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SODIUM-COOLED REACTORS FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

FORTY-FIRST QUARTERLY REPORT
NOVEMBER 1971-JANUARY 1972

U.S. ATOMIC ENERGY COMMISSION
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**SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM**

Forty-First Quarterly Report

November 1971—January 1972

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LIST OF ABBREVIATIONS, SYMBOLS, AND ACRONYMS

AEC-RDT	=	Atomic Energy Commission—Division of Reactor Development and Technology
ANL	=	Argonne National Laboratory
Ann.	=	annealed
Ann. V	=	annealed at VNC
ARC	=	Argonne Reactor Computation (System)
ASTM	=	American Society for Testing Materials
at.%	=	atom percent
bcd	=	binary coded decimal
BEHAVE	=	fuel modeling code
BOL	=	bulk outlet
BUBBLE	=	coolant boiling and channel voiding code, a subroutine of FREADM
co-ppt	=	coprecipitated
CSEWG	=	Cross Section Evaluation Working Group
C.W.	=	cold worked
DEMON	=	diffusion theory flux solution computer code
DFR	=	Dounreay Fast Reactor
DRL	=	Division of Reactor Licensing (AEC)
DTA	=	Differential Thermal Analysis
EBR-II	=	Experimental Breeder Reactor II
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

fiss/sec	=	Fissures per second
FRÉADM	=	Fast Reactor Excursion and Accident Dynamics Model
FUMBLE	=	fuel management computer code for startup-up cycles
GETR	=	General Electric Test Reactor
GMUG-2	=	Multigroup data file based on modified ENDF/B-II data
gpm	=	gallons per minute
GT	=	Garafolo Treated
GTV	=	Garafolo Treated at VNC
HCD	=	Hypothetical core disassembly
HEDL	=	Hanford Engineering Development Laboratory
h_{gap}	=	fuel cladding gap conductivity
i.d.	=	inside diameter
in./in./h	=	inches per inch per hour
I.P.	=	integrated power
kbar	=	thousands of atmospheres
k_{eff}	=	effective neutron multiplication factor
keV	=	kilo electron volts
ksi	=	thousands of pounds per square inch
kW/ft	=	kilowatts per foot
ℓ	=	angular momentum states
lb _f	=	pounds force
lb _m	=	pounds mass
LMFBR	=	Liquid Metal Fast Breeder Reactor
MARS	=	Computer code for core disassembly calculation
mCi	=	millicurie
MeV	=	million electron volts
MeV-GE	=	General Electric adjusted fluence for EBR-II ⁽¹⁾

mg	=	milligram
Mod	=	modified
mole	=	gram molecular weight
msec	=	millisecond
MWd/Te	=	megawatt-day per ton (metric, tonne, 2205 lb)
ND	=	Not determined
nyt	=	time integrated neutron flux
NOM	=	Nominal
o.d.	=	outside diameter
O/M	=	Oxygen-To-Metal ratio
%Δk	=	percent reactivity addition
%TD	=	percent of theoretical density of fuel
PEC	=	Pulsed Eddy Current (analyzer)
PISCES	=	Physics International multi-Spatial Code for Engineering and Science
P _L	=	linear power
P _{max}	=	peak power
PNL	=	Pacific Northwest Laboratory
Phys Mix	=	Physically Mixed
ppm	=	parts per million
PPS	=	plant protection system
psi	=	pounds per square inch
psia	=	pounds per square inch (absolute)
psig	=	pounds per square inch (gauge)
PUNT	=	service code for maintaining GMUG data files
R	=	Roentgen, unit of radiation dosage
RABBLE	=	Computer code for special resonance self-shielding homogeneity
SA or S/A	=	subassembly

S Ann.	=	Solution Annealed
SEFOR	=	Southwest Experimental Fast Oxide Reactor
S.o.L.	=	Start of Life
SS	=	stainless steel
(ST)	=	solution treated
Stab.	=	Stabilized
STP	=	standard temperature and pressure
SUR	=	strong unbroken ring
S _x	=	Standard deviation
SYN	=	combined synthesis-fuel cycle code system
TD or T.D.	=	Theoretical Density
TDOWN-2	=	computer code for cross-section processing, version 2
TEM	=	transmission electron microscopy
Theo	=	Theoretical
THTD	=	transient heat transfer code (version D)
THTE	=	transient heat transfer code (version E)
TOFF	=	transient overpower fuel failure model
TREAT	=	Transient Reactor Test Facility
TRIAL	=	Computer code for testing accuracy of Three-Dimensional (TRI) (A)xial (L)inking
UTS	=	Ultimate Tensile Strength
VENUS	=	disassembly calculation code
VERA-11	=	critical assembly 11A in VERA facility
VNC	=	Vallecitos Nuclear Center
VOIDUM	=	systems hydraulics model, a subroutine of FREADM
vol %	=	volume percent
W/cc	=	watts per cubic centimeter
wt %	=	weight percent

watts/cc	=	watts per cubic centimeter
XPORT	=	a computer routine to prepare code for shipment to outside users
ZEBRA-3	=	critical assembly 3 in ZEBRA facility
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 forty-first 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 November 1971 through January 1972 is described for tasks related to defected fuel testing, safety experimentation, fuel materials properties, fuel testing in EBR-II, core design analysis, reactor safety analysis, cladding development, physics, and fuel element design analysis.

HIGHLIGHTS

- The F10A and F10B high cladding temperature experiments began irradiation in EBR-II. These experiments each contain 19 encapsulated fuel rods, with Types-304, -316, -321, -347, and Incoloy 800 stainless steel cladding in several metallurgical states. The fuel rods will operate at peak conditions of ~ 16 kW/ft with $>1250^{\circ}\text{F}$ cladding temperature to >12 at.% target burnup.
- Two 37-rod grid assemblies continued operation at 1000°F in the Breeder Test Facility for a total of approximately 2200 hours at a maximum sodium velocity of 22.3 ft/second. Single-rod hydraulic tests showed that fuel rods supported by staggered grids at a 3-inch axial spacing vibrate in the same manner as rods which are simply supported at 3-inch intervals. Engineering support was provided for the fabrication of the EBR-II FTR diagnostic grid assembly F9F. A fuel rod comb support was designed for F9F that allows removal of each rod during examination by the EBR-II project.
- 593°C (1100°F) burst tests on Type-316 clad fuel rod sections (F3A) show that ductility is substantially reduced by intergranular attack on the inner surface. The burst strength, however, appears to be much less affected.
- The design of the first SEFOR capsule (TOP-1A) has been completed. The final design reviews have been reviewed and comments are being incorporated for the first SEFOR transient overpower capsule TOP-1A.
- An unencapsulated fuel rod reached 7.3 at.% burnup in EBR-II in excellent condition. The metallographic examination of F9C-31 is complete. This rod with 84% theoretical fuel density and clad in annealed Type 316, operated at ~ 13 kW/ft with $\sim 1025^{\circ}\text{F}$ cladding temperature. Attack at the cladding inside surface was negligible. Migration of plutonium and of the fission product Cesium was observed.
- Transmission electron microscopy (TEM) examination of cold worked Type-304 and Type-316 fuel rod cladding samples show that regions with a heavily cold worked microstructure exhibit little if any voids after irradiation to 3×10^{22} n/cm², $E > 0.1$ MeV-GE. Solution annealed samples of both alloys, however, exhibit substantial void populations after similar fluences.
- Capsule E2R, designed to investigate the time for central hole formation, has been irradiated a total of six hours. Central holes developed in the 88% T.D. pellets during the 11- to 26-minute exposure at 2500°C . No center holes have yet formed in the 94% T.D. pellets.
- Machining of the main components of the C4M capsule has been initiated. Quality control documentation through the Fabrication Specification and machining portion of the Shop Plan is approved, and machining of components of the C4M capsule has begun.
- High-temperature solution-annealing has been shown to have a beneficial effect on long-term mechanical properties of Type-321H tubing. Tests show that tubing processed with a 1148°C (2100°F) final anneal exhibits significantly superior stress rupture properties than similar tubing (same heat) with a 1065°C (1950°F) anneal.
- Fabrication of the second in the series of grid spaced fuel subassemblies for irradiation in EBR-II is nearing completion. EBR-II has completed review of the compatibility of F9F with the Mark S test vehicle concept for grid spacer irradiations. All the fuel rods have been loaded and welded. The procedure for Xenon tagging the fuel rods was approved by EBR-II.
- Out-of-pile thermal gradient tests of fission product behavior have shown that in a stoichiometric fuel environment, tellurium reacts intergranularly with Type-316 cladding. This element may participate more aggressively in cladding attack than previously assumed from electron microprobe results.

1. SUMMARY

1.1 TASK B—DEFECTED FUEL BEHAVIOR

Irradiation of capsule B9D was successfully completed. This capsule contained two mixed oxide fueled rods defected after ~ 3900 MWd/Te. and then irradiated for an additional ~ 3300 MWd/Te. Data on the changes in the activity and composition of the radionuclides released to the capsule gas plenum were obtained. Preparations were also made for post irradiation examination of the capsule.

Reviews of the capsule B9B Design Basis Specification and conceptual design were completed. Capsule B9B is to investigate the performance of mixed oxide fuel defected at high burnup and measure the rate of gas release from the plenum of a high burnup pin.

Testing of a heat pipe to remove the excess heat from a forced convection capsule in the event of pump failure was initiated.

Examination of two series of scoping tests to develop a self-actuating defecting mechanism for a fuel rod was completed. The results indicate that CsOH, if present in large concentrations, will promote clad attack and could be used to promote fuel rod deflection.

The Design Basis Specification for a series of capsules to determine the rate of fuel rod diameter changes caused by fuel sodium reactions and the volume changes associated with the reaction was completed and reviewed.

A report on the capabilities and feasibility of conducting run-beyond failure tests in U.S.A. facilities was completed.

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

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

Machining of the main C4M capsule components began in early January 1972 following BRD approval of the Fabrication Specification and machining portion of the Shop Plan.

Testing order has been reversed for the two HEDL-supplied EBR-II rods (PNL-2-2 solid fuel and PNL-2-7 annular fuel). The PNL-2-7 rod will be tested first and will be mounted in the C4M capsule.

Sections from C4G fuel column have been prepared for microprobe study of radial fissile density variations.

Preliminary design of the Series 6 C6C and C6D tests have been initiated. These capsules will each contain three EBR-II-type pre-irradiated fuel rods.

The SEFOR TOP-1A transient overpower capsule design has been modified to remove the $ZrH_{1.6}$ moderator from within the capsule primary containment. Final design reviews for this configuration have been completed, and incorporation of comments is in progress.

Task C-1 participated in a discussion on October 29, 1971 with ANL/HEDL/RDT on transient overpower test design and analysis methods and their objectives.

Task C-1 participated in a GE/AEC-RDT meeting in early December to update the planning of the SEFOR Option 1 test 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)

An analysis of the Mockup 1A test using the BUBBLE/VOIDUM model has been completed. Mockup 1B data evaluation is in progress. Computer algorithms to retrieve and plot Mockup 1B test data have been developed. The Design Basis Specification for capsule C11A has been issued and preliminary design is underway. Planning for a loss-of-flow test in the ANL Mark II-A loop is in progress.

1.4 TASK E—FUEL MATERIALS PROPERTIES AND PERFORMANCE EVALUATION

Out-of-pile experiments designed to investigate fission product behavior and reactions with cladding have operated over a 500 to 850°C temperature range, approaching expected LMFBR conditions. Gamma scans and pulsed eddy current tests indicate movement of Cs_2O to the cool end and cladding attack at these lower temperatures. Results from earlier experiments indicate that Cs may destroy the protective $\text{CrO}_{1.5}$ layers on stainless steel cladding.

Krypton impregnated wire and tubing have been obtained for investigating the Kr-85 release method of determining cladding temperature. A program to develop standards to investigate the "memory effect" and to utilize some of this material in in-pile tests is being implemented.

Electron microprobe examination of an irradiated stoichiometric mixed oxide fuel sample irradiated to 5.3 at.% indicated an increase in the $\text{Pu}/(\text{U} + \text{Pu})$ ratio from ~25 to 35% over the inner 10% of fuel annulus volume. Substoichiometric fuel irradiated to ~8 at.% burnup, however, showed essentially no change in this ratio suggesting that the plutonium migration may be related to oxygen.

A transducer-instrumented stainless steel vessel has been constructed to monitor fuel expansion upon exposure to hot sodium. The vessel is undergoing tests at high temperatures.

Out of pile thermal gradient testing has shown that free tellurium exceeding any Cs_2Te relationship reacts detrimentally with cladding grain boundaries in a high O/M (low $\Delta\bar{G}_{\text{O}_2}$) fuel environment.

1.5 TASK F—FAST FLUX TESTS AND SUPPORT

Irradiation of the F10A and F10B high cladding temperature experiments is underway. The encapsulated fuel rods are clad in SS304, SS316, SS321, SS347, and Incoloy 800 in several metallurgical states. They will operate at peak conditions of ~16 kW/ft with >1250°F cladding temperature to >12 at.% target burnup.

All the F9F rods are loaded and welded. EBR-II has completed review of the compatibility of F9F with the Mark S test vehicle concept for grid spacer irradiations. The procedure for xenon tagging the rods was approved by EBR-II. Fabrication of the grid spaced channel is in progress.

