPACER IRRADIATIONS

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

The F9E experiment will involve 37 fuel pins in a grid-spaced channel operating at ~ 15.5 kW/ft peak with $\sim 1100^\circ\text{F}$ maximum cladding temperature to > 11 at.% target burnup. The fuel pins and the grid spaced channel have been shipped to EBR-II for assembly.

6.5.2 F9F Experiment (FFTF Diagnostic Grid Spaced Fuel Assembly Irradiation)

The F9F experiment with 37 fuel pins will test grid spacer performance under near-FFTF conditions. Approval-in-principle has been received. Procurement of hardware began in advance of receipt of the Approval-in-Principle, as requested by RDT. Approximately 300 ft of 0.230 in. diameter, 20% cold-worked Type 316 stainless steel clad tubing with 15 mil wall thickness has been received from HEDL. The 20% cold worked hexagonal channel has been ordered. Preparation of the final design and the fabrication specification is underway. The EBR-II Project is reviewing the experiment for feasibility and compatibility with the proposed Mark S standardized subassembly for grid spacer experiments.

6.6 IRRADIATION EVALUATION SUPPORT

6.6.1 Fuel-Cladding Attack

A review of Task F fuel pins which have been irradiated in EBR-II and in a thermal flux has been undertaken in order to gain an understanding of the mechanisms responsible for the observed fuel-cladding attack. Approximately 60 fuel pins are presently being evaluated. Variables which are being investigated include O/M ratio, fuel density, burnup, cladding type, cladding temperature, fuel pin power, etc.

Preliminary analysis indicates that:

1. No effect of starting O/M and fuel density is discernible from the data on cladding attack in the range from 1.97 to 2.00 O/M and from 86 to 97% TD (pellet density) respectively.
2. Fuel pin power and cladding temperature are important parameters as expected. High power and high cladding temperature appear to be necessary but not sufficient conditions for appreciable cladding attack (less than 1 mil of cladding attack is considered to be insignificant).
3. No difference in attack behavior can be distinguished between annealed Types 316 and 347 stainless steel at temperatures up to 1150°F.

Fuel chemistry is also presently under investigation. Analysis of the fuel used in the F3A experiment showed low impurity content with the exception of carbon. The carbon content was 420 to 1120 ppm.

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 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.

7.2 F9E DIAGNOSTIC GRID SPACER EXPERIMENT IN EBR-II

7.2.1 F9E Summary

The F9E group irradiation is the first in-pile test of a grid spaced fuel assembly utilizing the straight-wave beam grid-spacer concept, which was conceived by General Electric. F9E employs three different variations of the basic wave beam concept to obtain test information on the effect of these design alternatives.

In addition, F9E is a "diagnostic" irradiation test of an LMFBR grid-spaced assembly. "Diagnostic" is used in the sense that the experiment is designed to produce conditions in the EBR-II assembly that simulate design limiting conditions in the LMFBR. Assembly design methods can be evaluated and improved based on a comparison of predicted and actual grid assembly performance.

In order to obtain diagnostic information for grids irradiated to a high fluence, the loads, stresses, and strains have been amplified through design and material selection. Post-irradiation observations will be compared to analytical predictions in order to increase our understanding of grid performance.

The F9E test will provide a direct comparison of three grid spacer concepts:

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

The F9E experiment has been shipped to EBR-II. The fuel pins will be loaded at EBR-II. As discussed in Section 6, it is planned to start irradiation testing before the end of 1971.

* Contributors are listed in Appendix B.

7.2.2 F9E Description

An EBR-II irradiation assembly Mark F-37 with a modified inner hexagonal channel will be the vehicle for this test (see Figure 7-1). The assembly will contain 37 fuel pins 0.250 inch outside diameter spaced at a pitch of 0.320 inch. This results in a pitch/diameter ratio of 1.28. The inner hexagonal channel is sized to obtain a fuel pin-to-wall spacing of 0.042 inch to minimize the lateral temperature gradients across the outer pins. The nominal test assembly design parameters are summarized in Table 7-1. Sections through the wave beams of the three grid concepts are shown schematically in Figure 7-2. In the combination modified-unmodified straight wave beam full grid, the upper unmodified beam, as characterized by the short leg at the lower end of each beam, is welded to the longer leg of the lower modified beams.

Figure 7-3 contains photographs of two of the grid spacers. Figure 7-3(a) shows a top view of a modified straight wave beam full grid (F) before it has been inserted in a channel. Figure 7-3(b) shows a modified straight wave beam staggered grid (S) after it has been inserted into a notched channel. Figure 7-3(c) shows a section through a straight wave beam full grid after it has been inserted into a channel and had simulated fuel pins inserted. The transverse spacing between the pins shown in the Figure 7-3(c) model is not meant to be prototypical of the more uniform spacings expected in a complete assembly with grid supports over the length of the fuel pins. A schematic layout of the axial location of grids in F9E is shown in Figure 7-4.

The EBR-II test environment requires U-235 enrichment of the fuel to achieve the design operating power in the fuel bundle. This results in a lower fluence-to-burnup ratio than expected in an LMFBR assembly. Consequently, the irradiation induced swelling of the cladding, grid spacer and channel material will be too low relative to the fuel burnup when compared with the LMFBR core conditions. To compensate for this, annealed Type 316 stainless steel will be used to develop, in the low fluence of the test, the material swelling expected in more resistant materials in the higher fluence of an LMFBR assembly.

Comparison of the predicted performance of the grid design in the LMFBR using Type-321 stainless steel with the F9E diagnostic test using annealed Type-316 stainless steel shows that the pin-to-grid differential swelling and elastically calculated stresses are comparable. However, the values of those stresses and contact loads, which are relaxed by irradiation induced creep, are considerably higher in the F9E test. In this regard, the EBR-II is a more severe environment for the diagnostic tests of grid spacers than the design operating conditions expected in the LMFBR.

The grids have been designed to operate for 470 full power days to a total peak fluence of 8.8×10^{22} (n/cm²), which corresponds to a peak burnup of 100,000 MWd/Te. However, an interim examination of the assembly will be conducted in order to determine the condition of eight of the fuel pins which have been designed to be removable from the assembly.

A comparison of predicted pin and grid spacer behaviors is shown in Table 7-2 for three environments; a simulated LMFBR full grid assembly in an FFTF open loop, a staggered grid spaced FFTF driver fuel assembly in an FFTF open loop and a full grid and a staggered grid in the EBR-II F9E assembly. The F9E consists of several different grids combining the diagnostic benefits of both the hypothetical FFTF assemblies listed.

The EBR-II F9E diagnostic assembly results are shown for:

- the end-of-life condition (11,280 hours) which corresponds to a peak burnup of 100,000 MWd/Te
- the scheduled interim examination at 70% of this full irradiation time
- a hypothetical shortened irradiation time of 50% of the full irradiation time.

Table 7-2 shows that bowing loads, pin stresses and grid stresses are comparable to or exceed the values expected in an LMFBR and FFTF at irradiation times of 50% of full life. Although pin-to-grid differential swelling is considerably below LMFBR and FFTF values at 50% of life, because of the reduced irradiation creep of stainless steel in

Figure 7-1. EBR-II Fuel Assembly (F9E)

SECTION A

SECTION A

MODIFIED STRAIGHT WAVE BEAM FULL GRID (F)

SECTION B

SECTION B

SECTION A

COMBINATION MODIFIED-UNMODIFIED STRAIGHT WAVE BEAM FULL GRID (R/F)

SECTION A

MODIFIED STRAIGHT WAVE BEAM STAGGERED GRID (S)

Figure 7-2. Schematic Comparison of Grid Concepts

(a) STRAIGHT WAVE BEAM FULL GRID

(b) STRAIGHT WAVE BEAM STAGGERED GRID IN CHANNEL

(c) STRAIGHT WAVE BEAM FULL GRID IN SIMULATED FUEL ASSEMBLY

Figure 7-3. Diagrams Showing Straight Wave Beam Grid Patterns

Table 7-1
SUMMARY DESCRIPTION OF F9E TEST ASSEMBLY GEOMETRY

Parameter	Nominal Value
Fuel Assembly Outside Distance Across Flats (in.)	2.290
Gap Between Outer Channel i.d. and Inner o.d. (in.)	0.062
Wall Thickness of Outer Channel (in.)	0.040
Wall Thickness of Inner Channel (in.)	0.035
I.D. of Liner Across Flats (in.)	2.006
Pin-to-Liner Spacing (in.)	0.042
Pitch-to-Diameter Ratio (in.)	1.28
Cladding o.d. (in.)	0.250
Cladding i.d. (in.)	0.220
Number of Pins in Assembly	37
Length of Fuel Region (in.)	13.5
Length of Lower Blanket (in.)	2.5
Length of Upper Axial Blanket (in.)	1.0
Total Fuel Pin Length (in.)	39.594
Total Assembly Length (in.)	91.812
Axial Spacing of Full Grid Spacers (in.)	6
Axial Spacing of Staggered Grid Spacers (in.)	3
Total Spacer Beam Height (in.)	0.640
Effective Spacer Beam Height (in.)	0.350
Spacer Beam Thickness (in.)	0.012
Apparent Beam Thickness (in.)	0.027

EBR-II*, contact loads and stresses are comparable to those found under LMFBR and FFTF design conditions. EBR-II beam-to-channel and rod-to-channel differential growth at end-of-life are below the levels expected in the FFTF test assemblies.

F9E pin bowing induced grid stresses are considerably higher than in the FFTF test assemblies. This is due to the difference in fluence to burnup ratio in EBR-II and FFTF that results in less creep in the EBR-II environment. The E.O.L. value of grid stress shown exceeds the criteria allowable; the criteria allowable being exceeded at 10,700 hours. However, since the stress predictions for F9E are conservatively high, the implication of this behavior will be further assessed prior to shortening the design life of the experiment.

It is planned to update the prediction of the F9E grid assembly performance in the EBR-II as improved methods of analysis and especially as new irradiation induced swelling and creep correlations become available.

* Solution annealed Type 316 stainless steel swells at a higher rate than cold-worked Type 321 or 316; However, it has a lower creep rate. Therefore, the lower fluence/burnup ratio results in comparable swelling rates but reduced creep rates.

Table 7-2
COMPARISON OF LOADS, STRESSES AND DEFLECTIONS
EXPECTED IN F9E WITH FULL AND STAGGERED GRIDS IN FFTF

Structural Materials, SS	LMFBR Full Grid in FFTF	FFTF Staggered Grid 316 20% C.W.	F9E in EBR-II Full Grid			F9E in EBR-II Staggered		
			E.o.L. 316 Ann.	70% 316 Ann.	50% 316 Ann.	E.o.L. 316 Ann.	70% 316 Ann.	50% 316 Ann.
Beam-to-Channel Differential Growth (mils)	27	12	3	2	1	3	2	1
Pin Bowing								
Pin Bending Stress (psi)	15900 (36% ult)	11800	23300	15300	10900	23300 (70% ult)	15300	10900
Contact Load (lb)	0.48	0.35	3.01	1.88	1.18	3.01	1.88	1.18
Grid Stress (psi)	5500	6350	39890	24950	20000	39890	24950	20000
Pin-to-Grid Swelling								
ΔD (mils)	2.20	0.71	1.05	0.55	0.30	1.55	0.81	0.44
Contact Load, Elastic (lb)	2.31	0.23	1.02	0.53	0.29	0.67	0.35	0.19
Contact Load, Creep	0.23	0.07	0.52	0.34	0.21	0.34	0.22	0.09
Grid Stress, Elastic (psi)	26600	3840	12300	6450	3490	8240	4330	2340
Grid Stress, Creep	2660	1036	6240	4050	2510	4130	2680	1170
Pin-to-Channel Differential Growth								
Grid Bending Stress (psi)	15900	3840	5020	3230	1870	3360	2030	960

7.3 F9F DIAGNOSTIC GRID SPACER EXPERIMENT IN EBR-II

7.3.1 F9F Summary

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

The F9F assembly is designed primarily to demonstrate:

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

In the F9E design, the principal modeling variation was to substitute annealed steel for cold-worked steel to obtain high relative pin-to-grid swelling on a shortened time scale. In the design of F9F, the principal modeling variation is to increase the distance between the edge fuel pins and the channel to cool the channel and obtain high beam-to-channel and pin-to-channel relative growths. This will be the first grid test to experience appreciable amounts of these growths. The structural material used for the fuel pins, the grid spacers and the channel will be 20% cold-worked, Type-316 stainless steel. F9F will use the staggered grid concept exclusively in the core region; this is the reference design for the FFTF optimized grid spaced driver fuel assembly being developed by General Electric (described in Section 7.4). All of the fuel pins will be similar to the FFTF design. The target operating conditions will be similar to previous Group F9 series irradiations, i.e., maximum cladding outside temperatures in the range of 1050 to 1200°F and linear powers in the range of 14 to 16 kW/ft.

7.3.2 F9F Description

Current estimates of the F9F operating characteristics and test geometry are summarized in Table 7-3. These parameters are based on the assumption that EBR-II positions 5N2 or 5N4 (or equivalent) will be available for this test. The key parameter used for the selection of position is flux. Since the F9F assembly test results are to be used as input for the design of a grid assembly to be ready for irradiation in FFTF by June 1974, F9F irradiation time is limited. F9F is scheduled to begin irradiation in the first half of 1972. Therefore, the in-reactor time is limited to less than two years. In order to obtain the greatest possible fluence in this limited time, an EBR-II core position as close to the core center as possible is desirable. In the assumed position, F9F would receive a peak fluence > 0.1 MeV equal to 6.8×10^{22} n/cm², according to the data shown in the EBR-II Guide.

A peak linear power density of 14 to 16 kW/ft is proposed because it is high enough to be prototypical of FFTF and LMFBR designs, and low enough to provide reasonable peak temperatures at the low flow rates required to give large temperature differences between grid spacers and the channel.

These temperature differences are used to compensate for low EBR-II flux and a shorter-than-prototypical grid-beam length; this compensation provides substantially more grid-channel relative expansion than F9E, but less than in the FFTF environment.

Fuel pin-to-channel wall spacing is selected in F9F to provide the required temperature differences. In general, the temperature differences increase as edge spacing increases and flow rate decreases. Preliminary pin bowing calculations indicate that it is desirable to keep the maximum temperature differences acting across edge pins below ~200°F. The smallest edge spacing that yields large temperature differences between grids and the channel, (a temperature difference $\leq 200^\circ\text{F}$ across the outer rods) and keeps the maximum coolant temperature less than 1100°F is 75 mils. The required conditions are obtained when the flow rate of sodium through the assembly is about 20,000 lb/h. This flow rate yields a mixed mean coolant outlet temperature near 980°F which is within the limits imposed by EBR-II. Edge spacing and flow may be changed to reduce pin bowing after detailed calculations are performed.

Table 7-3
CONCEPTUAL DESIGN VALUES OF F9F GEOMETRY,
THERMAL CONDITIONS, FLUENCE, AND POWER

Parameter	Value
Outside Distance Across Flats of Outer Channel (in.)	2.290
Inside Distance Across Flats of Inner Channel (in.)	1.890
Wall Thickness of Inner Channel (in.)	0.120
Fuel Pin-to-Channel Spacing (in.)	0.075
Fuel Pin Diameter (in.)	0.230
Pitch-to-Diameter Ratio	1.263
Fuel Pin Cladding Thickness (in.)	0.015
Fuel Hot-Smeared Density (% TD)	85.5
Peak Linear Power Density (kW/ft)	14-16
EBR-II Core Position	5N2 or 5N4
Total Assembly Power (kW)	495-565
Total Assembly Flow (lb/h)	~20,000 (~50 gpm)
Mixed Mean Coolant Outlet Temperature (°F)	980
Maximum Coolant Outlet Temperature (Without Uncertainties) (°F)	1100
Maximum Bowing ΔT Across Corner Pin (°F)	160
Maximum Bowing ΔT Across Edge Pin (°F)	200
Maximum Grid-Channel ΔT (°F)	220
Peak Fluence > 0.1 MeV (n/cm ²)	6.8×10^{22}
Irradiation Time (Full Power Days)	365
Maximum Relative Grid-Channel Lateral Growth (mils Closure of Edge Spacing)	5
Peak Atom Percent Burnup in Lead Pin	10.3-11.8

An inner and outer hexagonal channel will be required in the F9F experiment in order to obtain an inner channel wall thickness prototypical of the FFTF channel, used to obtain the necessary rod-to-wall spacing, and to maintain the channel outside dimension required by EBR-II. This approach duplicates the F9E design and utilizes a Mark S EBR-II irradiation assembly, except that the dimensions of the inner channel are chosen to satisfy F9F objectives.

A primary objective of F9F is to investigate tab behavior under limiting in-reactor conditions. Specifically, the possibility of tab disengagement from its notch and/or tab failure due to lateral and axial loads will be investigated. With reference to Sketch A below, "lateral loads" are caused by grid-channel differential growth (in the plane of the paper), and "axial loads" are caused by relative rod-channel growth (out of the plane of the paper).