The metallographic examination of fuel rod F9C-31 is complete. This unencapsulated rod was in excellent condition after operating at ~13 kW/ft with ~1025°F cladding temperature to 7.3 at.% burnup. The rod was clad in annealed SS316, with fuel pellets of 88% theoretical density (84% smeared). Attack at the cladding inside surface was negligible. A large fuel-clad gap remained. There was evidence of plutonium and fission product cesium migration. Plutonium was concentrated near the central void and near the fuel-clad interface. A marked migration of cesium toward the bottom of the rod and a sharp concentration at the top of the rod were observed.

The non-destructive examination of the F4 capsules is complete. A preliminary report of the interim examination of the F3B experiment has been prepared for ANL and RDT. Reconstitution of the F9C fuel rods into two 19-rod Mark E run-to-failure subassemblies is complete, and the continued irradiation is underway. The data package for continued irradiation of the combined F3B and F5 experiments has been forwarded to EBR-II for review.

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

A topical report on the first GE EBR-II grid-spaced assembly F9E has been written, reviewed and publication is forthcoming.

Engineering support for the fabrication of the second grid-spaced EBR-II assembly F9F was provided. A fuel rod support comb that allows removal of each of the 37 rods by the EBR-II project was designed for F9F, and for a standardized grid assembly test vehicle.

Table 1-1
TASK B-IRRADIATIONS

Design Parameters													
Fuel									Operating Conditions			Status	
Capsule	Objective	Diameter (inch)	Length (inch)	Fuel-Cladding Gap (in.-diam)	Composition			Cladding Material	Irradiation Facility	Cladding Surface Temp (°F)	Peak Rating (kW/ft)	Peak Burnup (MWd/Te)	
					Density (Smear (% TD)	PuO ₂ (%)	O/M						
B8B (2 rods)	Fuel-sodium compatibility. Sodium-bonded fuel central temperature versus exposure time.	0.216	2	0.010	80	25	2.00	347	V-Raft GETR	800	14	14,500	Geap-13729 completed
		0.216	2	0.010	83	25	1.93	347					
B8C (2 rods)	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel- sodium-clad compatibility.	0.144	2.2	0.006	86	25	1.97	316	V-Raft GETR	1100	21	113,500 57,000	Examination completed
		0.204	2.25	0.017*	84	25	1.97	316					
B8D	Fuel-sodium compatibility versus density, stoichiometry and Pu content	0.214	1.75	0.0065	80-88	40	1.97	316	V-Raft GETR	950	17.5	41,000	Examination completed
		0.214	1.75	0.0065	80-88	25	1.97					43,500	
		0.214	1.75	0.0065	80-88	25	1.94					41,400	
B9A (2 rods) axially aligned)	Defected fuel rod. Fuel swelling versus stoichiometry in flowing sodium	0.216	5	0.005	89	25	1.99	316	Special V-Raft GETR	1019	23.4	8,600	GEAP-13730 released
		0.216	5	0.005	89	25	1.95			971	23.4	8,600	
E1J-B1	Thermal expansion clad defect mechanism	0.221	5	0.0007	94	25	2.00	316	Trail Cable	835	17.5	50	Examination completed
E1J-B2	Thermal expansion clad defect mechanism	0.221	5	0.0007	92	25	2.00	316	Trail Cable	860	19.0	300	Examination completed
B9C (2 rods)	Swelling behavior of defected fuel rods in flowing sodium	0.216	4.25	0.004	89.6	25	1.97	316	Special V-Raft GETR	1015	20.0	~7300**	Test terminated
		0.216	4.25	0.004	89.6	25	1.97	20% CW 316 Annealed		1030	20.0	~7300**	Test terminated
B9D (2 rods)	Two defected fuel rods. Fuel swelling in clean sodium (< 5 ppm Oxygen).	0.215	4.25	0.005	89.5	25	1.97	316 20% CW	Special V-Raft GETR	1100 1060	15.0	~5500**	Fabrication in process
B9B	Performance of rod defected at high burnup.	0.215	24.0	0.005	85.0	25	1.97	316 20% CW	Special V-Raft GETR	1150	16.0	~30,000	Design in progress.
B11A	Rate of diameter change of defected fuel rod.	0.219	5.0	0.001	87.0	25	2.00	316 30% CW	Trail Cable	1200	23.0	~ 5,000	Design in progress.
B11B	Rate of diameter change of defected fuel rod.	0.219	5.0	0.001	87.0	25	1.97	316 30% CW	Trail Cable	1200	23.0	~ 5,000	Design in progress.
B11C	Rate of diameter change of defected fuel rod.	0.219	3.0	0.001	87.0	25	1.97	316 30% CW	Trail Cable	1200	23.0	~ 5,000	Design in progress.

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

A meeting with HEDL and ARD personnel was held to discuss the reference FTR driver fuel assembly design as it relates to the GE grid-spaced FTR driver assembly design. This meeting established a basis for the preliminary engineering phase of the grid assembly design.

The impact of the proposed EBR-II reflector change on the orificing and core locations for F9E and F9F was assessed. In both cases the assembly orifices can be resized to obtain target outlet temperatures.

Approximately 2200 hours of successful operation were logged for the two 37-rod grid-spaced fuel assemblies at 1000°F and 22.3 ft/sec in the Breeder Test Facility sodium loop in San Jose. The loop pressure drop has been constant as expected.

The vibrational amplitudes of a single rod based on staggered grid supports at a 3-inch axial pitch were measured. These hydraulic test results are almost identical with the predictions of the Reavis correlation for a simply-supported rod with a 3-inch span. This measurement will be compared with full scale 217-rod hydraulic test bundle results. Fabrication of the 217-rod test section continues.

1.7 TASK G-10—SAFETY METHODS AND ANALYSIS

Engineering specifications are being written for a system of models to predict post failure events for subassemblies experiencing transient overpower fuel rod failure.

A safety study was completed which considered the impact of fission gas accommodation and plant protection system features on the frequency, and severity of accidents caused by fuel failure propagation. Vented fuel appeared to present potential safety advantages over plenum fuel for equal instrumentation.

Modifications to the VOIDUM/BUBBLE models for pipe break analysis are being revised.

Debugging of the FREADM-VENUS interface is 95% completed.

The TOFF-II routines have been revised to allow multiple bundle types before continuing with the coupling to the FREADM program.

An exploratory analysis has been performed using the PISCES IDL hydrodynamics program to analyze conditions in a sodium filled channel during a hypothetical fuel coolant interaction. The analysis has demonstrated the utility of this modeling approach, although more sophisticated modeling will be required to produce useable results.

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

Assignment of a GE engineer at the ANL on a full-time basis in a liaison capacity, has proved to be effective in facilitating the interchange of technical information between ANL and GE-BRD. In addition to informal communications, some 42 documents were exchanged during the quarter.

1.9 TASK L—CLADDING DEVELOPMENT

The post-irradiation examination of the three L-6 capsules has been completed. (Total fluence of 8.3×10^{22} n/cm², $E > 0.1$ MeV-GE.) Further work on these specimens will include measurements of clad alloy swelling (solution annealed Types 304, 316, 321, 347 and Incoloy-800).

Pre-irradiation data for all of the specimens selected for irradiation in the L-16D capsules have been obtained. This included: dimensional measurements, density determinations, metallography and transmission and scanning electron microscopy examinations. In general, the previously irradiated F2 and L2 clad specimens exhibited similar decreases in density before and after removal of rings for elastic modulus measurements. Some of the sodium exposed specimens exhibited measurable decreases in density as a consequence of exposure to high temperature flowing sodium. Subsurface porosity was observed in the specimens which had decreased in density.

TEM examination of various Task F fuel rod cladding specimens continued. Results show that 50% cold-worked Type 316 was essentially free of voids after irradiation to 3×10^{22} n/cm², $E > 0.1$ MeV-GE at 526°C (979°F). Similar results on 13% cold worked Type 304 were also obtained. Areas exhibiting high dislocation densities were free of voids.

A variety of Type 316 solution annealed clad specimens were also examined. The void size increased with increasing temperature of irradiation similar to that observed earlier for annealed Type 304. The void number density decreased with increasing irradiation temperature.

Three thousand feet of Type 321H air-melted tubing is currently being evaluated. Both chemical analysis and dimensional measurements were completed. Mechanical property measurements have been initiated, including tensile, burst, and stress-rupture tests. Initial stress rupture results show that tubing processed with a 1148°C (2100°F) final anneal exhibits superior stress rupture properties than similar tubing (same heat) with a 1065°C (1950°F) final anneal. An order for 1500 feet of vacuum melted Type 321H tubing has been placed. The vendor will supply tubing (processed with a 1204°C (2200°F) solution anneal as well as tubing with a 1093°C (2000°F) anneal to further evaluate the effects of heat treatment on the resulting properties of Type 321H tubing.

Burst testing of the F9C-31 fuel rod clad with Type 316 was completed at 593°C (1100°F), as was the metallographic examination of burst specimens from the F3A fuel rods. Metallographic examination revealed up to 7 mils of attack of the inner surface of F3A burst test specimens whereas <0.5 mil of attack was found on the F9C-31 rod sections. Comparing the burst test results from both fuel rods shows that the major effect of the observed intergranular attack is the existence of notches rather than an overall wall thinning effect. The ductility of specimens with intergranular attack is substantially reduced. However, the burst strength appears to be much less affected. The reduced ductility is a consequence of the notches rather than wall thinning.

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

1.10 TASK P—FUEL ELEMENT MODELING

The first draft of the documentation of the BEHAVE-2 fuel rod performance code is being reviewed. The BEHAVE-2 code has as its most prominent features models of fuel cracking under applied tensile stresses, two dimensionality of the calculated displacement field, fuel densification due both to diffusion-rate controlled and stress-rate controlled processes, and assumed healing of cracks in the zone of rapid diffusion. The code calculates in great detail a number of cracking, plastic flow and restructuring processes. On the other hand, those processes involving the disposition of fission gases, including fission gas induced fuel swelling and fission gas release rates, are presently treated in a less detailed manner due to the lack of definitive models describing these processes.

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 loaded powder fuel is to maximize fuel-sodium heat transfer (transient magnitude will be varied).	0.212	24	86	80	0	10.0	2.000	316SS
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>	TREAT		Examinations [†]			Report
			Calibration	Test	Nondestructive		Destructive	
					Before Transient	After Transient		
C4A C4B C4C C4D C4I C4J								GEAP-13562
C4E C4F	C C	C* C*	P P	P P	C C	P P	P P	
C4G C4H	C C	C* C*	C C	C C	C C	C 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 C5R								GEAP-13543
C6A-1 C6A-2 C6A-3 C6B-1 C6B-2 C6B-3								GEAP-13722

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

Group	Objective	Total Number of Rods in Group	Date In	Projected Date Out	Actual Date Out	Number of Rods in S/A	S/A	Position
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	4/30/72		2	XG03A	8D2
			9/14/70	3/30/73		2	XG04A	8A7
F1	Short time performance and proof test	6	5/ 6/65		5/23/65	6	XG01	4F2
F2	Medium power scoping irradiation. 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 rod irradiation to evaluate fuel cladding and cladding-coolant interaction	16	8/15/68		6/30/69	16	X040	5B2
			10/ 4/69		5/18/70	16	X040A	5B2
F3B	Density and stoichiometry effects for encapsulated fuel rods	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
						19	X151	
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	9/10/72		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	6/25/75		9	X117	8B2
			2/24/71	5/ 8/74		10	X118	8D7

Range of Average Rod Powers ^(b) kW/ft	Target Burnup at. % MWd/Te		Average Calculated Burnup ^(c) at. % MWd/Te		Peak Calculated Fluence ^(d) (10^{22} n/cm ²)		Status
					Total	E > 0.1 MeV	
7.76	3.91	35000	3.34	30199	2.76	1.99	Post exam complete
5.47– 5.92	8.95	80000	6.88	62247	5.93	4.32	Reconstituted into XG03A
5.44– 6.24	11.18	100000	7.62	68916	6.62	4.90	Reconstituted into XG04A
5.52– 6.16	8.95	80000	8.36	75597	7.31	5.30	In reactor
5.37– 6.12	11.18	100000	9.07	82061	8.01	5.39	In reactor
12.32–14.49	0.22	2000	0.21	1882	0.17	0.15	Reported in GEAP-5570
11.42–13.91	8.39	75000	6.69	60543	5.69	4.88	Post exam complete
8.48–14.83	5.59	50000	5.02	45451	4.22	3.62	Reported in GEAP-13549
10.73–12.86	11.19	100000	11.75	106358	10.34	8.87	Post exam in progress
12.87–14.65	5.62	50000	2.88	25959	2.36	2.00	Reconstituted into X040A
12.53–14.25	5.62	50000	5.17	46620	4.30	3.65	Reported in GEAP-13735
11.48–12.80	13.49	120000	8.86	79922	7.54	6.47	Eight rods reconstituted into X151
13.74–14.53	5.65	50000	3.64	32758	2.98	2.57	Reconstituted into X039
13.41–13.83	9.06	80000	4.85	43633	4.00	3.44	Reconstituted into X070
12.72–13.37	9.06	80000	8.25	74217	6.95	5.97	Post exam, two rods reconstituted into F20
11.63–13.75	11.23	100000	5.76	51970	4.80	4.12	Eleven rods reconstituted into X151 Out of reactor
8.59– 9.34	5.59	50000	5.71	51645	4.80	3.91	Post exam complete
12.54–12.65	5.64	50000	5.40	48599	4.56	3.90	Transient tests in TREAT
11.91–12.25	5.63	50000	5.43	48914	4.49	3.84	Transient tests in TREAT
8.53– 9.15	5.62	50000	4.97	44777	4.12	3.42	Post exam complete
8.37– 9.11	11.25	100000	6.15	55398	5.24	4.38	Reconstituted into X081
6.69– 8.33	11.24	100000	9.31	83935	8.03	6.49	In reactor
4.99– 6.39	11.24	100000	4.73	42600	4.01	3.03	Reconstituted into X117 and X118
4.33– 5.52	11.25	100000	5.42	48815	4.67	3.41	In reactor
4.99– 6.20	11.24	100000	6.20	55889	5.33	3.98	In reactor

Table 1-3 (Continued)

Group	Objective	Total Number of Rods in Group	Date In	Projected Date Out	Actual Date Out	Number of Rods in S/A	S/A	Position
F9A	High power, high burnup, unencapsulated fuel rod irradiations	37	2/20/69		5/18/70	37	X043	4D2
			10/27/70	2/18/73		37	X043A	7D4
F9B	Medium power, high burnup, unencapsulated fuel rod irradiations	37	5/23/69	2/16/73		37	X062	7F3
F9C	High power, high burnup, unencapsulated fuel rod irradiations	37	4/ 3/69 10/27/70		7/16/70	37	X056	5C2
					5/13/71	37	X056A	6B2
						19	X143	
						19	X144	
F9D	Medium power, high burnup, unencapsulated fuel rod irradiations	37	4/24/69	6/25/73		37	X058	7B6
F9E	High power, high burnup, unencapsulated fuel rod irradiations in a grid spacer subassembly	37				37	X145	
F9F	High power, high burnup, unencapsulated fuel rod irradiations in a grid spacer subassembly	37				37	X155	
F10A	Cladding performance at peak cladding temperatures	19				19	X121	
F10B	Cladding performance at peak cladding temperatures	19	12/5/71	6/7/74		19	X122	6D4
F10C	Fuel rod performance at peak cladding temperatures	19						
F11A	Cladding performance at peak FCR cladding temperatures	37				37	X141	
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 rod in the subassembly with the highest fuel burnup regardless of position in subassembly. All values are corrected for fuel depletion. Burnup values in MWd/t are based on a value of 192 MeV/Fission and calculated from data supplied by EBR-II.
- (d) Fluence values reported are calculated for the peak axial flux for the rod in the subassembly located closest to the center of the reactor core. Values are based on data supplied by EBR-II.

Range of Average Rod Powers ^(b) kW/ft	Target Burnup at. %	Burnup MWd/Te	Average Calculated Burnup ^(c)		Peak Calculated Fluence ^(d) (10 ²² n/cm ²)		Status
			at. %	MWd/Te	Total	E > 0.1 MeV	
12.06–13.36	11.25	100000	5.57	50189	4.64	3.98	Reconstituted into X043A
8.08–10.40	11.25	100000	7.40	66682	6.27	5.74	In reactor
7.91–10.34	11.24	100000	7.47	67321	6.33	05.09	In reactor
10.50–11.55	11.24	100000	5.67	51157	4.72	4.01	Reconstituted into X056A
10.20–11.71	11.24	100000	6.68	60207	5.61	4.76	Reconstituted into X143 and X144
							Out of Reactor
							Out of Reactor
6.91–9.42	11.24	100000	6.91	62266	5.89	4.74	In reactor
							Shipped to EBR-II
							Fabrication in progress
							Shipped to EBR-II
12.14–14.29	11.25	100000	0.59	5311	0.48	0.41	In reactor
							Design in progress
							Shipped to EBR-II
							Fabrication in progress

Table 1-4

SUMMARY OF TASK-L CAPSULES IN EBR-II

Group	Purpose & Major Parameters	Total Rods 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 & Incoloy 800	4	None	288	08/15/68 08/24/70 10/20/71	12/24/68 ⁽²⁾ 06/10/71 ⁽²⁾ 01/00/73 ⁽¹⁾
MT-4 ⁽⁴⁾	P.I. Mech. Prop. Eval. of Incoloy-800, Inconel-625, & Hastelloy-X	1	None	48	12/13/65 08/24/70 10/20/71	06/26/69 ⁽²⁾ 06/10/71 ⁽²⁾ 01/00/73 ⁽¹⁾
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	11/16/70	01/00/76 ⁽¹⁾
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 10^{22} n/cm ² Total	S/A Average Fluences ⁽³⁾ 10^{22} n/cm ² Total >0.1MeV		Status
4-XG06	4E2	9,317	4.0	3.3	2.9	Report complete
4-X009	4A2	5,355	2.0	1.9	1.6	Report complete
2-X014	2D1	3,674	2.0	1.6	1.4	Report complete
4-X016	4D3 & 4F3	7,894	2.0	2.8	2.4	Report complete
4-X025	4E2	8,399	2.0	2.9	2.5	Post exam complete
4-XG05	4C2	12,632	7.5	5.4	4.6	Post exam in progress
3-X039	3C2	2,532				
4-XG05	4C2	12,632	7.5	9.0	7.5	Post exam in progress
4-X050	4C2	11,706				
4-XA08	4F2	19,280	10.0	6.4	5.5	Post exam complete
2-X018	2B1	10,853	20.0	8.3	7.1	Post exam complete
4-X018A	4E2	2,334				
4-X018B	4E2	10,351				
3-X039	3C2	2,532	3.0	2.9	2.5	In reactor
4-X080	6E5	5,351				
6-X148	6B3	3,100				
4-XA08	4F2	19,280	20.0	8.4	7.3	In reactor
6-X080	6E5	5,351				
6-X148	6B3	3,100				
4	4C2	—	0.1			Fabrication in progress
4-X098	4F2	9,155	3.0	2.9	2.5	In reactor
—	—	—				Design in progress
—	—	—		9.0 ⁽⁵⁾	7.6 ⁽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 rods 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 rod under normal or near normal LMFBR operating conditions, 2) provide experimental and analytical data to permit selection of fuel rod design parameters and operating conditions which limit the consequences of fuel failure, 3) provide experimental data on failure kinetics to permit modeling and prediction of fuel failure behavior for the projected range of LMFBR operating conditions, and 4) identify the fuel and fission-product species distribution in the reactor coolant and cover gas as a function of the failure mode.

2.2 FORCED CONVECTION CAPSULES

2.2.1 B9B Capsule

Capsule B9B is designed to investigate the performance of a defected mixed oxide fuel rod operated beyond failure in contact with high purity (<10 ppm O_2) flowing sodium. The capsule will contain a single mixed oxide fuel rod which will be defected at high burnup ($\sim 25,000$ MWd/Te). After defecting, the rod will be operated for approximately one month to determine the effect of continued operation of a defected fuel rod on the dimensional change of the rod and the release of fuel and fission products from the rod. The fuel rod and test operating parameters are listed in Table 2-1.

Two pressure transducers will be connected to the fuel rod plenum to measure the rate of release of fission gas from the rod after defecting.

During this quarter, the Design Basis Specification was completed and reviewed and a design concept was drafted. As a result of the success obtained with the Fission Gas Detection System (FDS) in Capsule B9D**, specifications for a similar but more sophisticated design were prepared for capsule B9B. This system will allow the cover gas in the plenum to be monitored for gamma activity and samples of the cover gas to be taken for gamma spectroscopy analysis.

A review of the feasibility of incorporating the Fission Gas Analysis System (FGAS) into the B9B capsule was initiated to ensure that it would not compromise the primary objective of the experiment.

The final draft of the B9B Design Basis Specification was initiated.

2.2.2 B9D Capsule

The purpose of the B9D capsule experiment is to investigate the performance of mixed oxide fuel rods defected in high purity (<5 ppm O_2) high temperature (1050 to $1100^\circ F$) sodium and then operated several days after defecting. The capsule contained two mixed oxide fuel rods which were axially aligned. The fuel rods are shown in Figure 2-1. The top rod (B9D-1) was defected in the plenum and the bottom rod (B9D-2) in the center of the fueled region. The rods were defected by pulling on a wire attached to notched plugs welded over 0.030 inch diameter holes in the fuel rod cladding. The notched plugs and their location in the fuel rods are shown in Figure 2-1.

Fabrication and irradiation of capsule B9D were successfully completed. All instrumentation in the capsule operated without failure or evidence of degradation throughout the irradiation.

* Contributors are listed in Appendix B.

** See Subsection 2.2.2

Table 2-1
B9B CAPSULE FUEL AND OPERATING PARAMETERS

Fuel

Material	Coprecipitated ($\text{Pu}_{0.25}\text{U}_{0.75}\text{O}_{1.970}$)
Form	Solid Pellets
Pellet Density	$89 \pm 2\%$ T.D.
U-235 Enrichment	$\geq 90\%$
O/M	1.97 ± 0.015
Column Length	24.0 ± 0.125 inch
Pellet Diameter	0.2150 ± 0.0010 inch
Cladding i.d.	0.220 ± 0.002 inch
Cladding o.d.	0.225 ± 0.001 inch
Cladding Material	316—20% C.W.
Back Fill Gas	Helium
Blanket Length	15 inches
Insulator Length	0.250 inch

Operation

Average Linear Power	16 ± 2 kW/ft
Sodium Inlet Temperature	1025°F
Sodium Outlet Temperature	1150°F (maximum)
Coolant Velocity	22.5 ± 2.5 ft/sec
Burnup at Defection	$\sim 25,000$ MWd/Te
Final Burnup	$\sim 30,000$ MWd/Te
Days of Operation After Defection	~ 30
Plenum Pressure at Time of Defection	~ 175 psi

Figure 2-1. B9D Capsule Fuel Rods

The fuel and test operating parameters are listed in Table 2-2.

Fabrication and sodium filling of the capsule were completed at the beginning of November 1971. Verification of the purity of sodium in the capsule was accomplished by three techniques: 1) determination of the plugging temperature of the sodium in the filling loop, 2) analysis of a sodium sample removed from the loop, and 3) vacuum fusion analysis of a niobium wire removed from the loop. The results from each method of analysis are listed in Table 2-3. The niobium wire analysis is judged to be the most accurate determination of O_2 content in the capsule sodium. After sodium filling the capsule was maintained at $600^\circ F$ by storing in a portable furnace.

Prior to transporting the capsule from the sodium filling station to the reactor for insertion, a "dry run" of the transfer and insertion was carried out using a dummy capsule. Experience gained during the "dry run" enabled the actual insertion to proceed as planned. The capsule is shown in the portable furnace during the transfer from the sodium filling station to the reactor in Figure 2-2. During the transfer, electrical power was provided to the furnace by the portable generator on the truck bed (see Figure 2-2).

Figure 2-3 shows the capsule after completing the installation into the pump stator and turning on the pump cooling water, which can be seen flowing from the bottom of the pump stator. The capsule was then lowered down into the reactor pool and secured to the wall of the pool next to the reactor pressure vessel.

Irradiation of the capsule was begun on November 15, 1971. The fuel rods were operated at 20 kW/ft for approximately six days (150 hours) to promote fuel restructuring. The power was then lowered to 15 kW/ft and defecting of the fuel rods initiated on November 22, 1971. Based on test data obtained from mockups of the defecting system, loads greater than expected were required to pull the defecting wire. A thorough review of the fabrication history and design of the defect system was held to determine the cause of the increase in force required to pull the defect wire. It was decided that the increased force was the result of increased friction in the bellows assembly through which the defect wire passed. Analyses showed that the magnitude of the increased force required to pull the defect wire would not result in damage to the capsule. It was therefore decided to continue the defecting of the rods. Defecting of both fuel rods was completed on November 30, 1971. At the time of defecting, the burnup on the fuel rods was calculated to be 3900 MWd/Te. Irradiation of the rods in the failed state continued for 15 days. Total burnup was calculated to be 7200 MWd/Te at the end of the irradiation.

A summary of the capsule power and the inlet and outlet sodium temperatures is shown in Figure 2-4. Starting on December 1, 1971, a downward drift in the sodium inlet and outlet temperatures was observed. The temperatures continued to decrease until December 9, 1971, after which little change was observed. The cause of this behavior is presently unknown.

The capsule design included a system for sampling the cover gas in the plenum above the sodium in the capsule. The original purpose of this system was to look for the isotopes Ar-36 and Ar-40 which were used to tag the top and bottom fuel rods, respectively, to confirm defection. However, additional samples were taken periodically during the irradiation and the levels of the noble gas radionuclides assayed by gamma spectroscopy. The times at which the samples were taken are indicated by the numbered circles on the power history plotted in Figure 2-4. Results of the gamma spectroscopy analysis are listed in Table 2-4. Samples 1, 2 and 3 were obtained prior to defecting the fuel rods. These samples should contain an activity level corresponding only to the background emitted as recoils from the exposed surfaces of the metallic uranium getter in the capsule.* The calculated recoil fission rate for the exposed surfaces of the getter is 2.26×10^{10} fiss/sec. Fission rates calculated from the measured activities of samples 1, 2 and 3 compare quite well with the calculated recoil fission rate for the getter. The values listed in Table 2-4 range from 0.78 to 4.34×10^{10} fiss/sec. It is concluded from these data that the fuel rods had not defected prior to taking sample 3. Spectrographic analysis of the gas samples confirmed this conclusion since the fuel rod tag gases Ar-36 and Ar-40 were not detected.