Therefore, both of these types of differential growth are to be produced in F9F. Since the axial force is transmitted to grids by pin-grid friction, the pin-grid contact forces will be designed to produce an axial force on each grid tab that at least equals the predicted force on FFTF tabs (~ 4.0 lb). Several tab heights may be incorporated to test the influence of tab behavior on the ability of a grid to move axially with the pins for various distances.

To test the effect on tab seating of tolerances in tab, notch, and channel dimensions, various degrees of tab-notch engagement may be incorporated. The nominal dimensions and tolerances will be such that the clearance for F9F will be the maximum tab-to-channel engagement of FFTF for all of the grids within the core except one. This one grid will be manufactured and installed such that an interference exists between the tab and the channel at installation. For the FFTF tab dimensions, each mil of interference will result in 0.1% maximum strain in the tab. Assuming all strain during installation to be equivalent to the plastic strain induced during the tab bending phase of manufacturing, with the exception of 0.2% strain which is recoverable, the relative beam-to-channel growth will result in a maximum strain of 0.7%.

The grid spacers at the top of the assembly will be dimensioned such that a nominal to minimum engagement exists as currently specified for FFTF. As no relative beam-to-channel swelling will occur, the tab engagement remains minimal throughout its life in the reactor. The relative pin-to-channel growth will be the maximum occurring in F9F. As no irradiation or thermally-induced creep is expected after the initial pin-to-grid preload due to pin-to-grid interference, high friction loads will occur through life. The bearing load at the F9F tab will be in excess of that of FFTF by more than 100%.

A backup method for grid-channel attachment may also be tested in F9F at a position above the core region. This concept does not depend on notches in the channel; instead, "stops" are attached to the channel and grids are allowed to float between these stops. Design and development of this concept is still in a preliminary stage.

Grid beams will be fabricated from 10 and 12 mil material. The reference beams are 10 mil. Grids may be fabricated with selected beams 12 mils thick, or entire grids may be made using 12 mil material. Grid beams 12 mils thick would be used to increase the pin-grid spring constant and tab stresses at critical locations. In addition, entire grids of these beams will be tested as a backup design alternative.

The design of the F9F grid beam differs from that of the F9E beam; a cutout has been introduced below the tabs as shown in Sketch B below:

The purpose of the cutout is to increase the effective tab length which allows easier tab bending. This modification is not expected to affect the behavior of the tab under axial loading.

7.4 FFTF GRID SPACED DRIVER FUEL ASSEMBLY

7.4.1 FFTF Driver Fuel Summary

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

Following a technical evaluation of several design alternatives, the modified straight wave beam staggered grid was selected as the reference design for the FFTF driver fuel assembly. However, a modified straight wave beam full grid

will be used at three axial locations in the assembly. A fuel assembly using full grids exclusively is the backup design to the reference FFTF grid assembly design. Figure 7-5 is a conceptual assembly drawing of the reference FFTF driver fuel assembly. The reference FFTF grid spacer concept offers potential advantages over the wire-wrap spacer in:

- increased accommodation of differential structural swelling
- reduced volume fraction of steel in the core
- more uniform temperature distribution around the circumference of the fuel pins
- reduced pressure drop through the bundle of fuel pins.

The reference modified straight wave beam, staggered and backup full grids, are basically a 217 pin version of the grids shown in Figures 7-2 and 7-3. Further analysis, and mechanical, hydraulic and irradiation tests will be conducted to support this reference grid spacer selection.

Discussion

The 217 pin assembly was designed using a standard FFTF driver fuel assembly channel and standard FFTF fuel pins. Figure 7-5 shows the axial positions of the grid spacers. The staggered grids have an axial spacing of 3 inches in the core region and 4-1/2 inches in the gas plenum region above the lower support pad. No grid spacers are located in the pad region of the channel resulting in a 10-inch unsupported length in this region. Full grid spacers are located above and below this pad region as well as at the top of the pin in order to minimize pin vibration in these locations. A 217 pin hydraulic flow test will be run in order to investigate the vibrational characteristics of this system (Section 7.6). An edge-pin-to-channel-wall spacing of 0.040 inch was selected in order to minimize the edge bypass flow and resulting thermal gradients. As a result of the reduced edge spacing and the fact that no pin-to-channel clearance is needed to accommodate differential swelling, the pitch-to-diameter ratio has been increased to 1.263 from the reference FFTF wire wrapped ratio of 1.24.

Table 7-4 summarizes the design and operating characteristics for the FFTF grid spaced and wire wrapped driver fuel assemblies. The reference and backup designs are basically similar to the modified straight wave beam, staggered and full grids used in the EBR-II diagnostic assemblies F9E and F9F. The coolant and cladding temperatures are calculated for both the grid and wire wrap assemblies on a consistent basis using GE LMFBR design methods resulting in values that differ somewhat from published FFTF predictions.

Table 7-4
REFERENCE CONCEPTUAL DESIGN BASIS OF FFTF
GRID SPACED AND WIRE WRAPPED DRIVER FUEL ASSEMBLY

Parameter	Grid	Wire
Axial Spacing		
Core (in.)	3	12
Plenum (in.)	4.5	12
Pad (in.)	10	12
Pitch-to-Diameter Ratio	1.26	1.24
Edge-Pin-to-Channel-Wall Spacing (in.)	0.040	0.056
BOL Peak Temperature Assembly Nominal Bulk		
Outlet Coolant Temperature (°F)	994	994
Maximum Coolant Outlet Temperature		
with Uncertainties (°F)	1103	1132
Maximum Cladding Inside Surface Temperature		
with Uncertainties (°F)	1166	1196
Spacer Thickness (in.)	0.010	0.056

Figure 7-5. FFTF Driver Fuel Assembly (Grid Spacer)

The grids are located within the channel by tabs which seat in notches within the channel wall. Protrusions caused by the notches are removed from the outside of the channel in the high swelling region of the core.

The fuel pins apply an axial load to the grid tabs since the pin axial growth exceeds channel axial growth. The loading applied to the grid by these pins results from pin-to-grid differential swelling, pin bowing, pin straightness tolerances and pin diameter tolerances. The tabs are further loaded as the grid beams swell faster than the channels because they operate at a higher temperature than the channel. The investigation of tab behavior is one of the objectives of the EBR-II diagnostic test F9F.

7.5 SODIUM FLOW TEST

7.5.1 Summary

A 6-month sodium flow test is underway in the General Electric Breeder Test Facility Control Drive Test Loop to evaluate the endurance characteristics of the reference FFTF Grid Spacer Design and to provide engineering information that will support the design basis for the F9E, F9F, and reference FFTF grid-spaced fuel assembly. The test assembly consists of two separate 37 pin dummy fuel bundles positioned in series. The FFTF shield/orifice region has been simulated for both assemblies to generate prototypical inlet turbulence in each bundle. One 37-pin assembly utilizes staggered grids, the reference for the GE grid-spaced FFTF driver assembly. The other 37-pin assembly utilizes full grids, the backup concept for the FFTF driver fuel assembly.

The test is scheduled for approximately 4000 hours of operation in sodium at 1000°F with a peak velocity of 22.3 ft/sec. The test assembly was installed in the drive test loop on September 23, 1971 and operation began on October 5, 1971.

This section describes the test bundles and operating conditions of the proposed test.

7.5.2 Test Objectives and Conditions

The prime experimental objectives of the 4000 hour sodium flow test are to:

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

Secondary objectives are to determine:

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

A summary of the test parameters and conditions for the two 37-pin sodium flow bundles are shown in Table 7-5. This test will operate at these conditions for approximately 4000 hours.

The loop temperature is to be maintained at 1000 +10/-20°F as recorded at the inlet of the test vessel. High and low temperature alarm points are set at 1020°F and 980°F respectively. Oxygen impurity level in the sodium is to be maintained at a level appropriate with a plugging temperature of 300°F or less and will be monitored every 340±50 hours. Impurity control is provided by a by-pass cold trap system and is confirmed by a plugging indicator.

Table 7-5
SUMMARY OF SODIUM FLOW TEST PARAMETERS

Environment	
Coolant	Sodium
Temperature	1000°F
Purity (Plugging Temperature)	<300°F
Flow Rate	110 gpm as-built
Velocity (maximum)	22.3 ft/sec
Duration	~4000 h
Dummy Pins	
Number	37
Outside Diameter	0.250 in.
Wall Thickness	0.015 in.
P/D	1.28
Edge-to-Channel Clearance	0.045 as-built
Material (19 Pins)	Type 316, Ann.
(18 Pins)	Type 316, C.W.
Grids - Bundle 1	
Type	Full
Number	5
Pitch	6 in.
Material	Type 316 Ann.
Grids - Bundle 2	
Type	Staggered
Number	10
Pitch	3 in.
Material	Type 316, Ann.

The as-built main loop flow is maintained at 120 ± 15 gpm. This will provide a flow of 110 gpm through the experimental bundles and approximately 10 gpm through a drain orifice in the entrance of the loop assembly. This flow rate is monitored by a magnetic flowmeter in the main loop piping and results in a peak velocity of 22.3 ft/sec within the test bundles.

Loop cover gas pressure is maintained in the range of 6 to 30 psig consistent with the need to maintain adequate pump suction pressure.

During operation, the primary data to be taken include the main loop flow rate, sodium temperature, and sodium plugging temperature. Main loop flow rate and temperature are continuously recorded throughout the test along with other secondary loop data. Complete operational loop data are recorded twice daily as required. Sodium plugging temperature will be monitored every 340 hours as explained above.

7.5.3 Description of Test Section

The test assembly inserted into the Control Drive Test Loop is shown in Figure 7-6. This assembly consists of the bundles, numerous internal housings, and a spring holddown tower located at the top of the assembly. With this arrangement, the 37-pin dummy fuel assemblies are positioned in series and will thus see identical flow conditions. Since the loop is isothermal, operating temperatures for the two sections will be identical. Photographs of the lower and upper portions of the test assembly are shown in Figures 7-7 and 7-8, respectively.

(G1)

NOTES:

1. THESE PARTS (P8-P21, P32-P35) SUP'D AS AN ASM BY ENGR.
2. NITRIDE PER G.E. SPEC. FA 231909 TO A MIN. DEPTH OF .0015 TO .0050 + HARDNESS OF R_C 30-15N.
3. TORQUE 10-50 FT-LBS MIN.
4. SAFETY-WIRE PER 209A4287.

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Figure 7-6. Sodium Flow Test Assembly

Figure 7-7. Lower Section of Test Assembly

Figure 7-8. Upper Section of Test Assembly

Mechanically, the dummy fuel elements are identical except for grid spacer design. The top element utilizes 10 staggered grids, the reference for the GE grid-spaced FFTF driver assembly, and the bottom element 5 full grids, the backup concept.

Details of the dummy fuel element design are shown in Figure 7-9. These elements consist of six main components:

1. Seal Flange
2. Channel
3. Grid Spacers
4. Dummy Fuel Pins
5. Pin Support Assembly
6. Element Nose Piece

Figure 7-10 is a photograph of the upper full grid fuel element. Except for the grid spacer design variations, all components are identical in each bundle.

The seal flange provides the main structural support for the dummy fuel element. It is also an integral part of the test assembly and contains a pin holddown device. It is constructed of Type 304 stainless steel and is machined to a fine finish to provide a good sealing surface between sections of the assembly.

The channels are designed similar to the F9E inner channel. They have an inside across flats dimension of 2.006 with a 0.063 wall thickness. For enhanced high temperature operation, Type-316 stainless steel has been used. Pocketing was provided to accept the specific grid spacer design using F9E production tooling.

Two basic grid spacer types have been used—the staggered grid spacer at 3-inch spacing and the full grid spacer at 6-inch spacing. These spacer types have been used in the top and bottom bundles respectively and are shown in Figures 7-11 and 7-12. These grid spacers are identical to those used in the F9E in-reactor bundle for EBR-II.

In order to provide a maximum utilization of the test, two different grid beam-to-channel clearances have been incorporated to provide Engineering information on the grid/channel interface.

As a final test variable, one grid in each bundle has been constructed without locating tabs. This grid is located at the top position in each bundle and is held in place by pin-to-grid friction only. A precise measurement of its location has been recorded, which will be compared with its location at the completion of the test.

The dummy fuel pins are similar in design to the F9E fuel pins. The top end plugs are identical to the F9E in-reactor design and incorporate both removable and non-removable design features. The cladding is Type-316 stainless steel 0.250 inch in diameter with a 0.015-inch wall thickness. About half the cladding in each bundle will be annealed and half cold worked. The bottom endplugs incorporate removable and non-removable features compatible with the pin support assembly. The fuel pellets have been replaced by 0.216-inch outside surface carbon steel pellets one-half inch long to simulate the fuel mass.

All pins have been identified and their location in the bundle recorded.

The nose piece design is a compromise between the requirements of low pressure drop and a desire to simulate the inlet turbulence of the FFTF bundle. As such, the final design consists of four shield blocks sequentially spaced at 90-degree increments. An inlet nozzle is also provided on each bundle.

A support spider on the element nose piece provides radial support at the lower end of the element which is compatible with a mating part within the test assembly.

Figure 7-9. Sodium Flow Test Bundle

Figure 7-10. Upper Full Grid Fuel Element

Pressure drop calculations have been performed for the 110 gpm flow rate and are as follows:

	Full Grid	Staggered Grid
Bare Bundle ($\ell = 33$ inch)	9.1	9.1
Grid Spacers	9.6	6.6
Entrance Loss	1.0	1.0
Exit Loss	2.0	2.0
Nose Piece	28.1	28.1
Total	49.8 psi	46.8 psi

The bundle components have been designed for 75 psi at 1000°F. The internal housings which support the bundles within the test loop have been designed for 150 psi at 1000°F.

7.6 HYDRAULIC FLOW TESTS

7.6.1 Hydraulic Flow Test Background

Flow induced vibration is of concern with any fuel bundle designed for extended periods of operation. In order to determine the flow induced vibration response in the reference FFTF fuel bundles, a two-point test program is in progress:

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

APPLIED PRACTICES	VERIFIED	TESTED	APPROVED	DATE
209 A4270	✓	—	—	—

GRID SPACER (STAGGERED)

NO.	NAME	DESCRIPTION
2 1 1	BEAM	145C6607 P1
2 1 2		P2
2 1 3		P3
2 1 4		P4
1 2 5		P5
1 2 6		P6
1 2 7		P7
1 2 8	BEAM	145C6607 P8

FCF - 188F 475 (G1 & G2)
816E 140 (G1, G2 & G3)
FOR DESIGN DATA SEE 325 X 157
325 X 158

ALL WELDS TO BE
MADE SIMULTANEOUSLY
(12 TOTAL)

030 WELD SET DOWN

GI G2 G3

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3	1/11/81 1000	1/11/81 1000
TITLE WAS, MODIFIED SOURCE HAVE BEAM - STAGGERED GRID		TITLE WAS, GRID SPACED ALTERNATE. PL. REVISED NAME COULMAN WAS, "BEAM"
4	1/11/81 1000	1/11/81 1000
ADDED STAGGERED PL. PL - PL DELETED STAGGERED FROM SA SK		ADDED STAGGERED PL. PL - PL DELETED +200 WMS ZERO +200 WMS ZERO

145C6774

Figure 7-11. Grid Spacer (Staggered)

7.6.2 Single Pin Tests

The single-pin flow tests initiated during the latter part of this quarter will provide preliminary measurements of vibrational amplitudes and frequencies as a function of flow velocity and identify the basic vibrational mode shape for pins supported by staggered grids. This will permit vibration sensors to be located at points of maximum displacement in the following tests. They will also provide an opportunity to use and check out the accelerometers required to measure vibrations of interior pins of the full bundle tests.

The single-pin test section consists of a cylindrical tube of plexiglass inside which the pin is located. The inside diameter of the plexiglass tube is 0.4 inch, matching the hydraulic diameter in the prototypical FFTF bundle. Pairs of grid gams, oriented at 120-degrees to each other, are placed at 3-inch intervals along the tube to duplicate beam supports in a full bundle. The pin is supported on the beams and fixed by screw supports on one end. A cylindrical accelerometer and copper pellets (to mockup the fuel mass) are mounted inside the pin. The pin can be adjusted axially to move the accelerometer to any location from one grid to another. In addition, the accelerometer can be rotated to determine if there is a preferred angular orientation for vibration. Flow through the test section is provided by a small loop with a recirculation pump and surge tank. Flow rates up to 10 gpm can be achieved, corresponding to test section velocities in excess of the maximum FFTF velocity.

7.6.3 217 Pin Flow Tests

The principal flow tests are with the 217 pin bundle, modeling at full scale the FFTF geometry. These tests will be run in the BRD hydraulic test facility at 500 gpm. This will produce a flow velocity of 22.3 ft/sec matching the maximum FFTF flow velocity.

A summary of the FFTF bundle geometry to be modeled is given below:

Number of pins	217
Rod diameter	0.230 inch
Wall thickness	0.015 inch
Pin pitch	0.2905 inch
P/D ratio	1.2630
Channel edge spacing	0.040 inch
Channel across-flats dimension (inside)	4.335 inch
Channel wall thickness	0.120 inch
Grid beam thickness	0.010 inch

The spacers in the bundle will be wave-beam grid spacers similar to the FFTF design shown in Section 7.4. Both full and staggered grids will be used in the upper and lower halves of the bundle respectively. By accommodating two grid designs in one bundle, comparison between the two can be made. A schematic of the test bundle is shown on Figure 7-13. A removable inlet region has been provided to investigate the effects of various inlet turbulence levels.

Vibrations of pins at the bundle periphery will be measured with proximity transducers mounted on the hexagonal channel wall; interior pin vibrations will be measured with accelerometers identical to that used in the single-pin tests. The locations of the sensors are summarized in Table 7-6.

A "traveling probe" concept will be used on two interior pins. An extra length pin will be loaded approximately 18 inches with copper pellets, the accelerometer inserted, and annular pellets threaded on the lead wires for the remaining bundle length. The pin will extend out through the top flange of the test section thus providing a means of positioning the accelerometer. During flow testing, the accelerometer will be positioned at various axial locations along the bundle and also rotated 360-degrees to provide a complete displacement profile.

The results will include the mode shapes, pin displacements and vibrational frequencies for FFTF fuel pins supported by full and staggered wave-beam grids. The vibrational characteristics will be presented as functions of flow velocity for each pin type: interior, edge and corner. The data will be sufficient to permit calculation of pin bending stresses and grid loads for each pin-grid combination. In addition, the entrance region and free end effects will be investigated.

Table 7-6
LOCATION OF VIBRATION SENSORS

Axial Location	Elevation (in.)	Pin Type Interior^(a)	Edge^(a)	Corner^(a)
Entrance Region Between Grids 1 and 2	~10.6	F	F	P
Mid-Staggered Grids	~22.6	M,M	P,F	P
Mid 10 in. Span	47.145	M,M	P	
Mid Full Grids	61.145	M,M	P	F
Free End	77.870		P	

(a) Sensor Types: F - Fixed Accelerometer
M - Movable Accelerometer
P - Proximity Transducer

Figure 7-13. FFTF Water Flow Test Assembly

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816E168
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CONTROL

8. TASK G-10—SAFETY METHODS AND ANALYSES

Task Leader: D. L. Brown*

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

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

A failure threshold analysis for the Task C-1 C4B⁸⁻¹ experiment has been performed with the coupled TOFF and THTE computer programs.

Test Parameters

Fuel column length (inches)	14.2
Pellet density (% of theoretical)	94.8
Fuel to cladding gap (diametral, inches)	0.004
Fuel pin outer diameter (inches)	0.25
Fuel pin cladding thickness (inches)	0.015
Effective gas plenum length (inches)	12.046
Initial sodium temperature (°F)	800
Cladding failure strain (%)	4

Sodium temperatures, power transient, and estimated failure time are shown in Figure 8-1. The constructed TOFF model corresponds to the THTE model, and consists of eighteen radial fuel nodes, three cladding nodes, and four coolant nodes. The analysis is one dimensional because the fuel is fresh and the TREAT axial power distribution is flat over the capsule fuel length.

Figure 8-2 shows the cladding strain and molten fuel fraction as a function of time. Figure 8-3 shows the molten fuel pressure increase as a function of time. Failure time was reasonably well predicted when compared to the estimated failure time indicated by thermocouple failure. From the thermocouple data, the estimated failure strain range is 3.05 to 4.06%. Post irradiation measurements adjacent to the failure point indicated failure at 4% strain.

8.2.2 Loop Hydraulics and Channel Voiding

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

The VOIDUM loop hydraulics routines and the BUBBLE sodium boiling routines have been combined to provide a free-standing analysis capability for primary system thermal hydraulic transients for loop and tank type systems.

The revised VOIDUM-BUBBLE program is being expanded for pipe-break analysis. When completed, the routine will simulate breaks at piping connecting pump-to-core, core-to-IHX and IHX-to-pump. Time-dependent tabular input for pump pressure rise and for pump characteristic functions are also incorporated.

* Contributors are listed in Appendix B.

Figure 8-1. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4B Transient Irradiation Results (Test Transient No. 1057)—Reference 8-1

Figure 8-2. Fuel Melt Fraction and Cladding Strain versus Transient Time

Figure 8-3. Fuel Pin Interior Pressure History as Predicted by the G-10 Overpower Failure Model TOFF

8.2.3 Probabilistic Modeling

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

A preliminary study of core surveillance instrumentation designed for limiting a postulated fuel failure propagation accident was completed. For a plant protection system of the 1000 MWe LMFBR range, it was assumed that three scram systems would be required to cope with a fuel failure propagation accident. For each system, a limiting probability of failure to scram the reactor was assumed irrespective of the number of sensors providing signals. These systems are:

	failure probability
neutron flux	10^{-6}
bundle instrumentation	10^{-3}
noise	10^{-3}

The probability of a failure to scram after an event capable of activating all systems is thus 10^{-12} .

The possible end states of a failure propagation accident are shown in Table 8-1. The associated costs permit the economic consequences to be evaluated.

Results of a simplified probability calculation are shown in Table 8-2 for an accident progression beginning with local boiling. Boiling, fuel meltdown and subsequent propagation to adjacent assemblies by FCI (fuel-coolant interaction) is the assumed accident progression mode. The results indicate that the frequency of an excursion is very low and not significantly different for the full PPS (plant protection system) response or the flux-only response. This is due to the assumed slow (with respect to PPS response time) accident progression.

The principal effect of the added instrument signals in the PPS (noise and bundle signals) is in reducing the frequency of experiencing bundle failure propagation. As indicated in Table 8-2 the result is to reduce the mean-annual-cost of failure propagation accidents from \$220,000 per year to \$135,000 per year. The mean-annual-savings of \$85,000 per year is achieved at the expense of a capital investment of approximately 10 million dollars plus an additional operating expense of an undetermined amount.

Referring to Table 8-2 suggests that the dominant cost occurs for shutdown to remove a single bundle (S-3 and S-6 in Table 8-1) regardless of PPS action.

8.2.4 Core and Reactivity Modeling

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

Improvements in the treatment of Doppler feedback with sodium voiding in the FREADM code were completed. Previously, the Doppler coefficients for all axial sections of a bundle type were changed to the sodium-out values at the point in time of void initiation. All subsequent Doppler feedback reactivities for each axial section were then calculated based on the sodium-out Doppler coefficient for that axial section whether or not that axial section was voided. This gave rise to spurious step reactivity insertions.

The modifications will allow a more detailed analysis of the change in the Doppler reactivity with core voiding. The code follows the void pattern with time and permits the Doppler coefficient to be adjusted as the bundle voids. The fraction of voiding for each axial section of each bundle type is calculated and is used to update the Doppler coefficient for each kinetic time step after void initiation. This is accomplished in the following manner:

$$A' = A[f_1 f_2 + (1 - f_2)]$$

Table 8-1
END STATES AND COSTS PER EVENT

1000 MWe LMFBR			
End State	Description	Downtime (Days)	Cost ^(a) (10 ³ \$)
S-1	Many crack failures		
S-2	Many burst failures		
S-3	Replacement, single sound bundle	2.6	150
S-4	Replacement, a few (5) sound bundles	2.7	180
S-5	Replacement, many (12) sound bundles	2.8	230
S-6	Replacement, a single damaged bundle	4	220
S-7	Replacement, a few (5) damaged bundles	10	570
S-8	Replacement, many (12) damaged bundles	34	1880
S-9	Replacement, many (12) control rods	14 extra days	1000
S-10	Cleanup, sodium system	14 extra days	750
S-11	Replacement, core, head and core container	365	30000

(a) Based on yearly average cost of replacement power of 2.2 mils/kW-h.

where

- A' is the current Doppler coefficient for a given bundle axial section
- A is the input sodium-in Doppler coefficient for the axial section
- f₁ is the input ratio of sodium-out to sodium-in Doppler for the core
- f₂ is the current fraction of the axial section which is voided

These modifications were accomplished without any new input being required.

The effect of this change on the analysis of hypothetical accidents was computed for both a loss of flow and a reactivity insertion accident in a 350 MWe LMFBR. The transient power with the old and the new Doppler treatment is shown in Figure 8-4 for the loss of flow case and in Figure 8-5 for the reactivity insertion case. Calculations were carried out to the point of disassembly for each case as determined by the energy density of the hottest fuel. The power level at the time of disassembly was significantly reduced by the new Doppler treatment for both types of accident. Although disassembly calculations were not performed, previous work⁸⁻² indicates that energy release is proportional to initiating power level for cases in which the disassembly phase is initiated from a transient mode.

8.2.5 FREADM Update

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

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

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

A linked version of the VENUS code was prepared and run on the GE-635 computer. This version of the code will be compatible with core storage requirements of FREADM. Interface routines to provide direct coupling of FREADM and VENUS are being prepared. Similar routines to couple TOFF and FREADM are in progress.

Table 8-2

EFFECT OF PPS ON MEAN-ANNUAL-COST

Item	End State	Assumed Probability of Propagation	Frequency Per Year No PPS Action	Cost (10 ³ \$) No PPS Action	Frequency Per Year $\phi(a)$ Only	Cost (10 ³ \$) $\phi(a)$ Only	Frequency Per Year (full PPS)	Cost (10 ³ \$) (Full PPS)
Prompt critical	S-11,10,9	1	5×10^{-4}	15.9	5×10^{-10}		5×10^{-16}	
Subprompt critical	S-10,9,8	0.1	5×10^{-3}	16.3	5×10^{-9}		5×10^{-15}	
Many melted bundles	S-8	0.5	10^{-2}	9.4	10^{-6}		10^{-13}	
Many boiling bundles	S-5	1	10^{-2}		10^{-6}		10^{-13}	
Many voided bundles	S-5	1	10^{-2}		10^{-6}		10^{-11}	
Few melted bundles	S-7	0.1	10^{-1}	51.3	10^{-3}	0.6	10^{-9}	
Few boiling bundles	S-4	1	10^{-1}		10^{-3}		9×10^{-8}	
Few voided bundles	S-4	1	10^{-1}		10^{-2}	1.8	9×10^{-6}	0.001
Bundle meltdown	S-6	0.1	1	198	1	218	9×10^{-4}	0.2
Bundle boiling	S-3	1	1		1		0.9	135
Local boiling	-	1	1		1		1	
Accident Progression								
Total Cost				291		220		135

(a) ϕ = neutron flux

Figure 8-4. Fuel Average Power Density versus Elapsed Time for Loss-of-Flow Accident

Figure 8-5. Fuel Average Power Density versus Elapsed Time for Reactivity Insertion Accident

8.2.6 Parametric Analysis

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

8.2.6.1 Exploratory Hydrodynamics Analysis

An exploratory analysis using the PISCES IDL computer code is in progress. This code is one of a family of codes developed by Physics International to obtain solutions to the general hydrodynamics problems including stress wave generation and propagation. The code is being applied to problems associated with core disassembly analysis. Initial applications are primarily for investigating code operation and the cost of problem solving. The major input information required for the code is the equation of state of the material involved.

Equation development has continued throughout the quarter. The approach has been to employ the equation of state model developed by DASA.⁸⁻³ This model requires as input the parameters presented in Table 8-3. The program has been used to compute pressures both for UO_2 (as an approximation for mixed oxide), and for sodium, as a function of energy and specific volume. The programmed equations cover the full range of material forms: solid, solid-liquid, liquid, liquid-vapor and vapor.

Table 8-4 presents some of the specific values which have been used to model sodium. It should be noted that a molecular weight of 26.5 has been used for sodium for purposes of comparing with computations made by Hirshfelder's method.⁸⁻⁴ This has necessitated modifying some of the mole-specific data used which were based on the normal molecular weight of 23. Figure 8-6 presents the pressure versus density for the three sets of input listed in Table 8-4. Cases I and III use densities modified by the ratio of 26.5/23 times the densities in Case II. Case III has modified enthalpies at the melt temperature.

The effort is continuing to determine the most appropriate set of parameters. A more critical review of the parameters related to the solid state is being made because predicted liquid pressures are computed in the equation-of-state program based on an extension of solid state properties. Accurate modeling of this compressed liquid may be important for predicting the effects of fuel-coolant interaction.

A set of input parameters has been developed for UO_2 also. Table 8-5 and Figure 8-7 present the values input and the computed pressures for the saturation curve. The code was not intended for this application but uses such a general formulation that acceptable results may be obtained by arbitrary adjustment of the input parameters. In the cases presented the parameter G was varied to bring the results closer to the data derived by Hirshfelder's method.

G is determined by,

$$G = K\alpha\rho_0/C_v\rho$$

K = Bulk modulus

α = Coefficient of volumetric expansion

C_v = Specific heat at constant volume

Evaluating the above expression at a temperature of 3080°C gives 0.530 as a value of G; a value near 0.4 produces a better agreement with data derived by using Hirshfelder's method.

8.2.6.2 Transient Bundle Thermal-Hydraulics

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

Table 8-3.
INPUT PARAMETERS FOR EQUATION OF STATE³

M	: Atomic Weight (gm/mole)
ρ_o	: Normal Density (gm/cm ³)
T_c	: Critical Temperature (°K)
c	: Coefficients in Liquid Density Equation
d	$\rho_l/\rho_c = 1 + c(1-t)^{1/3} + d(1-t) \quad t = T/T_c$
K_o	: Parameter in Vapor EOS
	$a(t) = K_o + K_1 t$
	$p + a(t) \rho^2 + a'(t) \rho^3 = \frac{\rho t / Z_c}{1 - b \rho + b' \rho^2}$
	$k_1, a'(t), b, \text{ and } b'$ are all determined by the code
T_M	: Melting Temperature (°K)
$H_S(T_M)$	: Solid Enthalpy at T_M (kcal/mole)
$H_L(T_M)$	: Liquid Enthalpy at T_M (kcal/mole)
$H_v(T_M)$	: Vapor Enthalpy at T_M (kcal/mole)
$\rho_S(T_M)$	: Solid Density at T_M (gm/cm ³)
$\rho_L(T_M)$	: Liquid Density at T_M (gm/cm ³)
T_B	: Boiling Temperature (°K)
$H_L(T_B)$	: Liquid Enthalpy at T_B (kcal/mole)
$H_v(T_B)$	: Vapor Enthalpy at T_B (kcal/mole)
C	
D	: Coefficients in Solid EOS (kbar)
S	
	$P(\rho, E) = C\mu + D\mu^2 + S\mu^3 + G\eta E$
	$\mu = \eta - 1$
	$\eta = \rho/\rho_o$
G	: Coefficient in Solid EOS [kbar/(cal/g)]

Table 8-4
PARAMETERS USED FOR MODELING SODIUM

Variable	Unit	Case I	Value Case II	Case III
M	gm/mole	26.5	26.5	26.5
ρ_l	gm/cc	1.115	0.9684	1.115
T_c	$^{\circ}\text{K}$	2803	2803	2803
c		1.8	1.8	1.8
d		2.8	2.8	2.8
K_o		5.5	5.5	5.5
T_M	$^{\circ}\text{K}$	370.82	370.82	370.82
$H_S(T_M)$	kcal/mole	2.2365	2.2365	2.370
$H_l(T_M)$	kcal/mole	2.995	2.995	3.02
$H_v(T_M)$	kcal/mole	29.28	29.28	29.85
$\rho_S(T_M)$	gm/cc	1.0985	0.9514	1.0985
$\rho_l(T_M)$	gm/cc	1.0675	0.9275	1.0675
T_B	$^{\circ}\text{K}$	1154.4	1154.4	1154.4
$H_l(T_B)$	kcal/mole	9.95	9.95	9.95
$H_v(T_B)$	kcal/mole	33.95	33.95	33.95
C	kbar	68.038	68.038	68.038
D	kbar	45.150	45.150	45.150
S	kbar	28.30	28.30	28.30
G	kbar/cal/g	0.5604	0.5604	0.5604

Table 8-5
INPUT VALUES FOR UO_2

Variable	Unit	Value	Variable	Unit	Value
M	gm/mole	270	$\rho_S(T_M)$	gm/cc	9.67
ρ_l	gm/cc	10.96	$\rho_l(T_M)$	gm/cc	8.74
T_c	$^{\circ}\text{K}$	6723	T_B	$^{\circ}\text{K}$	3574
c		1.68	$H_l(T_B)$	kcal/mole	103
d		1.50	$H_v(T_B)$	kcal/mole	225
K_o		8.1	C	kbar	1957
T_M	$^{\circ}\text{K}$	3080	D	kbar	18040
$H_S(T_M)$	kcal/mole	71.1	S	kbar	65890
$H_l(T_M)$	kcal/mole	89.1	G	kbar/(cal/g)	Variable See Figure 8-7
$H_v(T_M)$	kcal/mole	218

Figure 8-6. Sodium Equation of State

Figure 8-7. UO_2 Equation of State

A model has been constructed for the THTE program representing a region near the channel wall in a 127-pin bundle. The region considered in the analysis is the first 3-1/2 rows of pins. The effects of circumferential conduction in the fuel and cladding are included as well as interchannel mixing by the eddy diffusivity model of Rogers and Todreas⁸⁻⁵ (with enhanced coefficients to represent wire-wrap effects).

The model consists of 10-axial planes of nodes composed of ~ 100 nodes per plane. The flow distribution has been determined assuming no radial pressure gradients exist at any plane normal to the axis.