Measured activities and calculated fission source rates** for samples 1 through 14 are listed in Table 2-4. The increased fission source rates for samples 4 and 5 compared to samples 1, 2, and 3 are attributed to gas release from the

* The purpose of the uranium getter was to maintain the oxygen level in the sodium below 5 ppm during the irradiation.

** $F \text{ (fiss/sec)} = A(\text{mCi}) \times 3.7 \times 10^7 / y_i$

Figure 2-2. Transport of Capsule B9D to GETR

Figure 2-3. Capsule B9D After Installation in Irradiation Facility

Figure 2-4. Summary of Capsule B9D Operating Conditions

Table 2-2
B9D CAPSULE FUEL AND OPERATING PARAMETERS

FUEL

Material	Coprecipitated ($\text{Pu}_{0.25}\text{U}_{0.75}\text{O}_{1.970}$)
Form	Solid Pellets
Pellet Density	$94 \pm 2\%$
U-235 Enrichment	$>90\%$
O/M	1.97 ± 0.010
Column Length	4.25 ± 0.125 inch
Pellet Diameter	0.215 ± 0.001 inch
Clad O.D.	0.250 ± 0.001 inch
Clad Material	316 — 20% cw
Gas Plenum Volume	0.6 cc
Gas Plenum Location	Above fuel in top rod Below fuel in bottom rod
Back Fill Gas	Top rod — He + Ar — 36 tag, Bottom rod — He + Ar — 40 tag
Blanket and Insulator	
Material	UO_2
Length	0.125 inch
Diameter	0.215 ± 0.001 inch

OPERATION^(a)

Average Linear Power	20 kW/ft before defection. 15 kW/ft during and after defection.
Sodium Inlet Temp	1000 - 1050°F
Sodium Outlet Temp	1040 - 1090°F
Coolant Velocity	22.5 ± 2.5 ft/sec
Fuel Burnup ^(b)	~3900 MWd/Te before defection. ~7200 MWd/Te total.
Duration of Irradiation	30 days
O ₂ Level in Sodium	<5 ppm (pre-irradiation)

(a) B9D parameters were chosen to reflect FFTF design conditions.
The steady-state operation of B9D simulates the peak fuel surface temperature in the FFTF hot channel at overpower.

(b) Calculated.

Table 2-3
ANALYSIS OF SODIUM IN B9D CAPSULE

Method of Analysis	O ₂ Concentration (ppm)
Sodium Plugging Temperature	<0.5
Chemical Analysis	8 ± 1.0*
Niobium Wire	<0.5 ± 1.0

* Value includes CO₃ and OH as well as O₂.

top fuel pin (B9D-1). The fission source levels for samples 6 through 14 are considered to include additional releases from the defect in the lower fuel rod (B9D-2).

Preparations were made for post-irradiation examination of the capsule.

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.

A prototype of the radial heat pipe to be used in capsule B9B has been fabricated and testing initiated. Testing will consist of determining the evaporation rate of the working fluid, response times, operating characteristics under overpower conditions, and the effect of gas overpressure on the heat transfer rate of the heat pipe. The first test at low power indicated that design calculations of the evaporation rate of the working fluid (lithium) versus temperature were accurate to within a few percent. A cartridge heater is being used to simulate a fuel rod during the testing. Additional lithium is being added to the heat pipe for the high power phase of testing. Figure 2-5 is a reproduction of the fabrication assembly drawing of the prototype. Figure 2-6 shows the heat pipe as fabricated.

2.3.2 Defecting Mechanism Development

Examination was completed of the first two series of scoping tests for development of a self-actuating fuel rod clad defecting mechanism for in-pile tests. These first two series of test capsules contained UO₂, and potentially corrosive chemicals (CsOH, AgF, and CsMoO₄) clad in Type-316 stainless steel. The two series of twelve capsules each were exposed to an axial temperature gradient over their 24-inch length with the UO₂ and chemicals placed in the capsules at temperatures of 800°F and 900°F, respectively, for 500 hours. The temperature gradients in each of the capsule series are plotted in Figures 2-7 and 2-8.

The quantities of chemicals loaded into each series of capsules and the sectioning diagrams for the capsules are given in Table 2-5 and Figure 2-9, respectively.

Metallographic examination of the sections cut from the capsules indicates that:

- Cladding attack occurred only in capsules which contained CsOH.
- Cladding attack was observed in higher temperature regions of the capsules away from the region where the CsOH was initially placed in the capsule.
- The depth of cladding attack increased with cladding temperature. Maximum attack observed at 800°F was 0.0007 inch. At 900°F and 1100°F the maximum depth of attack was 0.001 inch and 0.003 inch respectively.
- The amount of attack decreased with decreasing CsOH concentration.

Table 2-4
MEASURED ACTIVITIES AND RELATED SOURCE LEVELS IN B9D COVER GAS

SAMPLE NUMBER	1	2	3	4	5	6	7
Sample Time (hours)	1340	1510	1538	0927	1028	1456	1418
Sample Date	11/22	11/22	11/22	11/30	11/30	11/30	12/2
Activity in Cover Gas (mCi)							
5.29d Xe-133	68.4	51.4	27.5	42600	27400	45400	73800
2.26d Xe-133m	1.37	1.10	0.63	1340	778	1300	1710
9.16h Xe-135	58.5	47.6	29.4	2040	1500	4330	8370
4.4h Kr-85m	15.3	12.7	9.0	183	112	498	269
2.79h Kr-88	20.6	19.6	12.3	121	107	725	247
76m Kr-87	13.1	16.5	9.2		53	446	129
15.7m Xe-135m			2.9			570	788
14.2m Xe-138			12.4			<364	<163
Source Level (fiss/sec X 10 ⁻¹⁰)							
5.29d Xe-133	3.78	2.84	1.52	2356	1515	2511	4082
2.26d Xe-133m	3.17	2.54	1.46	3099	1799	3006	3954
9.16h Xe-135	3.44	2.80	1.73	120	88.1	254	492
4.4h Kr-85m	4.35	3.61	2.56	52.1	31.9	142	76.6
2.79h Kr-88	2.14	2.04	1.28	12.6	11.1	75.4	25.7
76m Kr-87	1.92	2.41	1.35	5.3	7.75	65.2	18.9
15.7m Xe-135m			1.07		3.6	211	292
14.2m Xe-138			0.78			<22.8	<10.2
SAMPLE NUMBER	8	9	10	11	12	13	14
Sample Time (hours)	1559	0854	1000	0755	0852	1404	0918
Sample Date	12/2	12/4	12/4	12/15	12/15	12/15	12/16
Activity in Cover Gas (mCi)							
5.29d Xe-133	80000	69100	61700	15000	12500	11500	8680
2.26d Xe-133m	1930	947	911	225	200	167	121
9.16h Xe-135	8830	809	668	1110	867	588	250
4.4h Kr-85m	187	8.58	6.19	90.1	65.1	33.1	10.8
2.79h Kr-88	211	3.41	3.93	107	74.7	20.3	4.03
76m Kr-87	89	5.11	12.7	88.8	49.0	5.91	3.52
15.7m Xe-135m	320	10.6	21.1	142	95.3	28.5	7.46
14.2m Xe-138	<82			118	<42.8		
Source Level (fiss/sec X 10 ⁻¹⁰)							
5.29d Xe-133	4425	3822	3412	830	691	636	480
2.26d Xe-133m	4463	2190	2107	520	463	386	280
9.16 Xe-135	519	47.5	39.2	65.2	50.9	34.5	14.7
4.4h Kr-85m	81.7	2.44	1.76	25.6	18.5	9.42	3.07
2.79h Kr-88	21.9	0.35	0.41	11.1	7.76	2.11	0.42
76m Kr-87	13.0	0.75	1.86	13.0	7.17	0.86	0.51
15.7m Xe-135m	118	3.92	7.81	52.5	35.3	10.6	2.76
14.2 Xe-138	<5.14			7.4	<2.68		

NOTE: Calculated fission rate at 15 kW/ft = 3.30×10^{-14} fiss/sec.

Calculated recoil source at 20 kW/ft operation = 1.13×10^{10} fiss/sec per foil side.

2-11

Figure 2-6. As Fabricated Heat Pipe Prototype

Figure 2-7. Thermal Gradient in Test Capsules with UO_2 and Chemicals at 800°F

Figure 2-8. Thermal Gradient in Test Capsules with UO_2 and Chemicals at 900°F

Table 2-5
SUMMARY OF CORROSION CAPSULE
DESIGN AND OPERATING PARAMETERS

Weight of UO_2 in Each Capsule		15 grams
Capsule Number	Corroding Species	Amount of Corrodant
1	Silver Fluoride (AgF)	0.0035g
2	Silver Fluoride (AgF)	0.035 g
3	Silver Fluoride (AgF)	0.35 g
4	Cesium Hydroxide (CsOH)	0.009 g
5	Cesium Hydroxide (CsOH)	0.071 g
6	Cesium Hydroxide (CsOH)	0.400 g
7	Cesium Molybdate ($\text{Cs}_2 \text{MoO}_4$)	0.013 g
8	Cesium Molybdate ($\text{Cs}_2 \text{MoO}_4$)	0.092 g
9	Cesium Molybdate ($\text{Cs}_2 \text{MoO}_4$)	0.600 g
10	Cesium Molybdate in Molybdenum Container	0.013 g
11	Cesium Molybdate in Molybdenum Container	0.092 g
12	Cesium Molybdate in Molybdenum Container	0.330 g
Furnace Time		505 Hours
Cladding Material		Type-316 Stainless Steel

Of all the specimens examined at 800, 900, and 1100°F, those which contained the largest quantity of CsOH are shown in Figure 2-10.

Based upon the results of these tests, it appears that a self-actuating defect mechanism could be developed using CsOH, but localized thinning of the rod cladding would be necessary in order to assure that the defect occurred at the desired location in a reasonable period of time.

2.4 FUEL-SODIUM REACTION STUDIES

The B11 series of experiments is designed to determine:

- 1) The rate of fuel rod diameter change resulting from sodium-fuel reactions.
- 2) The effect of initial fuel O/M on the rate law for reaction between sodium and fuel and the magnitude of fuel rod diameter changes resulting from sodium fuel reactions.
- 3) The reproducibility with which sodium fuel reaction rates can be determined by measurement of cladding diametral changes.

The rod parameters for the B11A, -B, -C capsules are listed in Table 2-6. Each fuel rod will contain a notch in the cladding. Differential expansion of the fuel and cladding coupled with the small fuel-clad gap will cause rupture of the cladding in the notched area. An oxygen getter will be placed in the capsules adjacent to the fuel pin defects to provide a sink for oxygen during the tests (simulates large bulk of low oxygen sodium available in a typical LMFBR).

Figure 2-9. Capsule Sectioning Diagrams

Review of the Design Basis Specification for the B11A, -B, and -C capsules was completed by the organizations responsible for the design, fabrication, testing, and examination of the capsules and procurement initiated of long lead time items. Difficulty was encountered in procuring the thermocouples to RDT Standard C7-6T, "Thermocouple Assembly, Nuclear Grade, Chromel-P vs Alumel, Stainless Steel Sheathed, Magnesium Oxide Insulated." The difficulty was encountered because Standard C7-6T specifies that tubing for the thermocouple sheaths be procured to RDT Standard M3-2T, "Seamless Ferritic and Austenitic Alloy Tubes." Since this is not a standard normally used by industry none of the vendors could obtain any material meeting the standard without procuring it directly from a steel mill. This resulted in higher prices because of the small amount of tubing required and an increase in the procurement times of from 4 to 10 weeks. Based on an engineering review of the design requirements for the thermocouples and past experience with thermocouples used in irradiations of similar capsules it was decided that tubing procured to ASTM Specification A-269, "Seamless and Welded Austenitic Stainless Steel Tubing for General Service," would meet the experiment design requirements. The thermocouple purchase order was modified to change the tubing specification from M3-2T to ASTM A-269 and submitted for bids again.

2.5 RUN-BEYOND-FAILURE TESTS

Preparation of a test proposal was continued for a series of tests to determine the consequences of post failure operation of mixed oxide fuel rods under typical LMFBR steady state operating conditions. These tests will determine the consequences of post failure operation of mixed oxide fueled subassemblies containing defected rods operating under typical LMFBR steady-state conditions.

A report was completed on the capabilities and feasibility of conducting run beyond failure tests in the EBR-II, SEFOR, FEFPL, FFTF, and forced convection capsule facilities.

250X

UPPER LEFT — 1100°F
 LOWER LEFT — 900°F
 BELOW — 800°F

250X

250X

Figure 2-10. 500 hr Corrosion of Type-316 Stainless Steel by CsOH

Table 2-6
FUEL ROD PARAMETERS FOR CAPSULES B11A, B11B AND B11C

Fuel	B11A	B11B	B11C
Material	Co-ppt (Pu, U)O ₂	Co-ppt (Pu, U)O ₂	Co-ppt (Pu, U)O ₂
Ratio Pu/Pu + U	0.25	0.25	0.25
Ratio U-235/U-235+U-238	≥90%	≥90%	≥90%
O/M Ratio	2.00	1.965	1.965
Pellet Density (% Theo)	88	88	88
Pellet Length (in.)	0.250 NOM	0.250 NOM	0.250 NOM
Fuel-Cladding Bond	Helium	Helium	Helium
Fuel-Cladding Diametral Gap, cold (in.)	0.001	0.001	0.001
Fuel Column Length (in.)	5	5	3
Total Cold Axial Gap (in.)	0.015	0.015	0.015
Blanket and Insulator			
Material	Natural or Depleted UO ₂	Natural or Depleted UO ₂	Natural or Depleted UO ₂
O/M Ratio	2.000	2.000	2.000
Pellet Density (% Theo)	94	94	94
Pellet Dimensions:			
o.d.	0.215	0.215	0.215
Pellet Length (in.)	0.25	0.25	0.25
Blanket Length (above fuel) (in.)	0.250	0.250	0.250
Insulator Length (below fuel) (in.)	0.250	0.250	0.250
Diametral Gap, cold (in.)	0.005	0.005	0.005
Cladding			
Material	316-30% C.W.	316-30% C.W.	316-30% C.W.
Inside Diameter (in.)	0.2205	0.2205	0.2205
Wall Thickness (in.)	0.015	0.015	0.015

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

Task Leader: G. R. Thomas*

3.1 GENERAL

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

3.2 HIGH-BURNUP FUEL EXPERIMENTS

The transient overpower failure thresholds and subsequent fuel-sodium interactions are being investigated for 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 rods irradiated at 13 to 15 kW/ft, and two HEDL supplied FFTF rods (PNL-2-2 and 2-7) which operated at a lower power level of ~10 kW/ft.

3.2.1 Capsule Fabrication and Assembly

The Series 4 (C4M and following) Fabrication Specification and the Series 4 Fabrication Guide were reviewed and approved by BRD. The guide, which is based on the approved Product Quality Plan and Fabrication Specification, represents the forepart of the complete Shop Plan, and is of sufficient detail to initiate machining operations. The balance of the Shop Plan (detailed assembly flow charts) is in preparation and will be reviewed and approved by BRD prior to initiation of assembly. Based on the approved Fabrication Guide, machining operations on main capsule components began in early January 1972.

3.2.2 TREAT Irradiation

The planned order of transient irradiation has been reversed, with the concurrence of HEDL, for the two HEDL-supplied EBR-II irradiated fuel rods, PNL-2-2 and PNL-2-7, (to be used in capsules C4M and C4N). As a result, the PNL-2-7 fuel rod, containing annular fuel, will be tested first in the C4M capsule, and the PNL-2-2 (solid fuel) rod will be tested in the C4N capsule. The objective of the reversal in order is to provide an earlier testing date for studying the potential of interior fuel movement and accommodation in the annular fuel of the PNL-2-7 fuel rod. Annular fuel represents a fuel configuration with a potentially higher damage and failure threshold than solid fuel.

Detailed study of the planned TREAT test conditions for the C4M irradiation continues. It is planned to use the shaped-transient capability of TREAT to achieve the desired test conditions. Figure 3-1 is an example of the ability of an initial flat-top power shape to simulate steady-state fast reactor temperature profiles as the initial conditions for the C4M test. In this case, a 35 MW flat-top transient produced the fit. Work is continuing to optimize the shaped power levels to produce an even closer fit to the reference conditions of Figure 3-1.

Request for Approval-In-Principle for the Task C-1 Series 4 experiments is in preparation. A preliminary information package is in internal review prior to distribution to HEDL and ANL as part of a "round-robin" agreement on intercomparison of pre- and post-test thermal analysis techniques. This information package contains sufficient information concerning the geometry and internal heat generation rate to permit thermal modeling of the C4M capsule. A second information package will be distributed when a test transient has been chosen and detailed physics predictions based on actual PNL-2-7 fuel data are completed.

3.2.3 Post-Irradiation Analysis

Metallurgical mounting has been completed for three sections from the C4G fuel column in preparation for microprobe analysis. The object of this study is to provide empirical data for correcting the predicted radial power

* Contributors are listed in Appendix B.

Figure 3-1. Comparison of Predicted Radial Temperature Profiles-C4M During a 35-MW TREAT Flat-Top Transient versus 15 kW/ft Steady State Profile

profile used in the thermal analysis of the TREAT irradiation of the C4G and C4H fuel rods (previously irradiated in EBR-II to ~50,000 MWd/Te burnup).

3.3 COMPARATIVE TESTS

Preliminary design has been initiated on two capsule assemblies (C6C, C6D) to investigate the effects of steady-state irradiation power levels (i.e., fission gas release rates) on fuel rod transient performance.

A preliminary study reported earlier⁽³⁻¹⁾ indicating the feasibility of testing fully-enriched, EBR-II-type fuel specimens in a C6 type capsule with a thermal flux filter is now being expanded to investigate axial power shaping filter configurations as well as evaluate several inner capsule designs.

3.4 SEFOR TRANSIENT OVERPOWER TESTS

The SEFOR is a unique fast reactor test facility capable of simulating LMFBR accident conditions initiated from zero or steady-state power in the controlled environment of an encapsulated, instrumented fuel specimen. Work on the first of a series of capsules included as part of a comprehensive test program is being funded under Task C1.

3.4.1 Test Planning

The first test capsule of this test series has been designated TOP-1A.

The primary purpose of this capsule is to:

1. Proof test the capsule design and demonstrate the potential of SEFOR for transient testing.
2. Establish a correlation between TREAT and SEFOR tests for simulating LMFBR accident conditions.

3.4.2 Capsule Design

The design of the first transient overpower capsule (TOP-1A) assembly for SEFOR has been finalized. Drawings and related documents to initiate fabrication are nearing completion. A cross-section of the capsule and $\text{ZrH}_{1.6}$ moderator container is shown in Figure 3-2. The $\text{ZrH}_{1.6}$ moderator container is a separate and independent assembly which also provides alignment and support for the capsule. Procurement of both the $\text{ZrH}_{1.6}$ moderator and capsule thermocouples has been initiated. This capsule assembly is designed for irradiation in an off-center core position (inner refueling port penetration). The fuel rods and associated support items will be removed from the hexagonal channel located below the refueling port. The $\text{ZrH}_{1.6}$ assembly and associated test capsule support items will then be installed with the test capsule positioned in the center of the hexagonal channel.

It is planned that most subsequent tests will be conducted in the core center position following relocation of the existing transient rod mechanism (FRED—Fast Reactivity Excursion Device) from this position to an off-center refueling port. The test capsule design will be similar, with possible material and/or dimensional changes in the capsule thermal barrier to utilize the larger access capability of the center position which will accommodate up to three test capsules in a triangular array. The design of the $\text{ZrH}_{1.6}$ assembly and capsule support will be modified to accept up to three capsules.

Figure 3-2. SEFOR Capsule TOP-1A Cross-Section (Outer Core Position), Centered in Hex Channel

3.5 LIAISON

Task C-1 participated in a discussion with ANL/HEDL/RDT on transient overpower test design and analysis methods and their objectives on October 29, 1971. The establishment of a format for providing cooperative exchange of basic transient test data and modeling information for completed ANL/GE/HEDL transient overpower tests was recommended. Task C-1 (PA-10) was assigned the task of developing the test data request format. This format has been completed and distributed to ANL, HEDL, and RDT for comment and review.

Task C-1 staff met with RDT in early December to update the SEFOR Follow-on Option I TOP-1A capsule design and test plans.

REFERENCES

- 3-1. *Sodium-Cooled Reactors Fast Ceramic Reactor Development Program, Thirty-Eighth Quarterly Report, February-April 1971, May 1971, (GEAP-10028-38).*

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 rods and the time characteristics of signals (such as temperature and flow) are of primary concern in the design of the fuel bundles and instrumentation system. Firm data in these areas must be available to provide reliable input to accident analyses for licensing purposes.

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

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

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

4.2 OUT-OF-PILE TRANSIENT UNDERCOOLING TESTS

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

4.2.1 Mockup 1A

Mockup 1A was the first out-of-pile transient undercooling test⁴⁻¹. The purpose of this experiment was to check out the loop operation and instrumentation response during a loss-of-flow test.

A transient thermal hydraulic analysis of the Mockup 1A test was made using the BUBBLE/VOIDUM model. The time predicted for the onset of boiling was in good agreement with the observed data. However, the extent of voiding was significantly overestimated. The model predicted complete expulsion in the upward direction while test results indicated very limited bubble growth. The difference between the calculated and observed results is attributed to radial condensation caused by the large radial temperature gradient resulting from the premature failure of one of the three heaters. This radial condensation potential is being explored further in the Mockup 1B test. A draft of the final report on Mockup 1A is nearing completion.

4.2.2 Mockup 1B

Mockup 1B was used to investigate the initial stages of loss-of-flow accidents in an out-of-pile loop experiment utilizing the same facility as the Mockup 1A test. This investigation includes the flow transient as well as the initial stages of void-formation.

Mockup 1B data evaluation is continuing. Accelerometer and sodium flow test data originally recorded on magnetic tapes were reproduced on oscillograph charts for visual inspection and correlation with the rest of the test data. Temperature, pressure and flow data originally recorded on oscillographs have been digitized and stored on tapes

* Contributors are listed in Appendix B.

to facilitate data reduction and interpretation. A data retrieval algorithm and plotting routine have been developed to process the Mockup 1B and future test data using the GE 635 computer and plotter.

4.3 SERIES C11

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

The Design Basis Specification for capsule C11A has been approved and issued. Work has been initiated on the conceptual design.

4.4 SERIES C12

The C12 series of tests will investigate transient undercooling accidents using flowing sodium multi-pin loops.

The first test in this series, C12A, is planned to be run in an ANL Mark II-A loop. Work on the experimental proposal for this test is in progress. A letter has been drafted to ANL requesting current copies of the Mark II-A loop QA/QC procedures and design specifications.

4.5 LIAISON

FFTF operating data have been received from WADCO as input to the test parameters for the C11A test. These data consist of FFTF operating parameters for the fuel subassemblies under the 600°F inlet operating condition.

Meetings were held with the GE representative at ANL (Task G-12) to discuss the R loop test capabilities and R-Series program. Evaluation of the R loop test vehicle is in progress.

REFERENCES

- 4-1. *Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Thirty-Third Quarterly Report, November 1969-January 1970, February 1970, (GEAP-10028-33), page 23.*

5. TASK E—FUEL PROPERTIES AND PERFORMANCE EVALUATION

Task Leader: B. F. Rubin*

5.1 GENERAL

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

5.2 CENTER TEMPERATURE LIMITS AND FUEL BEHAVIOR

5.2.1 Central-Hole Formation at Reactor Startup—Capsule E2R

This capsule, containing 88% and 94% T.D. mixed oxide pellets, is 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 minutes making a total exposure of six hours. Neutrographs taken after each exposure revealed that the central hole had formed in the 88% T.D. pellets at some time during the 15-minute exposure, between 11 and 26 minutes. The central void measured 25 to 30 mils. No central hole had yet formed in the 94% T.D. pellets.

5.3 ELECTRON MICROPROBE EXAMINATIONS

5.3.1 Examination of Irradiated Specimens for Fuel and Cladding Changes

5.3.1.1 Specimen F3B-3G of the F3A Test (~5.3 at.% Burnup)

This specimen was selected from the peak power region of the rod. Other sections of this rod have been examined with the electron microprobe and the results have been reported in previous quarterly reports^(5-1,5-2). The Pu/(U + Pu) ratio, xenon and oxygen distributions within this sample are presented in Figure 5-1, along with a radial microstructural view. The relative plutonium concentration increased from 25% to 35% Pu/(U + Pu) at the fuel rod center. A similar increase was found in the adjacent radial section. The Pu/(U + Pu) ratio in a specimen near the top of this fuel rod increased to about 40% within the center 10% of the fuel volume⁽⁵⁻¹⁾. In the present case the increase affects somewhat less than 10% of the volume of the fuel annulus, as Figure 5-2 shows.

The fuel/fission-product-cladding reactions in this section were slight and for the most part were similar to those seen in the other sections from this fuel rod^(5-1,5-2). A typical area, detailed in Figures 5-3 and 5-4 displayed molybdenum, tellurium, cesium and oxygen. Iron and chromium were not present in this region but were concentrated at the fuel interface.

5.3.1.2 Specimen E1H-30J of the F3B Test (~8 at.% Burnup)

This specimen was selected near the top of the fuel column of this rod. It was a mirror image of the sample illustrated in Reference 5-3 which was notable for the broad bands, up to six mils in width, of fission-product/cladding reaction product. It was clad in cold worked Type 304 tubing that had been given an extended heat treatment to improve its creep properties. **

The fuel/fission-product-cladding reaction band was examined at the widest, narrowest and intermediate locations for elemental composition and distribution. Point count concentrations for iron, chromium, nickel, cesium, tellurium, molybdenum, uranium and oxygen were taken with the energy dispersive spectrometer system at all three locations. The results are being run through the MAGIC-IV Computer Code (see Subsection 5.3.3) which reduces the values to quantitative concentrations. These will be included in the next progress report. Qualitative spatial

* Contributors are listed in Appendix B.

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

Figure 5-1. Photomicrograph of Specimen F3B-3G and Corresponding Radial Distributions of $Pu/(Pu + U)$, Oxygen and Xenon

Figure 5-2. Plutonium Distribution of Specimen F3B-3G (Figure 5-1), Plotted to Show Volume Relationships

distributions are presented in Figures 5-5 and 5-6 of the broadest interaction zone. The first of these shows that the cladding alloying elements are present within the entire region; iron seems concentrated closest to the cladding while a chromium-rich zone appears about 1/3 of the way across the band. Migrating nickel has concentrated at the interface near the fuel. Figure 5-6 shows the uranium and fission product distribution within the reaction band. Tellurium and cesium are present along with oxygen. The cesium concentration appears greatest in the center of the band where the tellurium concentration appears lowest. The oxygen distribution most closely follows that of the tellurium. Uranium is present closest to the fuel edge, and this marks the first known instance in which fuel has been a component of a reaction zone. Molybdenum was not detected in this reaction layer. Figure 5-7 shows the $Pu/(Pu+U)$ distribution across a radius of this fuel rod section.