Transient computations have been performed for two flow coastdown rates. The flow coastdown rates and resulting temperature distributions near the core exit are given in Figures 8-8 and 8-9. Both flow coastdown rates result in a significant temperature difference between the channel wall and sodium in the interior of the channel at the time boiling initiation would be expected. The depressed temperature region encompasses roughly the first row of pins from the channel wall or about 30% of the fuel for a 127-pin bundle. The effect of this depressed temperature region will be to provide a heat sink for radial condensation which will delay the initial voiding process.

The results of this analysis will be used in the formulation of a radial condensation model and to provide a basis for relating few pin tests to actual reactor conditions.

8.2.7 Experimental Guidance and Analysis

Detailed studies were initiated to provide a more explicit coordination of analytical safety effort and experimental plans. Efforts to resolve important design safety and licensing questions rely strongly on supportive information furnished by well planned testing of key analytical uncertainties. The consequences of large, hypothetical accidents are especially uncertain due to lack of understanding of the characteristic energetics, pressure-generation and fuel motion effects associated with various accident progressions. Preliminary recommendations have concerned the testing of fuel motion effects and activities to model the observed phenomena.

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- 8-4 Hirshfelder, et al., *Ind. Eng. Cham.* 50, 375, 386, (1958).
- 8-5 Rogers and Todreas, "Coolant Interchannel Mixing in Reactor Fuel Rod Bundles with Single-Phase Coolants," *Heat Transfer in Rod Bundles*, ASME (1968).

Figure 8-8. Nominal Flow Coastdown Radial Temperature Distribution at 28.5 Inches Above the Active Core Bottom

Figure 8-9. Rapid Flow Coastdown Radial Temperature Distribution at 28.5 Inches Above the Active Core Bottom

9. TASK G12—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 in safety and related areas.

9.2 LIAISON

An intensive bilateral flow of information between ANL and GE-BRD has resulted from the newly established office of a GE-BRD representative at Argonne National Laboratory, operated under G-12. The resident engineer prepares a weekly letter which is distributed to interested parties at General Electric BRD, Sunnyvale, California. No general dissemination of these informal reports is made, due to the preliminary nature of the information. This digest is formulated to call attention to some ANL activity, rather than to present detailed analyses. As a consequence, GE-BRD personnel make requests for specific technical literature available from ANL, frequently with reference to a previous weekly letter. The method of communication via a weekly report of this type has been chosen to ensure flexibility in internal distribution and to limit the amount of correspondence.

Frequent phone conversations supplement and speed up the exchange of information, particularly because the resident engineer at ANL is kept well informed about the activities of the GE-BRD staff. Beyond that, the GE representative visited his home base at Sunnyvale, in August to acquaint himself with the ongoing projects and to meet with individuals who need information from Argonne. His presentation of a seminar on "Current Development Activities at Argonne National Laboratory," was well received by a large attendance. The discussions have centered around the LMFBR safety program being carried out by the ANL Reactor Analysis and Safety Division, because it is more closely related to the PA-10 work. Nevertheless, quite a few requests were received for followups in other divisions of the laboratory, such as MSD, EBR-II Project and Chemical Engineering. Upon return to ANL the GE-engineer was able to fill most of the requests and has also obtained computer code material for use at BRD, such as BOW V, FCI parametric model.

On the other hand the GE projects of interest to ANL staff were PA-10, SEFOR, and Core Design. These liaison activities required either that some information be gathered for an ANL staff member or that the latter be put in contact with the cognizant GE task leader for a technical discussion. Also, recent preliminary test results and analyses from PA-10 activities and GEAP reports have been distributed to interested ANL personnel who expressed a need for this data. Finally the number of quarterly PA-10 reports distributed to the RAS division has been increased in an attempt to let more people gain access to it in a shorter time.

* 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 high performance of core structural members and the fuel system, especially the survival of the cladding at the temperatures and specific fuel pin powers of practical interest.

The objectives of the cladding materials development program are:

1. To 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. To investigate the rate of alloy swelling and the environmental and metallurgical factors which influence and control it in LMFBR primary alloys and those which are potential candidates.

10.2 MATERIALS IRRADIATION PERFORMANCE

The objective of this subtask is to document the irradiation history and to evaluate the resulting changes in mechanical and physical properties of the candidate metals for cladding.

A summary of this effort is shown in Table 1-4.

10.2.1 Series L-6 Materials Capsules

The original purpose of the L-6 experiment was to compare irradiation damage of Types-304, -316, -321, -347 stainless steel and Incoloy-800 to fluences of 2×10^{23} n/cm²-GE.

The three L-6 capsules (Mark-B7 type) accumulated a total fluence of 8.3×10^{22} n/cm²-GE in S/A X018B, and have been discharged from the EBR-II. These capsules contained tubular and precut mechanical property test specimens including Types-304, -316, -321, and -347 stainless steels and samples of Incoloy-800.

Nondestructive examination of these three capsules was completed including gross gamma scanning, profilometry, and visual examination of the capsules. Destructive examination of the capsules was initiated with capsule disassembly by sectioning and removal of the sample inserts. General overheating of the tubular specimens which were located in the core region during irradiation was apparent from visual observations. Several of these specimens were discolored and severely bowed as a result of restricted thermal expansion. The three tubular sections in the midcore region of capsule B were self welded together, suggesting that they were operating at temperatures above 760°C (1400°F).

10.2.2 Series L-15 and MI-4—Materials Capsules

The purpose of these experiments is to evaluate the effect of irradiation on the mechanical properties of various austenitic stainless steels and high-nickel alloys.

The four L-15 capsules and the MT-4 capsule (Mark-A19 type) were irradiated in S/A X080. This subassembly was discharged and examined in the FEF at EBR-II. The individual capsules were nondestructively examined by visual examination, neutron radiography, and profilometry. Aside from small increases in the outside diameter of the capsules (0.12% $\Delta D/D$ for the L-15 series and 0.32% for the MT-4 capsule), these capsules appear to be in satisfactory condition for further irradiation. The capsules are scheduled for reconstitution into S/A X 148 for continued irradiation. The total fluences accumulated to date are 2.3×10^{22} n/cm²-GE and 7.8×10^{22} n/cm²-GE respectively.

* Contributors are listed in Appendix B.

10.2.3 Series L-16 Materials Capsules

A group of materials capsules, designated the L-16 series, is being designed to provide quantitative data on the swelling behavior of candidate clad alloys including Types-304, -316, and -321 stainless steel as a function of temperature of irradiation, metallurgical condition, and fluence.

10.2.3.1 L-16A Capsules

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

10.2.3.2 L-16B Capsules

The 19 L-16B capsules are in the EBR-II core, row 4 (S/A X098). The irradiation exposure for this S/A is scheduled to be a total fluence of 3×10^{22} n/cm²-GE. The total fluence accumulated to date is 1.9×10^{22} n/cm²-GE.

10.2.3.3 L-16D Capsules

These three capsules were designed to accommodate highly irradiated ring specimens and unirradiated controls for continued irradiation to higher fluences. Selected cladding specimens are being taken from the high-burnup Task F fuel pins and Task L materials capsules. In addition, samples have been obtained from tubing after exposure to high temperatures flowing sodium. Fabrication of the capsule insert parts is continuing.

10.3 CLAD ALLOY SWELLING

The objective of this program is to measure clad alloy material and S/A hardware volume changes in available irradiated specimens to determine how swelling is affected by 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 Density Measurements of Fuel Rod Cladding Samples

Immersion density measurements have been completed on a total of 18 sections from seven mixed oxide experimental fuel pins irradiated in EBR-II. These data are listed in Table 10-1. Selected samples will be further examined using transmission electron microscopy (TEM).

In this quarter, the density measurements from F5Q cladding samples of 50% cold worked Type-316 stainless steel have been completed. They show that this cladding had essentially zero change in volume. This encapsulated pin was irradiated with midwall cladding temperatures of up to 625°C (1157°F) to a peak fluence of 3.4×10^{22} n/cm², $E > 0.1$ MeV which currently is the highest exposure attained by a cold-worked Type-316 stainless steel clad fuel pin in the United States. The results of zero clad swelling are in agreement with the data reported by HEDL¹⁰⁻¹ on a prototypical FFTF 20% cold-worked Type-316 stainless steel clad pin irradiated to a peak fluence of 3.0×10^{22} n/cm², $E > 0.1$ MeV; however, the results are at variance with United Kingdom data¹⁰⁻² from 20% cold-worked Type-316 cladding irradiated in DFR which showed appreciable swelling after similar exposures.

The F5Q data also support the HEDL results¹⁰⁻³ from 27 and 75% cold-worked Type-316 (swaged) materials specimens which show no volume increase after exposures of approximately 5×10^{22} n/cm², $E > 0.1$ MeV.

The solution annealed Type-316 data show some apparent anomalies. Figure 10-1 shows available GE swelling data for solution annealed Type-316 stainless steel fuel pin cladding previously obtained from F2 series fuel pin clad samples.^{10-4,10-5} The 505°C (941°F) specimen from EIH-26 is in good agreement with the previous data. However, the 573°C (1063°F) specimen from this pin shows 4.69% $\Delta V/V$ which is much greater than anticipated. The maximum temperature for swelling of solution annealed Type 316 is thought to be in the range of 500 to 520°C (932 to 968°F). The EIH-26 specimen irradiated at 573°C (1063°F) should show low swelling values; however, the volume change measured is high and approaches the anomalous high swelling values reported for solution annealed Type-316 claddings from DFR irradiated pins. This anomalous swelling has been speculated to be a consequence of "stress induced void growth." TEM examination of this specimen is planned for the next quarter.

Table 10-1
SWELLING MEASUREMENTS IN FUEL PIN CLADDING

Specimen	Cladding	Axial Location ^(a)	Fluence X 10 ²² (E > 0.1 MeV)	Temperature During Irradiation		Density (g.cm ⁻³)	$\Delta\rho/\rho$ (%)
				(°C)	(°F)		
F3B-2-D2	S/A 316 ^(d)	4.0 (midlength)	3.16	468	875	7.9359	-0.34
F3B-2G	S/A 316	5.7 (midlength)	3.26	494	922	7.9472	-0.19
F3B-2-K1	S/A 316	11.8 (midlength)	2.52	558	1035	7.9504	-0.15
F3B Control	S/A 316	----- Unirradiated -----				7.9626	---
E1H-26-D	S/A 316	1.85-3.35	4.53	505	940	7.8088	-1.96
E1H-26-I	S/A 316	7.1-8.85	4.84	573	1065	7.5916	-4.69
E033	S/A 316	----- Unirradiated E1H-26 Archive -----				7.9652	---
E1H-29-B	13% C.W. 304 ^(b)	1.8-2.8	4.74	500	930	7.8343	-0.85
E1H-29-H	13% C.W. 304 ^(b)	10.8-11.8	4.19	593	1100	7.8847	-0.21
H-043	13% C.W. 304 ^(b)	----- Unirradiated E1H-29 Archive -----				7.9014	---
E1H-30-B	13% C.W. 304 + HT ^(c)	1.9-2.9	4.71	503	935	7.5401	-4.32
E1H-30-H	13% C.W. 304 + HT ^(c)	10.9-11.9	4.10	593	1100	7.8405	-0.51
H-102	13% C.W. 304 + HT ^(c)	----- Unirradiated E1H-30 Archive -----				7.8808	---
F5M-B	S/A 321	2.8-3.8	3.12	527	980	7.8556	-0.52
F5M-F	S/A 321	7.5-8.5	3.21	595	1103	7.8196	-0.98
F5M-J	S/A 321	12.0-13.0	2.32	627	1160	7.8711	-0.32
21-020	S/A 321	----- Unirradiated F5M Archive -----				7.8966	---
F5F-B	S/A 316	3.0-4.0	3.20	526	979	7.9204	-0.38
F5F-E	S/A 316	7.2-8.2	3.33	592	1098	7.9135	-0.47
F5F-H	S/A 316	12.0-13.0	2.36	625	1157	7.9376	-0.17
16-028	S/A 316	----- Unirradiated F5F Archive -----				7.9507	---
F5Q-B	50% C.W. 316 ^(b)	2.5-3.5	3.08	526	979	7.9499	+0.07
F5Q-E	50% C.W. 316 ^(b)	6.8-7.8	3.29	592	1098	7.9467	+0.03
F5Q-I	50% C.W. 316 ^(b)	11.8-12.8	2.38	625	1157	7.9506	+0.07
16-566	50% C.W. 316 ^(b)	----- Unirradiated F5Q Archive -----				7.9447	---

(a) Inches above bottom of EBR-II core.

(b) Refers to reduction in area.

(c) HT was 24 hours at 900°F and 216 hours at 1300°F with air quenching.

(d) S/A denotes solution anneal.

Figure 10-1. Swelling of Solution Annealed 316 Fuel Cladding

By contrast, the three data points from pin F3B-2 show lower swelling than anticipated from the previous GE data. However, the swelling values are quite low and may be partially due to measurement errors.

The solution annealed Type 321 from fuel pin F5M shows more swelling than measured for solution annealed Type-316 cladding from a companion pin, F5F, although both sets of data fall within the general data scatter for solution annealed Type 316.

The cold-worked Type-304 cladding from EIH-29 displayed only moderate swelling showing that the cold work has at least delayed the onset of void formation. The Garafalo heat-treated Type 304 (EIH-30) showed considerable swelling (up to 4.32% $\Delta V/V$). The pre-irradiation heat treatment [24 hours at 900°F (482°C)] followed by 216 hours at 1300°F (704°C) apparently negated the beneficial effects of cold working.

Table 10-2 compares the measured increase in diameter of the fuel pins at the locations of the density specimens with the measured change in density. The F5 series pins show no correlation between the disparate cladding volume increases and the overall pin $\Delta D/D$'s. It would be expected that the pin which showed no clad swelling (F5Q) would also show the smallest overall diameter change, since the pins all used similar fuel (approximately 90% smear density, 3 to 6 mil diametral gap). It is probable that the increase in diameter due to clad swelling alone is small (maximum swelling occurred in F5M of 0.98% is equivalent to an increase in diameter of 0.0008 inch) compared to the inequalities found in the irradiated pin profiles.

In the case of the EIH pins, sufficient swelling has occurred to allow comparisons to be made. The overall $\Delta D/D$ of the EIH-26 (S/A Type 316) specimen from near the bottom of the core was found to be all due to clad swelling, whereas the specimen from near the midplane shows 0.6% plastic deformation despite the extensive clad swelling in this region. It is significant that the fuel cladding diametral gap in this region was much smaller than elsewhere in the pin.¹⁰⁻⁶ The cold-worked and heat-treated Type-304 cladding from pin EIH-30 (high fuel smear density of

approximately 93% TD) shows very high plastic deformation of up to 1.7%. This is slightly greater than the maximum value of 1.5% previously measured in pin F2G (Type-347 clad) of the F2 series irradiation.¹⁰⁻⁴ The result again clearly demonstrates that much greater ductility can be obtained in fuel pin claddings than would be predicted from post-irradiation burst testing alone, and points to the strong dependence of ductility on the loading mechanism.

10.3.2 Swelling of High-Nickel Alloys

Tensile specimens and precut electron microscopy discs of Incoloy-800, Hastelloy-X, and Inconel-625 were irradiated in a Row 4 EBR-II position in a GE materials capsule designated MT-3. Irradiation temperatures of the specimens were 600 to 650°C (1112 to 1202°F) and the maximum fluence attained was 5.5×10^{22} n/cm², E > 0.1 MeV-GE. The chemical analyses of the alloys are listed in Table 10-3. After irradiation and capsule disassembly, immersion density measurements were made of swelling in selected tensile specimens and the precut discs were examined by transmission microscopy.

The immersion density and the TEM results are listed in Tables 10-4 and 10-5, respectively. The unirradiated densities were obtained from archive specimens of the same heats of material. It was found that the Incoloy-800 showed 0.05 to 0.5% swelling, the Hastelloy-X 0.6 to 2.1% swelling, and the Inconel-625 showed densifications of up to 0.6%.

Table 10-2
PLASTIC DEFORMATION OF CLADDING SECTIONS

Specimen	Cladding Swelling ^(a) $\Delta V/V$ (%)	Diameter Change ΔD (mils)	$\Delta D/D$ (%)	Cladding Swelling ^(b) $\Delta D/D$ (%)	Difference Plastic Deformation (%)
E1H-26-D (S/A-316) ^(c)	1.96	1.7	0.69	0.65	0.04
E1H-26-1 (S/A-316)	4.69	5.4	2.15	1.56	0.59
E1H-29-B (13% C.W.-304)	0.85	3 ^(d)	1.2	0.28	0.9
E1H-29-H (13% C.W.-304)	0.21	2 ^(d)	0.8	0.07	0.7
E1H-30-B (13% C.W.-304 + HT)	4.32	6.5	2.60	1.44	1.16
E1H-30-H (13% C.W.-304 + HT)	0.51	4.6	1.83	0.17	1.66
F5M-B (S/A-321)	0.52	1.3	0.52	0.17	0.35
F5M-F (S/A-321)	0.98	1.6	0.64	0.33	0.31
F5M-J (S/A-321)	0.32	0.75	0.30	0.11	0.19
F5F-B (S/A-316)	0.38	1.1	0.44	0.13	0.31
F5F-E (S/A-316)	0.47	0.9	0.36	0.16	0.20
F5F-H (S/A-316)	0.17	0.5	0.20	0.06	0.14
F5Q-B (C.W.-316)	-0.07	1.1	0.44	0	0.44
F5Q-E (C.W.-316)	-0.03	1.3	0.52	0	0.52
F5Q-I (C.W.-316)	-0.07	0.6	0.24	0	0.24

Footnotes:

(a) As determined by immersion measurements

(b) $\frac{\Delta D}{D} \cong \frac{1}{3} \frac{\Delta V}{V}$

(c) S/A denotes solution annealed

C.W. denotes reduction in area

HT denotes 24 hours at 900°F and 216 hours at 1300°F with air quenching

(d) Micrometer readings

Table 10-3
CHEMICAL COMPOSITIONS OF NICKEL ALLOYS

Element	Incoloy-800 HH-9696A ^(a) (%)	Inconel-625 Y8668 ^(a) (%)	Hastelloy-X X1-4370 ^(a) (%)
C	0.06	0.03	0.10
Mn	0.89	0.12	0.60
Fe	45.70	1.86	18.7
S	0.007	0.007	0.006
Si	0.41	0.25	0.85
Cu	0.13	0.03	-
Ni	32.03	62.31	Bal
Cr	20.75	21.98	21.68
Al	0.10	0.17	-
Ti	0.21	0.23	-
Mo	-	8.75	9.17
Nb + Ta	-	4.24	-
B	0.0030	0.0065	0.0040
W	-	-	0.44
Co	-	-	1.87

(a) Heat numbers

Table 10-4
DENSITY DATA FROM NICKEL ALLOYS

Alloy	Fluence $E > 0.1 \text{ MeV-GE}$ $\times 10^{22}$	Temperature During Irradiation ^(a)		Density (Pre-Irradiation) (g.cm^{-3})	Density (Post-Irradiation) (g.cm^{-3})	$\frac{\Delta\rho}{\rho}$ (%)
		(°C)	(°F)			
Incoloy-800	5.4	643	1190	7.997	7.960	-0.46
Incoloy-800	5.4	643	1190	7.997	7.986	-0.14
Incoloy-800	5.4	643	1190	7.997	7.993	-0.05
Incoloy-800	4.5	603	1115	7.997	7.987	-0.13
Hastelloy-X	5.4	643	1190	8.252	8.206	-0.56
Hastelloy-X	5.4	643	1190	8.252	8.103	-1.8
Hastelloy-X	4.5	603	1115	8.252	8.083	-2.1
Inconel-625	5.4	643	1190	8.452	8.500	+ 0.56
Inconel-625	5.4	643	1190	8.452	8.471	+ 0.23
Inconel-625	4.5	603	1115	8.452	8.486	+ 0.40

(a) The minimum temperature. Temperature range over the entire specimen was 1190 to 1280 for 1190° F specimens, and 1115 to 1225 for 1115° F specimens.

Table 10-5
TRANSMISSION MICROSCOPY DATA FROM NICKEL ALLOYS

Alloy	Fluence ($E > 0.1$ MeV-GE)	Temperature During Irradiation		Void Concentration (cm^{-3})	Mean Void Diameter ($\text{\AA}$)
		($^{\circ}\text{C}$)	($^{\circ}\text{F}$)		
Incoloy-800	5.0×10^{22}	630	1164	$< 10^{13}$	~ 850
Hastelloy-X	5.0×10^{22}	630	1164	$\sim 5 \times 10^{12}$	2000-3000
Inconel-625	5.0×10^{22}	630	1164	No Voids	

The transmission microscopy results are equally dramatic. The Incoloy-800 discs contain voids with an average diameter of about 850 $\text{\AA}$ and a number density of less than 10^{13} voids per cm^3 . Figure 10-2 shows a typical area from a TEM specimen. Some very small pores were detected, especially at twin boundaries. These pores are probably helium bubbles. The microstructure is generally clean except for a few carbide particles (M_{23}C_6) which are of similar size to the voids, and frequently have voids associated with them. The general microstructure is very similar to solution annealed Type-304 stainless steel irradiated at the same temperature in the MT-3 capsule, i.e., a normal austenitic structure containing few matrix precipitates. By contrast, the Hastelloy-X discs contain very large voids and a considerable quantity of large rod-like precipitates as seen in Figure 10-3. The voids are so large (2500 to 10,000 $\text{\AA}$ diameter on the micrographs) that it is obvious that most of them have been enlarged during thinning. However, examination of very thick areas of the foil which were otherwise opaque showed clearly defined voids without extinction contours with diameters up to 3000 $\text{\AA}$. Also, the void number density is very small (approximately 5×10^{12} voids/ cm^3) and an average void size of approximately 2000 $\text{\AA}$ is required to achieve the volumetric swellings of up to 2.1% measured by liquid immersion in the tensile specimens. Thus it is evident that the swelling measured in the Hastelloy-X is due to a small number of very large voids. Comparison with unirradiated Hastelloy-X showed that the rod-shaped precipitates seen in Figure 10-2 appeared during irradiation and, from the work of Comprelli and Lewis,¹⁰⁻⁸ they are most probably M_6C .

Micrographs of the Inconel-625 are presented in Figures 10-4 and 10-5 and it is seen that no voids have been generated. Instead, the microstructure shows a very high density of finely divided precipitate which is probably Ni_3Nb , a gamma-prime precipitate.* The unirradiated structure showed only large cubic precipitates of M_{23}C_6 which are also evident in the post-irradiation structure (Figure 10-4). The Ni_3Nb has obviously appeared during in-reactor aging and has inhibited void formation as well as caused densification of the alloy. The precipitate is in the form of discs 1000 to 2000 $\text{\AA}$ in diameter and 100 to 200 $\text{\AA}$ thick located on (100) habit planes with approximately 500 $\text{\AA}$ interlamellar spacing. In some locations, particularly twin boundaries, the particles have coalesced.

The vast differences in swelling behavior of these alloys is attributed to their differing aging characteristics. Incoloy-800 with Ti content below 0.4% is a fully austenitic solid solution alloy with no purposeful additions for second-phase strengthening and thus behaves much as would be expected on the basis of results from solution annealed Type-304 stainless steel [i.e., showing moderate swelling at 630 $^{\circ}\text{C}$ (1164 $^{\circ}\text{F}$)]. Limited M_{23}C_6 precipitation appears and assists void nucleation. Otherwise, void formation is random throughout the structure excepting the grain boundary areas, which are denuded.

The Hastelloy-X forms a relatively coarse carbide precipitate which is rod-shaped, 7000 to 10,000 $\text{\AA}$ in length by about 1000 $\text{\AA}$ thick. The rods are incoherent and spaced far enough apart for dislocations to move between them. This is the likely cause of the appearance of the very large voids. The dislocations are able to move during irradiation

* Identification of this precipitate by X-ray diffraction techniques is in progress.

Figure 10-2. Incoloy-800 Irradiated to 5.0×10^{22} n/cm² $E > 0.1$ MeV-GE at 630°C (1164°F). Mean Void Diameter = 850Å. Helium Bubbles in Twin Boundary (B). Voids Denoted (A).

Figure 10-3. Hastelloy-X Irradiated to 5.0×10^{22} n/cm^2 $E > 0.1$ MeV-GE at $630^\circ C$ ($1164^\circ F$). Mean Void Diameter = $2000-3000 \text{ \AA}$. Coarse M_6C Precipitation.

Figure 10-4. Inconel-625 Irradiated to $5.0 \times 10^{22} \text{ n/cm}^2$ $E > 0.1 \text{ MeV-GE}$ at 630°C (1164°F). No Voids. Dense Overaged Ni_3Nb Precipitation. Scattered Large M_{23}C_6 Particles.

Figure 10-5. Inconel-625 Irradiated to $5.0 \times 10^{22} \text{ n/cm}^2$ $E > 0.1 \text{ MeV-GE}$ at 630°C (1164°F). High Magnification Showing Closely Spaced Ni_3Nb Disc-Shaped Precipitate.

accepting more interstitials than less mobile vacancies arising from the neutron bombardment. The resultant excess vacancies nucleate and grow voids on the surfaces of the precipitates. Once nucleated, the voids continue to grow from the excess vacancies arising from the imbalance of point defects caused by the moving dislocations.

By contrast, the Ni_3Nb precipitate in the Inconel-625 is on a much finer scale. Tilting the foil revealed very few dislocations and it appears that dislocations are intimately tied to the precipitate. Thus it appears that the precipitate particles are located close enough together to prevent major dislocation motion such that large excess vacancy concentrations do not occur and no voids nucleate.

Swelling resistance in Nimonic PE-16 has been ascribed to the presence of coherent gamma-prime precipitate $\text{Ni}_3(\text{Al,Ti})$. The coherence and interparticle spacing were considered critical. However in the case of the Inconel-625, precipitate coherence was not maintained and still no voids appeared. Thus incoherent particles, if sufficiently finely dispersed, can apparently inhibit swelling very effectively.

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

10.4.1 Tensile Testing of High-Nickel Alloys

Tensile tests have been completed on specimens of three high-nickel alloys, Incoloy-800, Inconel-625, and Hastelloy-X from the MT-3 capsule. The specimens had been irradiated in EBR-II to fluences of approximately $5 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. The temperatures of irradiation ranged from 594 to 650°C (1100 to 1202°F). Tensile tests on both irradiated specimens and on unirradiated controls were performed at three test temperatures: 482°C (900°F), 594°C (1100°F), and 704°C (1300°F). The effect of strain rate was investigated at the two higher temperatures by testing at three strain rates: $2 \times 10^{-4} \text{ min}^{-1}$, $2 \times 10^{-2} \text{ min}^{-1}$, and 2 min^{-1} . All stress/strain values were determined from load-deflection measurements. Tables 10-6 and 10-7 summarize all the test data obtained.

The results from these tests are shown in Figures 10-6, 10-7, and 10-8 for the three alloys.

The temperature of irradiation for all these alloys was well above 600°C (1112°F) and structural hardening by vacancy clustering and dislocation loop interaction would be expected to be minimal due to thermal annealing concurrent with the defect generation. Both the Hastelloy-X and the Inconel-625, however, showed hardening manifested by increased ultimate tensile strength and 0.2% yield stress values.

For example, at a test temperature of 594°C (1100°F), the 0.2% yield strength of Inconel-625 increased from 45 ksi to 103 ksi and the ultimate tensile strength increased from 115 ksi to 135 ksi. The 0.2% yield strength of Hastelloy-X increased from 30 ksi to 53 ksi and the ultimate tensile strength increased from 87 ksi to 100 ksi. In contrast, Incoloy-800 showed reduced ultimate tensile strength values and unchanged values for the 0.2% yield stress. Total and uniform elongation values for all the alloys were substantially reduced. Minimum total elongation values were recorded at the 704°C (1300°F) test temperature, i.e., Incoloy-800 7%, Inconel-625 2.5%, and Hastelloy-X 7%.

All mechanical property values resulting from the irradiated alloy tensile tests decreased with increasing test temperature. For the unirradiated material, all alloys showed a decreasing ultimate tensile strength and uniform elongation with increasing test temperature. The response of total elongation and 0.2% yield stress values to a test temperature increase, however, varied with each alloy. Incoloy-800 displayed a reduction in the 0.2% yield stress with temperature increase and an increase in total elongation. For Hastelloy-X, neither 0.2% yield stress nor total elongation values changed substantially with test temperature increase. Inconel-625 showed reducing total elongation values and an unchanged 0.2% yield stress.

At a test temperature of 704°C (1300°F), all mechanical property values for the irradiated alloys, with the exception of the Incoloy-800 elongation values, increased with increasing strain rate. Incoloy-800 showed a ductility minimum at the $2 \times 10^{-2} \text{ min}^{-1}$ strain rate.

Table 10-6
TENSILE TEST RESULTS FROM IRRADIATED HIGH-NICKEL ALLOY SAMPLES

Specimen Number	Alloy Type	Region Irradiation Temperature (°C)	Fluence X 10 ²² (>0.1 MeV)	Test Temperature (°C)	(°F)	Crosshead Speed (min)	Stress Yield Strength X 10 ⁻³	Ultimate Tensile Strength X 10 ⁻³ (kpsi)	Elongation Uniform (%)	Total (%)
L129	Inc 800	1(624)	5.0	482	900	0.02	29.3	58.2	13.5	14.9
L171	Inc 800	2(637)	5.4	593	1100	2.0	28.3	58.2	24.0	26.4
L112	Inc 800	1(624)	5.0	593	1100	2.0	25.0	58.2	24.4	26.8
L167	Inc 800	4(625)	5.0	593	1100	0.02	25.6	51.7	14.3	14.5
L145	Inc 800	6(509)	3.52	593	1100	0.02	35.0	56.6	13.3	14.8
L116	Inc 800	2(637)	5.4	593	1100	0.0002	27.0	32.3	3.7	5.3
L1 5	Inc 800	5(595)	4.25	704	1300	2.0	28.4	48.4	16.0	18.8
L155	Inc 800	3(637)	5.35	704	1300	0.02	23.3	23.6	0.7	1.6
L165	Inc 800	4(625)	5.0	704	1300	0.02	21.3	28.6	5.5	6.9
L1 7	Inc 800	5(595)	4.25	704	1300	0.02	24.3	33.7	6.4	10.2
L152	Inc 800	6(509)	3.52	704	1300	0.02	26.3	34.6	2.8	3.3
L118	Inc 800	3(637)	5.35	704	1300	0.0002	17.0	17.3	0.55	15.4
X1 3	Hast X	5(595)	4.25	482	900	0.02	84.3	131.0	11.6	11.7
X150	Hast X	2(637)	5.4	593	1100	2.0	78.3	95.5	7.8	10.0
X136	Hast X	1(624)	5.0	593	1100	2.0	79.0	107.8	8.4	10.8
X1 4	Hast X	4(625)	5.0	593	1100	0.02	53.2	101.2	8.6	8.7
X170	Hast X	6(509)	3.52	593	1100	0.02	71.4	103.0	3.8	3.9
X134	Hast X	2(637)	5.4	593	1100	0.0002	52.3	74.4	4.8	5.3
X137	Hast X	5(595)	4.25	704	1300	2.0	93.4	98.4	6.8	9.2
X140	Hast X	3(637)	5.35	704	1300	0.02	48.0	72.0	6.7	7.2
X113	Hast X	4(625)	5.0	704	1300	0.02	52.7	58.4	2.6	6.8
X169	Hast X	6(509)	3.52	704	1300	0.02	62.6	68.4	3.1	3.2
X131	Hast X	3(637)	5.35	705	1302	0.0002	38.0	39.4	1.1	3.0
V150	Incl 625	1(624)	5.0	482	900	0.02	102.7	148.5	15.4	15.5
V170	Incl 625	2(637)	5.4	593	1100	2.0	112.7	140.0	15.2	18.4
V110	Incl 625	4(625)	5.0	593	1100	0.02	102.3	135.0	5.3	5.4
V128	Incl 625	6(509)	3.52	593	1100	0.02	129.0	132.0	1.9	2.0
V1 4	Incl 625	2(637)	5.4	593	1100	0.0002	111.0	125.0	3.0	3.1
V179	Incl 625	5(595)	4.25	704	1300	2.0	166.5	171.5	7.2	11.2
V142	Incl 625	3(637)	5.35	704	1300	0.02	79.0	81.7	1.3	1.4
V133	Incl 625	4(625)	5.0	704	1300	0.02	86.0	98.4	2.8	2.9
V122	Incl 625	6(509)	3.52	704	1300	0.02	105.0	109.0	1.7	1.8
V1 9	Incl 625	3(637)	5.35	704	1300	0.0002	58.0	58.0	1.3	1.4

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Table 10-7
TENSILE TEST DATA FROM HIGH-NICKEL ALLOY CONTROL SAMPLES

Specimen Number	Alloy Type	Test Temperature		Crosshead Speed (min)	Stress Yield Strength $\times 10^{-3}$	Ultimate Tensile Strength	Elongation	
		(°C)	(°F)			$\times 10^{-3}$ (kpsi)	Uniform (%)	Total (%)
L11	Inc 800	482	900	0.02	28.7	74.2	32.2	35.0
L166	Inc 800	482	900	0.02	29.6	74.8	33.0	34.3
L130	Inc 800	593	1100	2.0	27.6	63.8	27.2	29.2
L117	Inc 800	593	1100	0.02	26.5	57.5	28.0	50.0
L115	Inc 800	593	1100	0.0002	26.6	41.7	17.1	61.7
L136	Inc 800	704	1300	2.0	25.2	51.1	25.6	38.8
L162	Inc 800	704	1300	0.02	25.5	36.8	14.0	61.0
L16	Inc 800	704	1300	0.0002	16.3	16.6	4.4	89.9
X151	Hast X	482	900	2.0	30.0	84.0	47.6	50.8
X123	Hast X	482	900	0.02	26.7	86.6	45.4	47.6
X121	Hast X	482	900	0.02	30.5	93.2	46.0	46.8
X149	Hast X	482	900	0.02	26.3	85.0	49.0	49.8
X153	Hast X	482	900	0.02	28.4	85.0	48.8	52.0
X174	Hast X	482	900	0.02	29.4	90.0	50.6	51.6
X187	Hast X	482	900	0.02	22.7	78.4	52.6	54.4
X1555	Hast X	482	900	0.02	22.7	78.5	53.6	56.2
X127	Hast X	593	1100	2.0	31.7	86.7	44.8	46.0
X186	Hast X	593	1100	2.0	30.0	83.4	43.2	47.2
X154	Hast X	593	1100	0.02	31.4	92.0	48.0	49.4
X178	Hast X	593	1100	0.02	27.6	84.2	45.0	47.0
X11	Hast X	593	1100	0.0002	28.3	69.0	30.7	51.4
X162	Hast X	704	1300	2.0	30.7	78.0	41.2	42.4
X161	Hast X	704	1300	0.02	27.4	62.1	30.0	50.0
X184	Hast X	704	1300	0.0002	29.3	37.0	5.7	57.6
V138	Incl 625	482	900	0.02	43.5	113.0	47.0	47.8
V172	Incl 625	482	900	0.02	45.3	110.0	44.6	45.6
V182	Incl 625	593	1100	2.0	71.6	109.0	31.6	34.8
V151	Incl 625	593	1100	0.02	45.2	118.0	45.0	45.0
V162	Incl 625	593	1100	0.02	47.0	117.4	47.0	48.4
V133	Incl 625	593	1100	0.0002	46.6	96.6	16.0	16.4
V168	Incl 625	593	1100	0.0002	78.8	105.0	10.0	10.6
V141	Incl 625	704	1300	2.0	46.1	108.0	46.8	53.2
V120	Incl 625	704	1300	0.02	43.5	82.4	18.0	31.0
V157	Incl 625	704	1300	0.0002	43.0	62.6	14.0	14.7

At the 594°C (1100°F) test temperature, all property values, with the exception of the yield stress, increased with increasing strain rate. The yield stress values tended to remain comparatively unaffected, although Hastelloy-X displayed a considerable increase at the fastest (2 min^{-1}) strain rate.