Molybdenum was present, however, in a fission-product ingot at the center of the fuel structure, Figure 5-7. This element makes up 46.7% of the chemical composition, with the balance comprised of 5.0% palladium, 20.0% ruthenium, a trace of rhodium, and, by difference, 28.3% technetium. In addition a ring of fuel was observed within the ingot.

5.3.2, Examination of Irradiated Specimens for Fuel Changes

5.3.2.1 Specimens of the E1J Test (<1 at.% Burnup)

The E1J test consisted of a triplicate series of three (U-25 wt % Pu) O_{2-x} rods irradiated in the thermal flux of GETR at power levels sufficient to produce center melting. Each group of three rods contained fuel with O/M ratios of 1.94, 1.97 and 2.00. These three rods, E1J-2, E1J-6 and E1J-9 had been analyzed for the radial plutonium concentration as reported earlier⁽⁵⁻⁴⁾ and have now been examined for oxygen distribution. No values could be obtained on E1J-2 with the lowest O/M ratio probably because of the lower electrical conductivity and more highly fractured surface of this sample. The radial intensity of the oxygen values from the samples with original O/M ratios of

A781-7

PHOTO MICROGRAPH

500X

2MA-2F

IRON X-RAY IMAGE

500X

2MA-2A

NEGATIVE SPECIMEN CURRENT IMAGE

500X

2MA-2G

CHROMIUM X-RAY IMAGE

500X

Figure 5-3. Photomicrograph, Specimen Current and X-ray Images of Iron and Chromium in the Fuel-Cladding Interaction Zone in the Mid-plane of F3B-3

2MA-2E MOLYBDENUM X-RAY IMAGE 500X

2MA-2C CESIUM X-RAY IMAGE 500X

2MA-2D TELLURIUM X-RAY IMAGE 500X

2MA-2B OXYGEN X-RAY IMAGE 500X

Figure 5-4. X-ray Images of Molybdenum, Cesium, Tellurium and Oxygen in Same Area as Figure 5-3

Figure 5-5. Specimen Current and X-ray Images of Chromium, Iron and Nickel in the Wide Fuel-Cladding Interaction Zone, Specimen E1H-30J

Figure 5-6. X-ray Images of Tellurium, Oxygen, Cesium and Uranium of the Same Area as Figure 5-5

Figure 5-7. Photomicrograph of Specimen EIH-30J and Corresponding Radial Distribution of $Pu/(Pu + U)$

1.97 and 2.00 are presented in Figure 5-8. Fuel microstructures and O/M and O/Pu ratios also appear in this figure. These values were derived from comparison with a set of O/M standards. These were characterized mixed oxide specimens that had analyzed O/M values of 2.000, 1.973, 1.968 and 1.935. Figure 5-9 illustrates the relationship between relative intensity of the oxygen signal and O/M for this composition. The transpositioning of these limits back to Figure 5-8 indicates one sigma variations around the mean value of $\sim \pm 1.5$ O/M units. The stoichiometric sample, E1J-9, displayed a much more uniform oxygen profile than E1J-6. Both samples exhibited a decrease near the cladding wall. The drop in oxygen intensity near the center of the hypostoichiometric sample would be predicted by theory and has been calculated by the method reported in Reference 5-1. Similar behavior was noted in the oxygen analyses of three other specimens from the E1J test reported in December 1970⁽⁵⁻⁵⁾.

5.3.3 Examination of Unirradiated Specimens for Cladding Changes

5.3.3.1 Specimens from the Thermal Gradient Capsule for Studying Fission Product/Cladding Reactions (5.4.2)

Three cladding samples from Series I and II of Table 5-1 exposed at 615 to 800°C for 100 hours have been examined. The samples were mounted in lead-bismuth alloy so that the mounting compound would not interfere with carbon and nitrogen determinations. The specimens were of Types-316, -321 and -347 stainless steel in which some second phase precipitation has been observed in the as-polished condition using light optical microscopy. Selected X-ray images of the Type 347 and Type 316 capsules are presented in Figures 5-10 and 5-11, respectively. The reactions with the Type 321 cladding closely resembled those of Type 347 except that the latter contained zones of attack or precipitation resembling sub-surface inhomogeneities (Figure 5-10). All three specimens had a surface layer that contained chromium, cesium and silicon as major constituents with carbon, oxygen, nitrogen and manganese in minor concentrations. The sub-surface structure in the Type-347 stainless steel contained some cesium and silicon and the

Figure 5-8. Radial Oxygen Concentration Across Two E1J Specimens

Figure 5-9. Relative Oxygen X-ray Intensity of Four Fuel Samples of Known O/M—One Sigma Error Band is Included

region was depleted in chromium but had approximately the matrix content of iron and nickel. The nitrogen-rich layer in the Type 316 tubing blended into the matrix, Figure 5-11, but in Types 347 and 321 material the nitrogen-rich region remained distinct and segregated, Figure 5-10. This difference might be expected since the titanium and niobium in Types 321 and 347 are strong nitride formers and hence would be expected to "getter" the nitrogen near the surface; but in Type 316 tubing the incoming nitrogen would dissolve within the lattice in an asymptotic approach to the matrix concentration.

It is concluded that the minor substitutional constituents (Si, Mn) may be important in the reaction of fission products and cladding, with silicon and cesium frequently found together. Oxygen, carbon and nitrogen also appear to be involved in this reaction, Subsection 5.4.4.

5.3.3 Computer Processing of Electron Microprobe Output

The computer program MAGIC-IV (Microprobe Analyses General Intensity Corrections) has been adapted for use with the GE 635 unit. This program corrects the point count output of the wave length dispersive spectrometer system for background, absorption coefficients and dead time. The code compares the measured intensity value with that from a standard and accurately determines the elemental concentration. A computer program is being prepared for reducing the X-ray data from the energy dispersive, often termed the non-dispersive system used for rapid elemental scanning. This program will subtract background from the characteristic peaks and determine the elements most likely to have caused them, and which are therefore most likely to be present. This will allow the analyst to select the responsible elements and their intensities for inputting into the above MAGIC-IV Code to yield quantitative estimates of their concentrations.

NOTE:

THE CAPSULE CONTAINED MIXED OXIDE FUEL AND Cs_2O . CESIUM AND SILICON SEEM COMBINED IN THE SUBSURFACE VOIDS WHICH ARE DEPLETED IN CHROMIUM. THESE VIEWS WERE AT 720°C . (SERIES IMA114) (IN HOMOGENEITIES)

Figure 5-10. Electron Microprobe Views of Treated Type-347 Cladding

NOTE:
THE CAPSULE CONTAINED MIXED OXIDE FUEL AND Cs_2O . THESE VIEWS ARE AT 730°C . (SERIES 1MA113)

Figure 5-11. Electron Microprobe Views of Treated Type-316 Cladding

Table 5-1
STATUS AND RESULTS FOR CLADDING SELECTION CAPSULES LOADED WITH FUEL AND Cs₂O

Capsule Series	Cladding ^(a)		Other Variables	Reaction ^(c) Temperature (°C, ±25°C)	Comments and Status
	Alloy	Condition ^(b)			
I-A	316	Ann.	Cs ₂ O at cold end	760	Grain boundary phase at 550°C ^(e)
I-B	316	Ann.	○	700	Film ^(f) at 550°C
I-B-D	316	Ann.	Discs in I-B ^(d)		Same as cladding at ~800°C
I-C	321	Ann.		575	This reaction not visible in as-polished state
I-D	347	Ann.	ASTM ^(g) 8-1/2	725-800	Film at 550°C, resembles I-B
I-E	316	Ann.	ASTM 5-1/2	700	Film at 550°C, resembles I-B
I-F	347	Ann.	ASTM 10	725-800	Film at 550°C
II-A	304	12% C.W.		800-900	Film at 500°C, reaction intergranular
II-B	304	20% C.W.		700-975	Film at 500°C reaction intergranular
II-C	316	10% C.W.		825-1000	Reaction intergranular
II-D	316	30% C.W.		775-960	Intergranular and intragranular reactions
II-E	316	50% C.W.		775-960	Film at 500°C, reaction resembles II-D
II-F	316	Ann.		775-960	Film at 500°C, reaction resembles II-C
II-F-D	304 ELC	Ann.	80 ppm C discs in II-F	590-890	Appears carburized, electron probe showed N, no C in reaction layer
III-A	316	Ann.		625	Similar to I-B
III-A-D	In 744	Ann.	Discs in III-A		Reaction (as polished)
III-B	316	Ann.	Low Si (0.17%)	800	More extensive reaction than I-B, III-A (normal Si = 0.609)

Table 5-1. (Continued)

Capsule Series	Cladding ^(a)		Other Variables	Reaction ^(c) Temperature (°C, ±25°C)	Comments and Status
	Alloy	Condition ^(b)			
III-C	1800	Ann.			General reaction. Intergranular attack.
III-D	1800	30% C.W.			General reaction. Intergranular attack.
III-E	304	12% C.W. + GT			General reaction (as polished)
III-F	304	20% C.W. + GT			Same as III-E
IV-A	321	Ann. V			General reaction (as polished)
IV-A-D	316 + Ti	Ann.	Discs in IV-A		At least as much reaction as in IV-A (std. 316, No-Ti)
IV-B	321	Stab.			Same as IV-A
IV-B-D	1414	CW?	Discs in IV-B	728-828	General reaction
IV-C	321	Ann.	200 mg Cs ₂ O (4 X normal)	1100	More reaction than in IV-A
IV-C-D	321 Mod	Ann.	Discs	~510	Similar to IV-C
IV-D	316	Ann.	2.16 O/M at hot end, no Cs ₂ O	650-1000	Reaction restricted to 2.16 O/M region (mean O/M = 2.00)
IV-E	316	Ann.			Same as I-B, etc.
IV-E-D	Nb		Discs in IV-E	628	Extensive reaction
IV-F	316	Ann.	200 mg Cs ₂ O		
IV-F-D	Nb		200 mg discs in IV-F	513	Very extensive reaction
V-A	316	Ann.	750-400°C gradient, tagged Cs	725	Some intergranular attack at hot end
V-B	321	Stab.	750-400°C gradient, tagged Cs	740	Some intergranular attack at hot end
VI-A	316	Ann.	870-510°C gradient		Pulsed eddy current testing after 100 hours indicated reactions. Reheating to 1000 hours.

Table 5-1 (Continued)

Capsule Series	Cladding ^(a)		Other Variables	Reaction ^(c) Temperature (°C, ±25°C)	Comments and Status
	Alloy	Condition ^(b)			
VI-B	316	Ann.	810-510°C gradient		Pulsed eddy current testing after 100 hours indicated reactions. Reheating to 1000 hours.
VI-C	321	Stab.	870-510°C gradient, Ann. 321 discs		Pulsed eddy current testing after 100 hours indicated reactions. Reheating to 1000 hours.
VI-D	321	Stab.	810-510°C gradient Ann. 321 discs		Pulsed eddy current testing after 100 hours indicated reactions. Reheating to 1000 hours.
VI-E	347	Stab.	870-510°C gradient Ann. 347 discs		Pulsed eddy current testing after 100 hours indicated reactions. Reheating to 1000 hours.
VI-F	347	Stab.	810-510°C gradient Ann. 347 discs		Pulsed eddy current testing after 100 hours indicated reactions. Reheating to 1000 hours.

- (a) Operated 100 hours in 1100° to 550° C thermal gradient, 50 mg Cs₂O added to hot end unless noted.
- (b) Ann. = annealed (1950° F, WQ), C.W. = cold worked, GT = Garafalo treated (900° F, 24 hours + 1300° F, 216 hours). Ann.-V = annealed at VNC, GTV = Garafalo treated at VNC, Stab. = stabilized (1650° F, 1 hour + He quench).
- (c) "Reaction" in a surface layer ~0.003-inch thick containing second phases observed either as polished or after oxalic acid electrolytic etching. If no range in temperature given, then value is lowest temperature at which reaction was observed.
- (d) Discs of specific alloys interspersed between fuel pellets.
- (e) This grain boundary phase contained no Cs according to electron microprobe analysis.
- (f) "Film" is a continuous, structureless oxidation product layer ~0.0002-inch or less thick.
- (g) "ASTM" = ASTM grain size of cladding. Normal value is ASTM ~8.

5.4 FUEL/FISSION-PRODUCT/CLADDING REACTIONS

5.4.1 Fission Product Migration and Reactions

Gamma scan results of a recent series of out-of-pile fueled thermal gradient capsules were reported earlier⁽⁵⁻³⁾. These were from Capsule Series VI of Table 5-2. In these, cesia and tellurium or cesia and iodine were added to the fuel in 1/1 or 4/1 ratios and heat treated 100 hours within a 550 to 1100°C temperature range. Metallographic results are presented below with related observations from Series V.