All unirradiated alloys tested at 706°C (1300°F) showed increased tensile strength and uniform elongation values. The response of other properties varied considerably, however. Inconel-625 and Hastelloy-X displayed the same lack of yield strength sensitivity to strain rate that they had previously shown to test temperature. Incoloy-800 showed an increase in yield strength. Total elongation values decreased with increasing strain rate for Incoloy-800 and Hastelloy-X, but increased for Inconel-625 (Figures 10-6, 10-7, and 10-8).

Figure 10-6. The Effect of Strain Rate on the Mechanical Properties of Incoloy 800 Tested at 1300°F (704.4°C)

Figure 10-7. The Effect of Strain Rate on the Mechanical Properties of Inconel 625 Tested at 1300°F (704.4°C)

Figure 10-8. The Effect of Strain Rate on the Mechanical Properties of Hastelloy-X Tested at 1300°F (704.4°C)

Little positive similarity in behavior was observed for the unirradiated alloys tested at 594°C (1100°F). With respect to general trends, Incoloy-800 and Hastelloy-X showed increasing values for ultimate tensile strength and uniform elongation, decreasing values for total elongation and comparative insensitivity of yield strength values to strain rate.

Inconel-625 showed effectively no change in ultimate tensile strength (although a slight increase was observed at the $2 \times 10^{-2} \text{ min}^{-1}$ strain rate). The 0.2% yield stress showed a pronounced dip at the intermediate strain rate $2 \times 10^{-2} \text{ min}^{-1}$ and maximum elongation values, total and uniform, at the intermediate $2 \times 10^{-2} \text{ min}^{-1}$ strain rate.

The data described above indicate that, despite high temperature during irradiation [greater than 600°C (1115°F)], the increased ultimate tensile strength and 0.2% yield stress values of Inconel-625 and Hastelloy-X after irradiation would at first be attributed to irradiation hardening defined as the interaction between dislocations and irradiation produced obstacles. There is evidence to suggest¹⁰⁻¹⁰ that impediments to screw dislocation movement are prime factors in determining the yield stress. Such impediments would necessarily require large strain yields such as those generated by vacancy clusters or dumbbell-shaped interstitials (two atoms sharing one lattice site). However, this type of hardening should be minimal at these irradiation temperatures. It has been shown¹⁰⁻¹¹ that at high irradiation temperatures the mechanism of hardening is by voids according to the Coulomb expression:

$$\Delta\sigma = \frac{2Gb}{L}$$

where

G = shear modulus

b = Burgers Vector

and

L = average spacing along a dislocation where

$$L = (Nd)^{-1/2}$$

N is the cavity density, and

d is the cavity diameter.

However, in the transmission electron microscopy studies discussed earlier, voids were not detected in Inconel-625 (a volume decrease was observed) and, in the case of the Hastelloy-X, the voids were few and very large (mean void diameter 2000 Å).

Furthermore, Incoloy-800 irradiated at the same temperature showed a decrease in strength. Comprelli¹⁰⁻¹² has reported strength increases from 200 to 250% in tests at 594°C (1100°F) and 704°C (1300°F) on unirradiated Inconel-625 after soaking for 5000 hours at 648°C (1200°F). The 122% and 78% yield strength increases noted in the present series of tests are well within these limits. Similarly, exposure of Hastelloy-X for 1000 hours at 594°C (1100°F) almost doubled the 0.2% yield stress (39 ksi to 70 ksi) tested at 594°C (1100°F) which is comparable to the increase from 30 ksi to 55 ksi noted in the present studies.

However, Comprelli showed a minimum total elongation of 10 to 15% for Inconel-625 and 20% for Hastelloy-X as compared to the 2.5% and 7% elongation observed in the irradiated material.

From a consideration of the above data, and as a result of the TEM studies, the hardening of the Inconel-625 and the Hastelloy-X can be attributed to thermally induced precipitation, Ni_3Nb probably associated with TiN and NbC in the case of Inconel-625, and M_6C in the case of Hastelloy-X. The relative insensitivity of the unirradiated Hastelloy-X 0.2% yield strength to increasing strain rate compared to the irradiated alloy would apparently be due to the more stable nature of the solid solution hardened matrix of the unirradiated material.

The most significant feature of these data is the similarity of response of all the irradiated alloys to a strain rate increase or a temperature decrease (in general, all property values increased) as compared to the unirradiated specimens whose mechanical property values show widely varying responses to changes in strain rate or test temperature.

10.4.2 Burst Testing of F3A Fuel Pins

Burst tests have been completed on 36 3-inch long sections from six F3A fuel pins. The F3A experiment consisted of unencapsulated mixed oxide fuel pins irradiated in EBR-II to approximately 5.6 at. % burnup (50,000 MWd/Te) at 14 to 16 kW/ft nominal peak linear power. The peak calculated fluence was approximately 3×10^{22} n/cm² > 0.1 MeV-GE. Earlier metallographical examination of these pins had revealed significant intergranular attack (0.004 to 0.009 inch maximum) at the internal surface of the cladding. No relationship was found between the extent of the attack and alloy type or any fuel fabrication parameter.

It was observed, however, that the depth of penetration increased with increasing cladding temperature during irradiation. Cladding attack was not detected in the regions exposed to temperatures calculated to be less than 538°C (1000°F). Burst tests were carried out on six fuel pins: F3B-6, F3B-9, F3B-11, F3B-12, F3B-15, and F3B-16 to determine the reduction, if any, in mechanical properties as a result of the intergranular attack. Pins F3B-6 and F3B-11 were clad with Type-304L stainless steel. The remaining four pins were clad with Type-316 stainless steel. In view of the experience of other workers,¹⁰⁻¹³ who noted substantial reductions in burst strength values after prolonged exposure of opened fuel pins to air, precautions were taken to limit air exposure. Each fuel pin was evacuated and backfilled with argon immediately after puncturing. All specimens were maintained in an argon atmosphere until just prior to testing. An attempt was made to complete the testing of all specimens from a single fuel pin within a 12-hour period after cutting the sections.

A high temperature extensometer specifically designed for the purpose was used to obtain a pressure-deflection curve which permitted determination of the 0.2% yield stress and a better estimate of ductility than is normally feasible in this type of test. Six 3-inch long specimens were taken from each fuel rod. Four of the specimens were from the fueled region and the remaining two were from the blanket regions.

The 3-inch long sections were sealed with "Swagelok" fittings and pressurized with argon. The mixed oxide fuel was not removed from sections cut from the fueled regions of the pins. All the tests were carried out at a pressurization rate of 1000 psi/min at 594°C (1100°F).

The burst test results are summarized in Table 10-8. Rupture values varied from 37.7 ksi and 65.7 ksi for the Type-304L clad pins and from 55.3 ksi to 89.1 ksi for the Type-316 stainless steel pins. The range of yield stress for the Type-304L and -316 cladding is 32.1 to 63.2 ksi and 48.1 to 85.5 ksi respectively. Ductility values were generally less than 1% for all cladding taken from the fuel region, but ranged from 0.14% to 6.9%.

Yield stress and rupture stress decreased very rapidly for the Type-316 cladding samples irradiated above 510°C (950°F) whereas the yield and rupture stress values decreased rapidly above an irradiation temperature of 454°C (850°F) for the Type-304L cladding.

As stated previously, the intergranular attack of the cladding was observed at temperatures greater than 538°C (1000°F) for both alloys. In this temperature region, recovery of irradiation damage would also be expected to become significant and it is not surprising to note that the rupture stress after irradiation is comparable to unirradiated values for Type-316 control samples also tested at 594°C (1100°F). A slight increase in ductility was generally observed with an increase in the sample irradiation temperature (Table 10-8). This is attributed to a decrease in the matrix hardening by dislocation loop and vacancy interaction. The contribution to embrittlement by helium bubble formation would not be expected to decrease markedly with increasing irradiation temperature.

To separate the individual contribution of the two effects, irradiation temperature and intergranular attack, comparable mechanical property data are required on materials irradiated without fuel. This comparison is presently in progress as part of this program.

The sharp reduction in strength properties of Type-304L and Type-316 stainless steel from the F3A fuel pins is due to the combination of radiation damage, annealing and intergranular attack. Metallography of the burst specimens will be utilized to establish the extent of intergranular attack present. Only one of the 36 specimens tested had burst rupture strengths below the unirradiated control values.

Table 10-8
RESULTS FROM F3A FUEL PIN BURST TESTS

Test Conditions—Temperature 1100°F (593°C)
Pressurization Rate 1000 psi/min
Fuel Not Removed

Specimen	Number	Alloy	0.2% Yield Strength (psi)	Burst Strength (psi)	Elongation (%)	Calculated Irradiation Temperature Cladding Inside Surface (°F) (°C)		Axial Location From Core Bottom (In.)
Control	OC1	304L	17380	45502	17.5			
Control	OC2	304L	17212	45562	17.5			
Control	OC3	304L	17550	47587	18.2			
Control	IC2	304L	17550	46710	19.0			
Control	IC3	304L	17550	45900	15.5			
F3B	9-A	304L	Est 59800	59872	0.38	700	371	— 3.1 → 0 Lower Insulator
F3B	9-B	304L	Est 63230	63432	0.48	875	468	0 → 3 Fuel
F3B	9-D	304L	54911	55317	0.48	958	514	3.5 → 6.5 Fuel
F3B	9-E	304L	41974	46036	1.6	1024	551	6.5 → 9.5 Fuel
F3B	9-G	304L	32122	37734	0.86	1100	593	10 → 13 Fuel
F3B	9-J	304L	36450	46035	8.0	1024	551	14 → 17 Upper Insulator
F3B	11-A	304L	—	65690	0.69 ^(a)	700	371	— 3.1 → 0 Lower Insulator
F3B	11-B	304L	—	56847	0.14 ^(a)	818	437	0 → 3 Fuel
F3B	11-D	304L	55317	57750	1.17	910	488	3.5 → 6.5 Fuel
F3B	11-E	304L	49970	52410	0.65	960	516	6.5 → 9.5 Fuel
F3B	11-G	304L	43875	46570	0.92	994	534	10 → 13 Fuel
F3B	11-J	304L	41366	48914	3.2	918	492	14 → 17 Upper Insulator
Control	OC4	316	21937	54000	18.4			
Control	OC5	316	21260	54675	15.5			
Control	IC1	316	21260	54000	17.3			
Control	IC4	316	20925	54000	17.25			
F3B	6-A	316	Est 66700	66700	0.3	700	371	— 3.1 → 0 Lower Insulator
F3B	6-B	316	83368	86673	0.4	822	439	0 → 3 Fuel
F3B	6-D	316	82380	86432	2.4	910	488	3.5 → 6.5 Fuel
F3B	6-E	316	Est 60000	60075	0.3	966	519	6.5 → 9.5 Fuel
F3B	6-G	316	61087	71550	3.1	990	532	10 → 13 Fuel
F3B	6-J	316	56974	72684	4.9	913	489	14 → 17 Upper Insulator
F3B	12-A	316	64680	70070	1.6	700	371	— 3.1 → 0 Lower Insulator
F3B	12-B	316	82350	89100	1.1	831	444	0 → 3 Fuel
F3D	12-D	316	80135	87710	1.8	917	492	3.5 → 6.5 Fuel
F3B	12-E	316	72790	80368	2.4	960	516	6.5 → 9.5 Fuel

Table 10-8 (Continued)

Specimen	Number	Alloy	0.2% Yield Strength (psi)	Burst Strength (psi)	Elongation (%)	Calculated Irradiation Temperature Cladding Inside Surface		Axial Location From Core Bottom (in.)
						(°F)	(°C)	
F3B	12-G	316	66383	77266	6.9	986	530	10 → 13 Fuel
F3B	12-J	316	60277	73225	1.9	898	481	14 → 17 Upper Insulator
F3B	15-A	316	63790	70875	1.83	700	371	- 3.1 → 0 Lower Insulator
F3B	15-B	316	85500	87500	0.96	840	449	0 → 3 Fuel
F3B	15-D	316	74800	76200	0.36	973	523	3.5 → 6.5 Fuel
F3B	15-E	316	65600	67500	0.59	1046	563	6.5 → 9.5 Fuel
F3B	15-G	316	48145	55370	1.3	1105	596	10 → 13 Fuel
F3B	15-J	316	49950	66825	8.0	1025	552	14 → 17 Upper Insulator
F3B	16-A	316	63787	70537	1.8	700	371	- 3.1 → 0 Lower Insulator
F3B	16-B	316	86600	88152	0.76	825	441	0 → 3 Fuel
F3B	16-D	316	81150	86221	1.4	920	493	3.5 → 6.5 Fuel
F3B	16-E	316	74525	82860	3.0	964	518	6.5 → 9.5 Fuel
F3B	16-G	316	56400	67000	1.8	990	532	10 → 13 Fuel
F3B	16-J	316	56800	73500	4.8	918	492	14 → 17 Upper Insulator

(a) From micrometer measurements taken adjacent to the burst position.

10.5 SODIUM EFFECTS

10.5.1. Carbon and Oxygen Content of Irradiated Type-316 Cladding Exposed to EBR-II Primary Sodium

The F3B-2* cladding from the F3A fuel pin experiment was exposed to EBR-II primary coolant sodium during irradiation for approximately 12,900 MWd (approximately 6800 hours total). Calculations using a diffusion model indicated that the Type-316 cladding could lose up to 200 ppm carbon in the peak temperature regions assuming 510 to 538°C (950 to 1000°F) peak cladding outside surface temperatures. Subsequent temperature analyses based on measured fuel burnups have resulted in the calculated peak cladding operating temperatures being revised downward.** Perturbation of these temperatures by an adjacent high-worth driver subassembly was calculated to increase the specimen temperatures less than 25°F. The high-worth driver assembly was in the core for approximately 29% of the exposure time of the F3A fuel pin series.

The carbon analysis was performed on 0.125-inch rings after alpha cleaning and decontamination. Approximately 0.002 inch of metal was removed from the inside surface by the cleaning procedure. Samples were cut from the tube with a hacksaw to avoid possible carbon contamination from a silicon carbide cutting wheel. After cutting, the specimens were cleaned in nitric acid to remove any surface contamination. Duplicate samples of each at three axial locations were analyzed for bulk carbon content. Also included were two sets of unirradiated controls obtained from archive tubing of the same Type-316 heat. One set of three controls was cut and cleaned in-cell to determine if there was any carbon contamination. The remaining set of controls was cut and cleaned outside the β - γ cell.

The analysis for carbon content was performed by heating the sample with an induction coil and burning in oxygen along with pure iron and tin to aid combustion. The CO₂ given off was collected and analyzed using a gas chromatograph. The sequence of carbon determinations on the irradiated material was alternated with controls, standards of known carbon content, and blank burns to detect any systematic variation in the calibration of the instrument. A blank run consisted of running a crucible with only the pure iron and tin to detect the amount of residual carbon and/or CO₂ in the system.

The data in Table 10-9 indicate that, excluding D2-2 (which was almost 200 ppm higher than any other sample), the average concentration of carbon, 583 ppm, in the irradiated and unirradiated samples is identical. There is a trend in the numbers of less carbon in the irradiated samples at the hot end of the fuel pin. However, considering the statistical accuracy of the measurements of ± 32 ppm, it is not possible to conclude that any significant decarburization has occurred in-reactor at the temperatures experienced by pin F3B-2.

In addition to the control samples run during the test, another set of controls was analyzed for carbon content at an outside laboratory (Anamet, Incorporated). The results of this test gave an average carbon content of 614 $\pm$ 50 ppm. This is in good agreement with the results of the measurements performed on the control samples.

Samples of the F3B-2 fuel pin were also analyzed for bulk oxygen content by a vacuum-fusion process. Two-tenths of a gram samples were melted in a nickel bath in a carbon crucible. The resulting gas released was analyzed in a gas chromatograph for CO₂ to determine the oxygen content of the samples.

The results of the analysis are shown in Table 10-10. One sample and one control were unavailable for analysis and will be run at a later date. As can be seen in the table, the oxygen concentration in the cladding in all cases is greater than the control samples. The sample at the hot end of the cladding had the highest oxygen concentration which was approximately 100 ppm higher than the unirradiated control material.

The nitrogen concentration in the steel was also obtained and the data are included in Table 10-10. As can be seen, there is no evidence of a change in nitrogen concentration due to irradiation or exposure to flowing sodium.

*The F3B-2 fuel pin was irradiated as part of the PA-10, Task F, F3A subassembly experiment.

**For example, the peak temperature of one section (K) was reduced from 997°F to 959°F.

Table 10-9
CARBON ANALYSIS OF F3B-2, TYPE-316 STAINLESS STEEL,
HEAT NUMBER 46533, IRRADIATED IN EBR-II

Irradiated Material					
Sample	Axial Location (a)	Temperature During Irradiation		Sample	Carbon (ppm) Average
		(°F)	(°C)		
D2-2	4.06	802	428	801	710
D2-3	4.25	806	430	618	
G-2	5.86	842	450	534	576
G-3	6.05	846	452	618	
K1-2	11.86	953	512	553	572
K1-3	12.05	956	513	592	

Average Excluding D2-2 583± 32 ppm

Unirradiated Controls

VNC	ANAMET
664	590
600(b)	590
571	600
595(b)	650
536	640
581(b)	
539	
Average 583± 31 ppm	614 ± 50 ppm

- (a) Inclies above bottom of EBR-II core
 (b) Control sample prepared in-cell

10.6 MATERIALS MANAGEMENT

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

10.6.1 Type-321H Air Melted Tubing

Procurement of Type-321H air-melted tubing has been completed. Equal quantities of 1,000 feet of annealed, 10% and 20% cold-worked tubing have been received and are presently being inspected for dimensions and defects.

10.6.2 Type-321 Vacuum-Melted Tubing

The procurement of Type-321 vacuum-melted tubing is in progress. This tubing order includes equal quantities (1,000 feet) of annealed, 10% and 20% cold-worked materials.

Table 10-10
OXYGEN AND NITROGEN ANALYSIS OF F3B-2, TYPE-316 STAINLESS STEEL
HEAT NO. 46533, IRRADIATED IN EBR-II

Sample	Axial Location ^(a)	Temperature During Irradiation		Oxygen (ppm)	Nitrogen (ppm)
		(°F)	(°C)		
Control A	Unirradiated			130	405
Control B	Unirradiated			129	415
D2-1	3.88	797	425	157	406
D2-4	4.44	810	432	146	405
G-1	5.68	838	448	173	395
G-4	6.24	850	454	206	435
K1-1	11.68	950	510	228	428

(a) Inches above the bottom of EBR-II core

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II. TASK N—CORE PHYSICS ANALYSIS

Task Leader: B. A. Hutchins*

11.1 GENERAL

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

11.2 NUCLEAR DATA EVALUATION

The evaluated data of Pu-239 (material 104) for ENDF/B-III were reviewed. It is expected that this material will provide a good basis for LMFBR computations, while being consistent with differential data measurements.

Several potential trouble areas were investigated and one modification was recommended. Spin states of one-half are used for all the resolved resonances. It was verified that this is consistent with the evaluation performed at Aerojet Nuclear Corporation and that the parameters in material 104 should produce the correct cross sections. It was also recommended that the number of degrees of freedom for the average fission width of the (1,0) unresolved state be changed from 1 to 0. This will prevent the processing codes from averaging over zero widths.

11.3 COMPUTER CODE ACQUISITION AND DEVELOPMENT

Several runs were made with the computer code TRIAL, which axially links together flux sets for specific horizontal planes to form three-dimensional fluxes. The problems were based on a simple cubic reactor used as a model for checkout of the 3DB code.¹¹⁻¹ While the cubic reactor model does not lend itself to the TRIAL prescription, the results indicate that the axial flux distributions are in reasonable agreement with the 3DB results. Initially, the input flux sets, for specific (x,y) planes, were computed with two-dimensional diffusion theory and did not agree well with the fluxes in the same planes from the 3DB solution. This resulted in relatively poor flux distributions in three dimensions. For subsequent problems, two-dimensional fluxes in two axial planes were taken directly from the 3DB results in order to test only the linking capability of TRIAL. In these cases the fluxes at all planes agreed well with those from the direct 3DB calculation. More stringent tests, using an LMFBR control cell model with the poison partially inserted, have been started.

Recent versions of the ETOX¹¹⁻²-IDX¹¹⁻³ codes have been obtained from WADCO and are being converted to the GE-635 computer. Implementation of a new compiler on the 635 has eased conversion due to the addition of an extended FORTRAN capability. However, IDX core requirements may pose another type of problem in which case linking to reduce these requirements may prove to be difficult.

Arrangements are being made to use the IBM 360-75 computer at Idaho Falls for analyses associated with ZPPR-3 experiments. Use of this computer will increase BRD capability for analyzing critical experiments by making specialized codes such as RABBLE and particular modules in the ARC system directly available.

The user's manual for the TDOWN code¹¹⁻⁴ was distributed. A package containing all the codes developed by General Electric for use with the rapid cross-section adjusting technique (ENDRUN-II, PUNT, and TDOWN-II) plus associated documentation^{11-4,11-5} were sent to the Argonne Code Center.

11.4 PLANNING AND ANALYSIS OF CRITICAL EXPERIMENTS

A meeting was held in Washington, D. C. on August 16, 1971 with RDT representatives to discuss a change in the scope of Task N. Under the terms of the agreed change in scope, General Electric will provide planning and direction to the experimental program in ZPPR starting with Assembly 3. This work will be carried out under a new Project

*Contributors are listed in Appendix B.

Agreement, No. 59. The purpose of Assembly 3 is to investigate the effects of a control system on the nuclear parameters of a demonstration reactor. The experimental program is scheduled to start about January 1, 1972. General Electric will work in a cooperative effort with ANL to define the configuration and experimental program for this assembly. It was decided at a planning meeting held in Idaho on September 13 and 14, 1971, that final specifications for the initial critical configurations and experimental program would be completed by December 1, 1971. Due to the tight schedule, planning and scoping analyses have already been started.

ZPPR Assembly 3 will remain similar to Assembly 2 and still contain most of the major features of the General Electric Demonstration Reactor. ZPPR-3 will be a 2-enrichment zone assembly with each zone containing ~50% of the core volume. The drawer fuel loading will remain the same as ZPPR-2. The major difference between ZPPR-2 and ZPPR-3 is the simulation of 37 control locations in ZPPR-3 and a larger core radius. The general characteristics of ZPPR-3 are listed in Table 11-1. As indicated, the experimental program of ZPPR-3 will be separated into 3 phases corresponding to the end-of-cycle, middle-of-cycle, and beginning-of-cycle reactor control conditions. A brief description of the 3 phases of ZPPR-3 is given below:

PHASE 1: Simulate conditions at the end of an operating cycle with all control rods withdrawn.

PHASE 2: Simulate conditions near the middle of an operating cycle with approximately one-half of the burnup control rods inserted, i.e., ~\$10 in a pattern that optimizes the power distribution. Several insertion patterns will be studied.

PHASE 3: Simulate conditions at the beginning of an operating cycle with all burnup control rods inserted, i.e., ~\$20. The insertion patterns will be chosen to optimize power distribution. A few different insertion patterns will be studied.

The major blocks of experiments for ZPPR-3 are listed and briefly described in Table 11-02.

An analysis is being carried out in support of the planning to determine critical sizes, locations and description of simulated control rod channels, and the effects and locations of fuel spiking for accommodation of reactivity loss due to control materials in phases 2 and 3. From this analysis a control arrangement has been defined for a triangular array with an average pitch of 12 inches. In ZPPR, the presence of ~10% voids requires an increase in dimension of ~10% over the equivalent demonstration plant dimension. Each control location will remove fuel from 2 drawers for the desired size control channel. A tentative control location simulation is shown in Figure 11-1. Criticality will be achieved in ZPPR-3 by loading the simulated control locations, locating the Zone 1 and Zone 2 boundary near the outer ring of control location, and increasing the outer core radius until critical. The calculated critical size for this approach is as follows: Zone 1 contains 540 drawers (radius = 74.1 cm) and Zone 2 contains 503 drawers (radius = 103.0 cm). This arrangement is shown in Figure 11-2.

The accommodation of \$10 and \$20 worth of poison material (B_4C) to be loaded into the Phases 2 and 3 assemblies, respectively, will be accomplished by fuel (Pu) spiking. This is the replacement of a 1/4-inch U_3O_8 plate by a 1/8-inch Pu-Al plate and a 1/8-inch Fe_2O_3 plate in selected drawers. The spiking pattern will attempt to simulate the changes in plutonium distribution between the 3 simulated reactor conditions (phases 1, 2, and 3). Preliminary calculations indicate that 38 and 57 spike locations (over the full core height) are required in the inner core zone and outer core zone, respectively. A tentative spiking pattern is shown by the x's in Figure 11-3. One-dimensional transport (S_{16}) calculations over spiked and normal drawers indicate that fuel spiking has a negligible effect on the Pu-239 fission distribution and little effect on the U-238 fission distribution rate 2 inches from the spike.

Preliminary calculations (1-D diffusion theory in 29 energy groups) of fission rate distributions have been performed for the central control channel experiment in ZPR6-7.¹¹⁻⁶ The calculated distributions are in good agreement in shape with the measured distributions for both sodium and poison channels, with and without BeO. This comparison is presented in Figures 11-4, 11-5, 11-6, and 11-7. Both the experimental data and computed values have been normalized to unity at the maximum value for comparative purposes.

Figure 11-1. Control Location Mockup

Table 11-1
GENERAL CHARACTERISTICS OF ZPPR ASSEMBLY 3

Core Height	36 in.
Core Diameter	~ 82 in.
Number of Simulated Control Channels	37
Poison Material	B_4C (All control)
Enrichment of Poison Material	Natural to ~ 90% B-10
Nominal Control Rod Worth	~ \$1 per rod
Total Inserted Control Worth	0, \$10, \$20
Moderator	BeO in all control channels

Figure 11-2. ZPPR-3 Control Configuration

Figure 11-3. ZPPR-3 Spikes and Control Drawers

Figure 11-4. ZPR6-7 Sodium Channel Surrounded by Fuel

Figure 11-5. ZPR6-7 Boron Carbide Surrounded by Fuel

Figure 11-6. ZPR6-7 Boron Carbide Surrounded by Beryllium Oxide

Figure 11-7. ZPR6-7 Sodium Surrounded by Beryllium Oxide

Table 11-2
PRINCIPAL AREAS TO BE STUDIED IN THE ZPPR-3 EXPERIMENTS

1. **POWER DISTRIBUTION:**
 - Reaction Rates Measured with Chambers and Foils
 - Overall and Local Distributions to be Measured
 - Influence of Control Insertion Pattern to be Measured
2. **CONTROL ROD WORTH:**
 - Worth of Single Rod in Nominal and Worst Patterns for Each Phase
 - Total Worth of Scram and Shim Systems Measured with Subcritical Techniques
 - Relative Worth of Different Insertion Patterns Measured in Phases 2 and 3
3. **SODIUM VOID REACTIVITY:**
 - Both Small Sample and Large Void Patterns Measured
 - Influence of Control Measured
4. **DOPPLER REACTIVITY EFFECT:**
 - Measure in Each Phase
 - Measurements Near Control Locations With and Without B_4C and BeO
5. **MEASUREMENTS TO INTERCALIBRATE DATA FROM EACH PHASE**

The fractional increase in B_4C worth due to BeO is computed accurately, but these preliminary values give ~30% higher absolute B_4C worth than the measurements. This information is presented in Table 11-3.

The analyses being performed on ZPPR, Assembly 2, have been completed and earlier results were reported in the 39th Quarterly Report. Effort during this quarter was directed at defining a set of energy- and region-dependent axial buckling values which would permit accurate and inexpensive one-dimensional computations, including those for ZPPR-3 scoping calculations. Axial leakage data from a 7-group 2-dimensional (r-z) computation of ZPPR-2 were obtained for each radial zone. The leakage values were plotted as a function of energy and weighted by the 29 group diffusion coefficients and fluxes from a one-dimensional calculation to obtain 29 group energy- and region-dependent axial buckling values. These buckling values reproduce the k_{eff} of the 2-dimensional calculation.

For further comparison, central fission ratios were computed by diffusion theory in a 1-D computation in 29 groups and in a 2-D computation in 7 groups. This comparison is listed in Table 11-4 along with the measured fission ratios.¹¹⁻⁷ The computed radial reaction rate distributions of $B-10$, $U-235$ fission, $U-238$ fission, and $Pu-239$ fission from the 2-D calculation are presented in Figures 11-8, 11-9, 11-10, and 11-11, respectively, along with the experimental values. Figures 11-12 and 11-13 show computed and measured radial fission ratios of $Pu-239/U-235$ and $U-238/U-235$, respectively.

The radial worth traverse of $Pu-239$, computed by one-dimensional perturbation theory, is presented in Figure 11-14 and shows that the computed worth is less flat than the measurements. All radial traverse results have been normalized to unity at the reactor center.

*Obtained from Argonne National Laboratory and given in ANL memorandum ZPR-TM-48, ZPPR-2 Benchmark Experimental Data to January 1971, January 20, 1971.

Table 11.3
ZPR-6 CENTRAL CONTROL ROD EXPERIMENT

Central Control Rod	Experimental	Calculated	C/E
With BeO	\$1.64	\$2.11	1.29
Without BeO	\$1.27	\$1.62	1.28

Table 11-4
CENTRAL FISSION RATIOS IN ZPPR ASSEMBLY 2

	Measured ¹¹⁻⁷	Calculated Using	
		SN2D 7 Groups	SN1D 29 Groups
U-238/U-235	0.0201 ± 0.0004	0.0206	0.0199
Pu-239/U-235	0.9372 ± 0.0142	0.8798	0.8699
Pu-240/U-235	0.1704 ± 0.0026	0.1712	0.1657
U-238 _c /U-235 _f		0.132	
U-238/Pu-239	0.0215 ± 0.0005	0.0234	0.0229
U-235/Pu-239	1.067 ± 0.014	1.137	1.149
Pu-240/Pu-239	0.1818 ± 0.0047	0.1946	0.1905

Figure 11-8. B-10 Radial Capture Traverse in ZPPR-2

Figure 11-9. U-235 Radial Fission Traverse in ZPPR-2

Figure 11-10. U-238 Radial Fission Traverse in ZPPR-2

Figure 11-11. Pu-239 Radial Fission Traverse in ZPPR-2

Figure 11-12. Pu-239/U-235 Fission Ratios in ZPPR-2

Figure 11-13. U-238/235 Fission Ratios in ZPPR-2

Figure 11-14. Radial Reactivity Traverse for Pu-239 in ZPPR-2

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12. TASK P—FUEL ELEMENT MODELING

Task Leader: F. E. Gelhaus*

12.1 GENERAL

The objective of Task P is the development of a model which will predict fuel performance and fuel lifetime. A performance and failure model is being developed for use in parametric studies of the sodium-cooled breeder reactor. This modeling and analysis effort relates to other portions of the Fast Ceramic Reactor Development Program in that:

1. Guidance is provided in planning fuel and cladding irradiations.
2. Assistance is provided in the analysis and interpretation of experimental results from fuel and cladding irradiations.

12.2 PERFORMANCE MODEL DEVELOPMENT

12.2.1 Power Increase Effect

Cladding strain as the result of a power increase has been studied using BEHAVE-II. The model predicts that after several months of steady operation at 8.5 to 9.5 kW/ft peak, the fuel is in contact with the cladding and that cracks have healed up from the fuel center out to an isotherm at $\sim 1300^{\circ}\text{C}$. The configuration at this point is the same as in the SUR (Strong Unbroken Ring) model used in BEHAVE-I and described in the 37th Quarterly Report.¹²⁻¹ However, the detailed mechanics used in BEHAVE-II predict a larger plastic strain increment in the cladding during the power increase to melt than results using the SUR mechanics of BEHAVE-I.