Specimen	O/M	Cs/Te		Reaction ~875 to 1000°C Range
		Atom Ratio	Wt., mg	
V-A	1.97	1/1	50/50	No grain boundary attack
VI-B	2.00	1/1	50/50	Grain boundary attack, ~2 mils
VI-A	2.00	4/1	50/15	No grain boundary attack
VI-C	2.00	0/1	0/12	Grain boundary attack, ~1 mil

These results suggest that:

- Free tellurium or the tellurium exceeding any Cs₂Te relationship reacts with cladding grain boundaries in a sufficiently high O/M (low ΔG_{O_2} environment). This is shown in Figure 5-12.
- When all of the tellurium is combined with the cesium, neither the resultant compound nor the excess cesium reacts with the cladding. (Specimen V-A.) It is possible, however, that the different behavior of specimens VI-B and VI-A results not from the combination and inerting of Te as Cs₂Te or Cs₂TeO₂ but from the greater concentration of tellurium in specimen VI-B. This is suggested by the deeper penetration in specimen VI-B compared to specimen VI-C.
- Cladding reactions are retarded by low O/M fuel (Specimen V-A).

Additional testing is planned to investigate O/M and the tellurium concentration effects.

Non-destructive testing from the capsules of Series VII (Table 5-2) was also reported in Reference 5-3. The most significant metallographic observation was the appearance of the CrO_{1.5} layers, about 2 to 3 microns thick, along the cladding interior of specimens C and D which contained Mo + Mo O₃ additives, no Cs₂O, in stoichiometric fuel. This is shown in Figure 5-12 (d).

This film was not present in capsules containing cesia nor in rod VII-E which had a similar loading except that it contained non-stoichiometric fuel. In addition, a fuel precipitate believed to be a carbide or nitride, and illustrated in Figure 5-12 (c), was present in most samples except those displaying the CrO_{1.5} film. It is possible that carbon or nitrogen in the fuel diffused into the cladding and caused this precipitate.

These findings provide some evidence for the supposition that Cs₂O destroys the protective oxide layer that normally shields stainless steel from corrosive attack and enables intergranular reactions to occur.

The protective layer developed in the capsules with molybdenum and the stoichiometric fuel. This film retarded the migration of interstitials from the fuel into the cladding as in the above case. The protective film did not form in capsules with substoichiometric fuel or in capsules containing Mo and Cs₂O.

5.4.2 Cladding Modification Testing

Etched metallographic sections of a number of specimens of this program have become available. The results are tabulated in Table 5-1. Cs₂O was added to these fueled capsules which contain tubing or discs of a number of austenitic alloys in several thermomechanical states which are subsequently heat treated in a thermal gradient. Type 321 tubing which showed no reaction in the as-polished condition⁽⁵⁻¹⁾ indicated structural changes at temperatures as low as 575°C after etching.

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

	O/M	Temperature Range (°C)	Container	Additive	Results
Series I	1.967	660-1550	Mo	Cs ₂ MoO ₄ /U	Cs ₂ O increased O/M redistribution rate.
Series 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 to 750°C range.
Series II-B	1.967	550-1550	Mo	Cs ₂ MoO ₄ /U	
Series II-C	1.967	660-1550	Mo	Cs ₂ MoO ₄ /U	
Series III-A	1.967	550-1150	316 SS	Cs ₂ MoO ₄ /U	Cs participated in surface reaction noted above. Cs ₂ O in either form reacted.
Series III-B	1.967	550-1150	316 SS	Cs ₂ O	
Series III-C	1.967	550-1150	316 SS	None	Stainless capsules suitable for testing.
Series IV-A	1.94	550-1150	316 SS	Cs ₂ O	Amount of CsO retained at high temperature end increased with O/M.
Series IV-B	1.967	550-1150	316 SS	Cs ₂ O	Intergranular attack with O/M ≥ 2.00.
Series IV-C	2.00	550-1150	316 SS	Cs ₂ O	
Series IV-D	2.01	550-1150	316 SS	Cs ₂ O	Excess Cs ₂ O ruptured Capsule E.
Series IV-E	2.00	550-1150	316 SS	Cs ₂ O Excess	
Series 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 .
Series V-B	1.97	550-1150 ^(b)	316 SS	Cs ₂ O + I ^(d)	No reaction with iodine or Te.
Series V-C	2.00	550-1150 ^(b)	316 SS	Cs ₂ O + I ^(d)	I did not affect Cs ₂ O redistribution.
Series V-D	2.00	550-1150 ^(b)	316 SS	Cs ₂ O + Te ^(c)	Cs and Te appear to cohabit; Te lessens migration of Cs ₂ O to cold end.
Series V-E	2.00	550-1150 ^(b)	316 SS	Cs ₂ O Excess	Excess of Cs ₂ O produced drastic intergranular attack.
Series V-F	2.00	550-1150 ^(b)	316 SS	Cs ₂ O Excess	Cs ₂ UO ₄ and Cs ₂ U ₂ O ₇ identified at cold end in E and F.

Table 5-2 (Continued)

	O/M	Temperature Range (°C)	Container	Additive	Results
Series VI-A	2.00	550-1150	316 SS	Cs ₂ O + Te ^(d,h)	Cs and Te cohabited. Excess Cs deposited along length, migrating to cold end, but inhibited by Te and less migrated than in D.
Series VI-B	2.00	550-1150	316 SS	Cs ₂ O + Te ^(c,i)	Cs and Te cohabited at about 725°C. Te restricted Cs migration to cold end.
Series VI-C	2.00	550-1150	316 SS	Te ^(h)	Intergranular attack, 3 to 4 mils. Intergranular attack, ~1 mil.
Series 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.
Series 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 soaking.
Series 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 soaking.
Series VII-A	2.00	550-1150	316 SS	Cs ₂ O + Mo ^(c)	Mo remained at hot end; Cs ₂ O deposited along length.
Series VII-B	2.00	550-1150	316 SS	Cs ₂ O + Mo ^(e)	{ Mo remained at hot end. Mo remained at cold end. Green Cr₂O₃ formed an 800 to 900°C region of clad }
Series VII-C	2.00	550-1150	316 SS	MoO ₃ + Mo	
Series VII-D	2.00	550-1150	316 SS	MoO ₃ + Mo ^(f)	
Series VII-E	1.96	550-1150	316 SS	MoO ₃ + Mo	Mo remained at hot end.
Series VII-F	2.00	550-1150	316 SS	Cs ₂ O (Reheat of Capsule VI-D)	No significant change in Cs ₂ O positions.

Table 5-2 (Continued)

	O/M	Temperature (Range (°C))	Container	Additive	Results
Series VII-A	1.97	550-1150	316 SS	Te ^(h)	
Series VII-B	2.00	550-1150	316 SS	Cs ₂ O + Te ^(c,h)	
Series VII-C	2.00	550-1150	316 SS	Cs ₂ O + Te ^(d,h)	
Series VII-D	2.00	550-1150	316 SS	Te ⁽ⁱ⁾	
Series VII-E	2.00	550-1150	316 SS	Cs ₂ O + Te ^(e,i)	
Series VII-F	2.00	550-1150	316 SS	Cs ₂ + Te ^(d) + Mo + I	
Series IX-A	2.00	500-750	316 SS	Cs	
Series IX-B	2.00	500-750	316 SS	Cs ₂ O + Te ^(d)	
Series IX-C	2.00	500-750	316 SS	Cs ₂ O + Te ^(c)	
Series IX-D	2.00	500-750	316 SS	Cs	
Series IX-E	2.00	500-750	316 SS	Cs ₂ O + Te ^(d)	
Series IX-F	2.00	500-750	316 SS	Cs ₂ O + Te ^(c)	
Series IX-G	2.00	500-750	321 Stub	Cs ₂ O	
Series IX-H	2.00	500-750	321 ST	Cs ₂ O + Te ^(d)	
Series IX-I	2.00	500-750	347 ST	Cs ₂ O + Te ^(c)	

- (a) Fueled capsules heated 100 hours in thermal gradient 550 to 1100°C
 (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 or Cs/Te
 (f) Additive positioned at cold end
 (g) Cs₂O additions are 50 mg (~2 at.% burnup) except IV-E (500 mg), V-E (100 mg) and V-F (200 mg)
 (h) Te addition is 12 to 15 mg
 (i) Te addition is 50 mg
 All tests last 100 hours except 1X-D, 1X-E, 1X-F (1000 hours) and 1X-G, 1X-H, 1X-I (900 hours)

In order to correlate findings with the lower temperature regime expected for LMFBR operation, a series of tests are being conducted at lower temperatures and in a less steep thermal gradient. Gamma scans indicated that after 100 hours in a 400 to 750°C thermal gradient, Cs₂O had migrated to the cool end. A second set of capsules (Series VI of Table 5-1) loaded with Cs₂O began a 1000-hour heat treatment with an intermediate halt after 100 hours for gamma scanning and pulsed eddy current testing. Three rods operated within a 510 to 870°C gradient while three others operated in a 510 to 810°C gradient. After a 100-hour exposure (Figure 5-13) the majority of the cesium remained at the hot end of the high temperature test where it was originally deposited and approximately fifteen percent was distributed uniformly along the remainder of the capsule. The Cs₂O movement in the lower temperature tests was less uniform than in the higher temperature specimens. More cesium migrated to the cooler region in the samples treated at the lower temperature. This may be explained in terms of a reaction: $2\text{Cs}_2\text{O} + (\text{U,Pu})\text{O}_2 \rightarrow \text{Cs} + \text{Cs}_2(\text{U,Pu})\text{O}_4 + (\text{U,Pu})\text{O}_{2-x}$, that occurs at the higher temperature, but occurs to a lesser extent at the slightly reduced temperature. With the reaction exaggerated at the higher temperature, less Cs is available to migrate down the thermal gradient in these capsules.

Six rods were pulsed eddy current (PEC) tested to determine whether this equipment could be used to determine the severity of attack after 100 hours. The indications in Type 347 tubing corresponded to less than 0.0005 inch in depth except that one defect, about one mil in depth, was found at a location corresponding to a temperature of 520°C. Types 316 and 321 tubing had some 0.0005 to 0.001 inch or smaller defects at the hottest fuel temperatures.

935-4

400X

(a) INTERGRANULAR REACTIONS INDUCED BY Te

932-5

400X

(b) INTERGRANULAR REACTIONS INDUCED BY Cs/Te ADDITIONS IN A 1/1 RATIO 50 mg CONCENTRATIONS

876-2

400X

(c) INTERSTITIAL SURFACE REACTIONS

873-5

400X

(d) $\text{CrO}_{1.5}$ LAYER IN CAPSULE WITH Mo + Mo O_3 ONLY

Figure 5-12. Cladding Reactions Resulting from Thermal Gradient Capsule Tests with Fission Product Additives

Figure 5-13. Cesium Activity in Capsules Treated 100 Hours in Thermal Gradient, (Series VI, Table 5-1)

An example of the traces for Type 321 tubing (Capsule VI-D) in the 810 to 510°C gradient is presented in Figure 5-14. The indications correspond to defect depths of about 0.001 inch at 780 to 760°C. These results indicate that PEC testing can be used to follow the progress of the cladding attack. Group VI capsules were returned to the thermal gradient furnace for an additional heat treatment of 900 hours.

5.4.3 High Magnification Examination of Fission Product Attacked Cladding

A program has been initiated to examine at high magnification cladding which has been attacked by fission products. Scanning and transmission electron microscopes will be utilized on replicas from out-of-pile and irradiated test specimens to examine topography and to seek to identify compounds involved in these reactions.

A sample from rod F3B-3 which had experienced 0.0075 inch fuel/cladding chemical attack has been replicated. The nature of the reaction phase seemed to be revealed at ~20,000X but the work is continuing to eliminate the possibility of misleading artifacts. A two-step replication process was used with polymeric replicas prepared in the hot cell from the polished sample. These transfer replicas were removed to the laboratory where sturdier carbon replicas for use in the microscopes were prepared. Heavy metal shadowing techniques for the carbon replicas are being optimized. Future work will involve the preparation of extraction replicas to identify the chemical and crystallographic makeup of portions of the reaction phases.

5.4.4 Fuel Impurity Effects

There are some indications that high concentrations of carbon or nitrogen in the fuel may be related to the extent of intergranular attack. An out-of-pile thermal gradient test series has been planned to investigate these reactions. Mixed oxide pellets that contain 500 ppm carbon or nitrogen have been prepared and loaded into Type-316 stainless steel capsules with and without Cs₂O. The radioactive carbon was added as BaCO₃, and the nitrogen was added as UN. These capsules will be heat treated for 100 hours in a 550 to 1100°C thermal gradient.

Figure 5-14. Pulsed Eddy Current Traces Taken on Rod V1-D (Table 5-2)

5.5 FUEL-SODIUM REACTION KINETICS

The transducer equipped stainless steel vessel for monitoring fuel swelling upon exposure to hot sodium has operated at 1000°F during several tests. Slight modifications are being made which are expected to overcome the galling that prevents zeroing of the transducer at high temperatures. Stainless steel capsules loaded with $\text{UO}_{2.05}$ pellets have been prepared for initial testing.

5.6 MEASUREMENT OF OXYGEN POTENTIAL OF MIXED OXIDE FUEL

ZrO_2 -CaO electrolyte tubes have been loaded with Ni/NiO electrodes and the platinum lead wires have been bonded to the tubes to insure continuous contact. Emf measurements have been taken on $\text{UO}_{2.02}$ and $\text{UO}_{2.04}$ during testing at 550 to 750°C and these have corresponded to previously tested values within ~3-4 mv (0.3-0.4 Kcal/mole). ThO_2 - Y_2O_3 tubes, intended for measuring the low $\Delta\bar{G}_{\text{O}_2}$ values of hypostoichiometric fuel are being loaded and will be tested with Cr/CrO_{1.5} and Nb/NbO. These couples have $\Delta\bar{G}_{\text{O}_2}$ approximately equivalent to that of (U-20 wt.% Pu)O_{1.97}, ~ -150 Kcal/mole.