A parametric study was performed to determine the effect of several variables on the predicted diameter increase; the results are presented in Figure 12-1. Diameter increase in the cladding due to plastic strain is plotted versus the linear power at which the crack healing process takes place. At low linear powers, there is no crack healing due to a low center temperature and no diameter increase is predicted to occur. As the outer boundary of the uncracked fuel region moves radially outward with increasing linear power, the stress at the root of the cracks decreases, and less plastic flow of the fuel occurs. The diameter increase reaches a peak and then declines as the crack-healing linear power is further increased. This decline is caused by the weakness of the fuel relative to the cladding at higher linear powers.

The parametric curves indicate that the effect is reduced by increasing the crack-healing temperature and also by the presence of a central void. Crack healing in UO_2 has been experimentally investigated by Bain^{12-2,12-3} who observed healing in fabricated cracks down to an estimated temperature of $1400 \pm 100^{\circ}\text{C}$. The crack-healing temperature is probably a function of time at temperature. If the rate of the crack-healing process can be described by an Arrhenius relation with an activation energy of 90 kcal/mol (an energy typical of creep and diffusional processes in UO_2 and mixed oxides), it follows that since Bain's experiment yielded a crack healing temperature of 1400°C for a period of steady reactor operation lasting about a month, a week of steady operation would yield a temperature of 1500°C and one day of steady operation, a temperature of 1650°C . The latter temperature is probably sufficiently high to negate the effect altogether. If the temperature for one month of operation is toward the lower limit of Bain's estimate, then the temperatures given should be lowered by 100°C . Since the proposed power increase effect is critically dependent upon the magnitude of the crack-healing temperature, and since this value is not known over a wide range of conditions, the absolute magnitude of this effect must presently be used with reservation.

12.2.2 Enhanced Neutron-Induced Metal Swelling

The immersion density and transmission electron microscopy examination results on the cladding and materials specimens of Types-316 and -347 stainless steel provide a basis for modeling the in-situ enhancement of metal swelling. Previously discussed in detail,^{12-4,12-5} the salient features of these data are:

1. The void number density is higher in stressed than in stress-free specimens;
2. Associated with the high void density is a high level of intragranular precipitation;

*Contributors are listed in Appendix B.

Figure 12-1a. Power-Time History Used to Investigate Power Increase Effect

Figure 12-1b. Permanent Cladding Diameter Increase Due to Plastic Strain versus Linear Power During Crack Healing

3. The incoherent precipitates can act as void nucleation sites;

4. The void density increase saturates.

From their modeling of accelerator-irradiated materials, Bullough and Perrin¹²⁻⁶ have concluded that, in the case of Type-316 stainless steel, the embryonic void density and the dislocation density apparently change with exposure. The former phenomenon is modeled in BEHAVE-I through the inclusion of a creep strain dependent term which effectively increases the nucleation site density (i.e., the embryonic void density) and thus enhances the metal swelling above that predicted by a data correlation equation (e.g., the PNL Theta Equation). This enhancement term has the following form:

$$1 + 100 (1 - e^{-A(\Delta D/D)_c})$$

where:

$$A = \text{a material-dependent constant, and}$$

$$\left(\frac{\Delta D}{D}\right)_c = \text{creep strain.}$$

Program logic determines the highest strain rate creep phenomenon in a given timestep and applies the corresponding $\Delta D/D$ to the above expression to enhance the metal swelling. Depending upon the strain rate, the creep strain used is either the sum of thermal and irradiation creep or is equal to the fuel-cladding mechanical interaction creep. No attempt is made to include a temperature dependency in this term since the data correlations available as options in BEHAVE-I each contain their own temperature variation and since these correlations were derived using data from both stressed and unstressed specimens. The results of applying this strain enhancement term to the BEHAVE-I modeling of Task F fuel pins F2G, F2H, and F2V are discussed in Section 12.3.

12.2.3 Center-Void Gas Retention

Quantitative evidence exists in support of the phenomenon of "gas bottling" of fission products within the center fuel void. Notley and MacEwan's direct pressure measurements¹²⁻⁷ showed that the release from UO_2 can occur in bursts, with discrete pressure increases occurring during reactor shutdown transients. Similar results were obtained by Burley and Freshley from vibrationally compacted mixed-oxide fuel pins.¹²⁻⁸ They noted that the stepwise pressure increases were directly related to the length of time of steady-state operation (up to 200 hours) prior to the reactor shutdown transient. In their analysis, Paulson and Springhorn¹²⁻⁹ have pointed out that the partial pressure of cesium at high temperatures can be 70 percent of the partial pressures of xenon and krypton.

Most of the precursors of cesium are noble gases that migrate up the temperature gradient and thus effectively brings cesium to the hot region of the fuel. With a critical temperature of approximately 2050°K , cesium behaves as an ideal gas at typical operating central void temperatures. The BEHAVE-I modeling assumes the same release behavior for cesium as for the noble gases, an assumption supported by experimental measurements¹²⁻¹⁰ on UO_2 . The central void pressure is computed in BEHAVE-I from the axial-node values of center void temperature and diameter. The fractional yield of Xe + Kr is 0.26 atoms/fission and the cesium yield is 0.184 atoms/fission. The hydrostatic pressure is assumed to act at the fuel plasticity radius, a boundary defined by the isotherm $T = 1650^\circ\text{C}$, and is transmitted to the cladding through the cooler structural fuel.

The possibility of central void containment of gaseous fission products has been discussed in conjunction with the post-irradiation analysis of EBR-II irradiated ANL Element Q12.¹²⁻¹¹ Somewhat similar fuel behavior was noted for Task F fuel pin F2V, where the fuel column length change was 2.5% as compared to a 1.8% change in the cladding length over the fueled length of the pin. Because of the large available void, no fuel-cladding mechanical interaction is predicted for F2V. However, the immersion density measurements show that enhancement of metal swelling occurred, indicating that a different clad-straining mechanism was operative in this pin. The details of applying the gas bottle effect in the BEHAVE-I modeling of F2G and F2V are given in Section 12.3.2.

12.3 FUEL ELEMENT BEHAVIORAL STUDIES

The BEHAVE-I model has undergone considerable revision since its initial use in the AEC-RDT sponsored Fuel Element Round Robin meeting. Of particular note are the additions of models to account for the in-situ increase in

metal swelling (Section 12.2.2) and to predict the effects of center-void containment of released gaseous fission products (Section 12.2.3). The modeling results are shown in Figures 12-2, 12-3, and 12-4 for F2H, F2V, and F2G, respectively. In each figure, the solid curve represents the post-irradiation outside diameter as measured by profilometry, and the dashed curves are determined by linear segments between the ten axial-node calculated values.

12.3.1 F2H

Fuel pin F2H was modeled using the power-time data in Table 12-1; irradiation was terminated with the removal of subassembly XG05. The calculated peak burnup is 6.71 at. % whereas 6.75 was determined by Nd-148 chemical analysis. A 73% release of noble fission gases is calculated and 67.5% is the measured release. The strain-enhanced metal swelling was calculated based upon the PNL Theta Equation using $A = 80$. This enhancement is in agreement with TEM analyses which indicate a higher void density in the F2H cladding than in unstressed Type-316 specimens at similar fluence and temperature. The calculated metal swelling and outside diameter change are compared with the immersion density and profilometry data in Figure 12-2.

12.3.2 F2V

The initial BEHAVE-I modeling of this low smear density Type-316 clad pin gave considerable disparity between the calculated and measured metal swelling. The only clad strain predicted was thermal and irradiation creep under an internal pressure determined by noble gases at the plenum temperature and volume.

To account for the enhanced metal swelling, the gas-bottle model has been invoked during the last two steady-state periods of the X050 irradiation (see Table 12-1). There is no quantitative data directly related to F2V which supports this modeling; there is, however, the qualitative observation (Section 12.2.3) that some mechanism moved the fuel axially inside the cladding.

Table 12-1
REACTOR POWER HISTORY FOR BEHAVE-I MODELING
OF FUEL PINS F2G, F2H AND F2V

Description	EBR-II Power (MW)	Comments
Startup		Strain predicted for F2H only
Steady State (8.7 days)	45	Subassembly XG05
Power Increase	45/45	XG05 rotated 180 degrees
Steady State (140.1 days)	45	
Steady State (26.2 days)	42.5	Reduced reactor power
Power Increase	42.5/45	Return to full power
Steady State (15 days)	45	
Steady State (2.7 days)	45	Core change stainless steel reflector installed
Power Increase	45/50	Reactor overpower
Steady State (2.7 days)	45	
Steady State (21.3 days)	30	Reduced reactor power
Power Increase	30/45	Return to full power
Steady State (45 days)	45	XG05 removed at the end of this run (F2H out)
Steady State (118.1 days)	50	Subassembly X050
Power Increase	50/52.5	
Steady State (11.4 days)	52.5	Reactor overpower
Power Increase	52.5/54.9	Reactor overpower
Steady State (54 days)	50	
Steady State (26 days)	50	No reactor shutdowns during this period
Steady State (24 days)	50	No reactor shutdowns during this period

Figure 12-2. BEHAVE-I Modeling Results for Fuel Pin F2H

The results are shown in Figure 12-3. The calculated (enhanced) metal swelling diameter change is not as large as that determined from the immersion density measurements but it is in better qualitative agreement than the un-enhanced PNL Theta Equation curve. The peak between eight and nine inches is due to a maximum in the sum of thermal and irradiation creep at that axial node. The over-prediction of the diameter change at the extremes of the fueled region is believed due to misrepresentation of the axial power shapes at various reactor power levels. This causes an overprediction of the power-increase strains at these end nodes.

12.3.3 F2G

For fuel pin F2G, the entire power-time data of Table 12-1 was used. The calculated peak burnup is 12.1 at. % compared to 10.84 at. % from the neodymium analysis. The burnup is calculated directly from the linear power, and the axial power and temperature profiles are determined from calculations using the fission density profiles in the EBR-II guide. The calculated fission gas release remains uncertain between 90 and 100 percent because of the difference in measured and calculated burnup.

Figure 12-4 shows the improved agreement between the measured and calculated metal swelling effects afforded by the strain enhanced metal swelling. For Type-347 cladding, the PNL "1.71 Equation" is used and the curves shown are for $A = 200$. This larger value for A is in qualitative agreement with the TEM results which showed more intragranular precipitation in Type-347 than in Type-316 stainless steel.

The gas-bottle effect was also invoked for this pin. However, since fuel-cladding mechanical interaction is predicted, the center void is believed to contain fission product debris; this perturbation is presently not accounted for in the volume/temperature calculations in the model.

Figure 12-3. BEHAVE-I Modeling Results for Fuel Pin F2V

Figure 12-4. BEHAVE-I Modeling Results for Fuel Pin F2G

12.4 FURTHER EXPERIMENTAL DATA IN SUPPORT OF CUMULATIVE STRAIN DAMAGE FAILURE CRITERION

The thirtieth Quarterly Report¹²⁻¹² presented a strain-oriented failure criterion. This approach, called the Cumulative Strain Damage Failure Criterion, provided a way of monitoring loading mechanisms that are strain rather than stress controlled. Ductility, which is a function of test conditions (i.e., strain rate, fluence, temperature, etc.), was defined as the controlling material property and it was used to normalize strain increments occurring under specific conditions. Thus, the ratio of the strain increment to the ductility for a given set of test conditions defines the fraction of life expended. The sum of several such ratios represents the cumulative life used and is defined as the "damage index." A damage index approaching 1.0 would indicate imminent failure. The Cumulative Strain Damage Failure Criterion is summarized by the following equation:

$$\text{Damage Index} = \sum_i \frac{\Delta \epsilon_i}{\epsilon_{iR}}$$

where:

$\Delta \epsilon_i$ = Strain Increment during conditions i, and

ϵ_{iR} = Rupture strain under conditions i.

Figure 12-5 of the thirty-sixth Quarterly Report¹²⁻¹³ shows this relation graphically under two specific strain rates (rapid tensile type and slow creep).

Two sets of data were presented in the thirty-eighth Quarterly Report¹²⁻¹⁴ which supported this theory. A third set of data¹²⁻¹⁵ has been analyzed.

The tests conducted at Northrop were sequential tests; a creep test at elevated temperature was followed by a tensile test to failure either at the same temperature or at room temperature. The resulting cumulative strain damage diagrams for Ti-6Al-4V in the annealed condition and Ti-8Al-1Mo-1V in the duplex annealed condition are shown in Figure 12-5. The shaded region indicates the uncertainty produced by variations in the normalizing values of creep and tensile ductility. Both materials were unirradiated.

These data, as well as that previously presented, show reasonable agreement with the proposed theory. It is disappointing that the data are not as well dispersed over the graph as were the previous data but the tests were designed to produce creep strains of the order of only 1%. It is very encouraging that so many points lie within the shaded region in view of the facts that both elevated and room temperature tensile tests were used in the test sequence and that strain rates during the creep portion ranged between 2×10^{-4} and 2×10^{-5} in./in. per hour.

The three points significantly below the shaded region for Ti-8Al-1Mo-1V demonstrate the effect of microstructural changes caused by a strain aging process. The loss in ductility was explained by fractography performed on the fracture surface of one of these specimens which indicated a brittle type quasi-cleavage fracture. Similar losses in ductility and poor correlation with theory were obtained for 2024-T3 specimens (not presented here). This latter material is well noted for both thermal and strain aging at elevated temperature. These deviations from the simplified cumulative strain damage failure criterion reinforce an earlier hypothesis that separate consideration may be necessary to distinguish between intergranular and transgranular failure modes and that special attention may be needed for materials that "age" with thermal exposure or strain. Fortunately, present cladding candidates are considered fairly stable in structure.

The limited data found to date support the Cumulative Strain Damage Failure Criterion for strain controlled deformation; consequently, it has been included as part of the latest recommended design criterion for the LMFB¹²⁻¹⁶.

Figure 12-5a. Cumulative Strain Damage Diagram, Sequential Creep-Tensile Tests (Unirradiated Ti-6Al-4V)

Figure 12-5b. Cumulative Strain Damage Diagram, Sequential Creep-Tensile Tests (Unirradiated Ti-8Al-1Mo-1V)

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APPENDIX A. PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Nos. 1 through 111 from July 1959 through August 1971.

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

GEAP-3888
First Quarterly Report
October–December 1961

GEAP-3957
Second Quarterly Report
January–March 1962

GEAP-3981
Third Quarterly Report
April–June 1962

GEAP-4080
Fourth Quarterly Report
July–September 1962

GEAP-4158
Fifth Quarterly Report
October–December 1962

GEAP-4214
Sixth Quarterly Report
January–March 1963

GEAP-4300
Seventh Quarterly Report
April–June 1963

GEAP-4382
Eighth Quarterly Report
July–September 1963

GEAP-4480
Ninth Quarterly Report
October–December 1963

GEAP-4601
Tenth Quarterly Report
January–March 1964

GEAP-4640
Eleventh Quarterly Report
April–June 1964

GEAP-4723
Twelfth Quarterly Report
July–September 1964

GEAP-4795
Thirteenth Quarterly Report
October 1964–January 1965

GEAP-4860
Fourteenth Quarterly Report
February–April 1965

GEAP-4931
Fifteenth Quarterly Report
May–July 1965

GEAP-4982
Sixteenth Quarterly Report
August–October 1965

GEAP-5098
Seventeenth Quarterly Report
November 1965–January 1966

GEAP-5158
Eighteenth Quarterly Report
February–April 1966

GEAP-5198
Nineteenth Quarterly Report
May–July 1966

GEAP-5292
Twentieth Quarterly Report
August–October 1966

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

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

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

GEAP-5541

Twenty-Fourth Quarterly Report
August–October 1967

GEAP-5584

Twenty-Fifth Quarterly Report
November 1967–January 1968

GEAP-5631

Twenty-Sixth Quarterly Report
February–April 1968

GEAP-5677

Twenty-Seventh Quarterly Report
May–July 1968

GEAP-5700

Twenty-Eighth Quarterly Report
August–October 1968

GEAP-5753

Twenty-Ninth Quarterly Report
November 1968–January 1969

GEAP-10028

Thirtieth Quarterly Report
February–April 1969

GEAP-10028-31

Thirty-First Quarterly Report
May–July 1969

GEAP-10028-32

Thirty-Second Quarterly Report
August–October 1969

GEAP-10028-33

Thirty-Third Quarterly Report
November 1969–January 1970

GEAP-10028-34

Thirty-Fourth Quarterly Report
February 1970–April 1970

GEAP-10028-35

Thirty-Fifth Quarterly
May 1970–July 1970

GEAP-10028-36

Thirty-Sixth Quarterly Report
August 1970–October 1970

GEAP-10028-37

Thirty-Seventh Quarterly Report
November–January 1971

GEAP-10028-38

Thirty-Eighth Quarterly Report
February–April 1971

GEAP-10028-39

Thirty-Ninth Quarterly Report
May–July 1971

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

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

In addition, the following topical reports have been
issued:

GEAP-3287

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

GEAP-3347

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

GEAP-3486

Fast Oxide Breeder Project
Fuel Fabrication

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

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

GEAP-3487

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

GEAP-3646

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

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Core Design Study for a 500 MWe
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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
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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;
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