A duplex electrolyte tube is being constructed to measure the broad range of oxygen potentials that occurs in irradiated fuel. In this, ThO_2 - Y_2O_3 will be deposited upon the ZrO_2 -CaO tube. (U-15 wt.% Ce)O_{2±x} pellets will be prepared to simulate the range of $\Delta\bar{G}_{\text{O}_2}$ values for out-of-cell testing.

A remotely operated fixture is being designed that will permit the measurement of oxygen potential at varied radial positions in irradiated fuel specimens.

5.7 MELTING POINTS OF IRRADIATED MIXED OXIDES

Initial results of the cooperative LASL-BRD program to measure this property by differential thermal analysis (DTA) techniques were reported in Reference 5-3. The release of large quantities of fission gas during the heating

period caused premature rupture of the tungsten capsules containing several samples. Smaller fuel loadings and thicker capsules are being tried. Results to date are listed in the following tabulation.

Sample	Pu wt%	O/M	Burnup at. %	Solidus (°C)	Liquidus (°C)
A-1	25	1.96	0	2790	2840
A-2	20	2.00	0	2745	2780
F3B-7	25	1.98	4.3	2745	2785

The relative (A-1)-(A-2) temperatures appear reversed although the results have been repeated three times. The O/M variations are not believed to account for the significant difference.

5.8 Kr-85 STANDARDIZATION PROGRAM FOR DETERMINING CLADDING OPERATING TEMPERATURES

One method of determining peak cladding temperatures in fuel element cladding is to utilize the Kr-85 evolution technique reported by Swanson⁽⁵⁻⁶⁾. Carter and Colligan⁽⁵⁻⁷⁾ have shown that when a surface impregnated with inert tracer gas is heated, it loses a predictable fraction of its initial activity. No further loss of activity takes place upon subsequent heating at or below this temperature. The maximum temperature reached by a krypton impregnated specimen can, therefore, be found by reheating the specimen and monitoring the release of krypton as a function of temperature. A typical graph of this procedure is shown in Figure 5-15. According to Swanson⁽⁵⁻⁶⁾, the best estimate of temperature is obtained by averaging the temperatures A and B, (476 and 500°C) yielding a value of 488°C in this instance. In addition to estimating temperatures of fuel element cladding utilizing fission product krypton, temperatures of non-fuel components such as material specimens may possibly be determined by krypton impregnation prior to irradiation using ion bombardment or high temperature, high pressure diffusion.

Figure 5-15. Release of Krypton from E1H-29 as Function of Temperature

The Kr-85 investigative and standardization program has the following objectives:

1. To determine the analytical precision and confidence limits of this technique by preparing "kryptonated" (krypton impregnated) standards that will be heat treated at several temperatures and then analyzed by this release method to ascertain this temperature.
2. To determine the "memory effect" by cycling kryptonated samples at several temperatures and times, then analyzing to confirm the maximum value.
3. To determine whether kryptonated non-fuel element components may be utilized for temperature monitoring both within and beyond the core region.

Kryptonated plenum tubes and spacer wire will be included in the fuel rods irradiated in the F20 Test, which consists of a series of short time (7 to 30 day) exposures in the EBR-II. In addition, kryptonated spacer wire will be utilized in the upcoming F10C test. The tubes and sections of wire wrap will be "thermanalyzed" for operating temperature and compared with calculated values and with values obtained in a similar manner from fuel element cladding. The kryptonated wire is on hand and the program objectives are being implemented.

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- 5-3. *Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, Fortieth Quarterly Report, August-October 1971, November 1971, (GEAP-10028-40), pp. 6-2-6-9.*
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- 5-6. Swanson, K. M., et al., "The Determination of Reactor Fuel Cladding Temperature by Recording Krypton-85 Release During Post-Irradiation Heating," *Journal of Nuclear Materials* **33**, pp. 302-310, (1969).
- 5-7. Carter, G. and Colligan, J. S., *Ion-Bombardment of Solids*, Heineman, London, 1968, p. 354.

6. TASK F—FAST FLUX TESTS AND SUPPORT

Task Leader: J. D. Stephen*

6.1 GENERAL

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

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

Selected irradiation tests in thermal test reactors have been performed in support of the fast flux irradiations in EBR-II. These tests include high-risk tests, instrumented 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 rod diameters without the complications of cladding swelling.

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

6.2 SCOPING AND STATISTICAL CONFIDENCE IRRADIATIONS

6.2.1 F3B Experiment (Fuel Void Deployment and Cladding Alloy Studies)

Detailed information on the destructive examination of 6 of the F3B fuel rods has been prepared for ANL. This experiment, containing 18 encapsulated fuel rods focused primarily on the effect of fuel void deployment, has reached ~10 at.% burnup at ~16 kW/ft with ~1150°F cladding temperature (nominal**). The results of the microprobe examination of F3B fuel rod number E1H-30 are reported under Task E. A data package for resumption of irradiation of the experiment, combined with the F5 fuel rods into a single subassembly, has been forwarded to EBR-II for review. The peak target burnup for the lead F3B fuel rod is now ~16 at.%.

6.2.2 F4 Experiment (Low Density Fuel Performance)

The non-destructive examination of the F4 fuel rods is complete. The F4 experiment, containing 7 encapsulated fuel rods with fuel densities from 84 to 87% theoretical, was irradiated to ~9 at.% burnup at >16 kW/ft with ~1100°F cladding temperature (nominal**). Profilometry of the SS304 capsules of rods F4G, -H, -J, -K, -L, showed a range of diameter increases from 3.2 to 4.1 mils (0.9 to 1.1%). Complete destructive examination of two rods (F4A, F4E) at the LASL facility is planned to provide control information for two similar rods (F4G, F4J) to be included in the F20 experiment.

6.2.3 F9C Experiment (Run-To-Failure)

The continued irradiation of the F9C experiment is underway. These unencapsulated fuel rods with wire spacers have been reconstituted into two Mark E subassemblies (F9C-1/SA-143 and F9C-2/SA-144) as part of the EBR-II

*Contributors are listed in Appendix B.

**See general discussion, Subsection 6.1.

run-to-failure program. Prior to reconstitution, the fuel rods had operated at ~ 14 kW/ft with $\sim 1075^\circ\text{F}$ cladding temperature to ~ 7 at.% burnup (nominal*).

Fuel rods F9C-31 and -13 were removed for destructive examination. The metallographic examination of F9C-31 is complete. The rod was in excellent condition. F9C-13 will be examined at LASL. The fabrication and irradiation parameters for these two fuel rods are given in Table 6-1.

Table 6-1
FABRICATION PARAMETERS AND IRRADIATION CONDITIONS FOR FUEL PINS
F9C-31 AND -13

	Fuel Pin	
	F9C-31	F9C-13
Fuel		
Wt.% Pu (91% enriched in Pu-239)	24.1	25.4
U-235 enrichment, (%)	92.9	92.4
Stoichiometry, (O/M)	1.97	1.98
Fabrication process	Phys Mix	Co-ppt
Configuration	Solid pellet	Solid pellet
Average pellet density, (% Theo)	88.4	92.9
Smeared density, (% Theo)	84.3	87.4
Diametral gap range, (mils)	3.8 to 6.0	6.0 to 7.5
Cladding		
Alloy	316 SS	316 SS
Fabrication method	Seamless	Seamless
Thermal treatment	Ann.	Ann.
Wall thickness, (mils)	15	15
Outside diameter, (inch)	0.250	0.250
Irradiation Conditions		
Nominal* peak burnup, (at.%)	7.26	6.52
Nominal* peak burnup, (MWd/Te)	65,400	58,800
Nominal* average burnup, (at.%)	6.51	5.85
Nominal* average burnup, (MWd/Te)	58,600	52,700
Nominal* peak fluence, total, (10^{22} nvt)	5.47	4.91
Nominal* peak fluence, $E > 0.1$ MeV, (10^{22} nvt)	4.64	4.13
Measured peak burnup, (at.%)	6.26	
Measured peak burnup, (MWd/Te)	56,400	
Nominal* peak linear power, (kW/ft(S.o.L.))	13.4	12.5
Nominal* peak linear power, (kW/ft(E.o.L.))	12.7	11.3
Nominal* average linear power (kW/ft(S.o.L.))	12.0	11.2
Nominal* average linear power, (kW/ft(E.o.L.))	11.4	10.2
Nominal* maximum cladding temperature, ($^\circ\text{F}$)	1033	935

* See general discussion, Subsection 6.1.

6.2.4 Nondestructive Examination

The neutron radiograph of F9C-31 indicated that a highly absorbent material (presumably fission products) was present within the central void about five inches below the top of the fuel column. The rod was not sectioned in this area. The gross gamma scan of both fuel rods showed no activity in the plenum area. A peak was present at the top of the fuel column, as shown in Figure 6-1. The activity at the top of the fuel column was depressed in general. The isotopic gamma scan for Cs-137, Figure 6-2, indicates: 1) that the Cs-137 is responsible for the peak at the top of the fuel column, and 2) that the Cs-137 concentration increases toward the bottom of the fuel. The increased concentration of the Cs-137 toward the bottom of the fuel has evidently distorted the gross gamma activity profile. There were no anomalies in the isotopic scans for Zr-95, Nb-95, Cs-134, and Ru-103.

The pulsed eddy current examination of F9C-31 indicated a slight irregularity on the inside surface of the cladding at 1.6 inches above the bottom of the fuel. Subsequent metallographic examination of several planes in the area failed to reveal the cause of the signal.

Profilometry revealed a maximum cladding diameter increase of 0.0016 inch (0.64 percent). Length measurements to a precision of 0.01 inch indicated no change. No significant change in fuel column length was discernible from the neutron radiograph.

6.2.5 Metallographic Examination

No intergranular attack was observed, but one area showed what may be the beginning of intergranular attack. The reaction of the cladding with the fission products and/or fuel was predominantly in localized areas in the region of highest cladding temperature. The maximum thickness of any reaction layer was less than one mil. Alpha autoradiographs showed increased alpha activity at the edge of the center void suggesting increased Pu concentration. The difference in activity was more pronounced in the specimens near the top of the fuel column. Increased activity near the outer edges of the fuel was also observed.

Figures 6-3 and 6-4 are montages of the transverse metallographic specimens at the center of the fuel column and near the top of the fuel. Figure 6-5 is a longitudinal specimen at the top of the fuel column. A gap of several mils between the fuel and the cladding is apparent on all of the specimens. Only a few areas indicating slight cladding reaction are present. The surfaces of the central void are irregular. Figure 6-6 is typical of a diametral plane near the center of the fuel column. Figure 6-7 shows fission products and cladding reaction in the transverse section near the top of the fuel column. An area of cladding reaction where intergranular attack may just be beginning, Figure 6-8, was found on the longitudinal specimen near the top of the fuel column. Figure 6-9 shows another area of cladding reaction near what appears to be metallic fission products. The cladding etched in 10 percent oxalic acid, Figure 6-10, shows the typical general precipitate, and also the heavy precipitate behind a denuded zone at the inside surface.

The alpha autoradiographs of the two specimens near the center of the fuel column and the two near the top are shown in Figure 6-11. All show some increased alpha activity near the edges of the central void, which may be indicative of increased plutonium concentration. The changes in alpha activity are more pronounced in the specimens from the top of the fuel column which also show increased activity near the outside edges of the fuel.

Diamond pyramid hardness measurements using a 200-gram load were made on the cladding of three transverse specimens. A total of nine indentations were made on each specimen; three each near the outside, the center, and the inside. No significant difference was found across the thickness of the cladding. The axial variation followed the fluence and temperature profile. The average value of the nine measurements for the three axial specimens from the top of the fuel column to the bottom were 166, 270 and 251. The center specimen which experiences the highest neutron fluence was the hardest. The fluence of the top and bottom specimens was almost equivalent, but the temperature of the top specimen was higher. Results of the burst tests on the cladding are reported under Task L.

6.2.6 Fission Gas Analysis

The rod was punctured for fission gas collection near the top of the bottom insulating pellets. The release was almost immediate with one half of the gas released in the first minute and all of the gas collected within five minutes. The void volume was measured as 14.1cc with the plenum volume calculated as 12.9cc. The composition analyses by gas chromatography and mass spectroscopy are underway.

BOTTOM

Figure 6-1. Gross Gamma Activity Profile of the Fueled Region of Fuel Pin F9C-31

BOTTOM

Figure 6-2. Cesium-137 Gamma Activity Profile of the Fueled Region of Fuel Pin F9C-31

A812-04 POLISHED

50X REDUCED TO 25X

Figure 6-3. Transverse Cross-Section of Fuel Rod F9C-31 Near the Center of the Fuel Column

A813-04 POLISHED

50X REDUCED TO 25X

Figure 6-4. Transverse Cross-Section of Fuel Rod F9C-31 Near the Top of the Fuel Column

A815-05 POLISHED

50X REDUCED TO 10X

Figure 6-5. Longitudinal Section at Top of Fuel Column of Fuel Rod F9C-31

A812-05 POLISHED

500X

Figure 6-6. Condition of Type-316 Stainless Steel Cladding Typical of Fuel Rod F9C-31 Near the Center of the Fuel Column

A813-05 POLISHED

500X

DARKER GRAY PHASE REMAINING
AFTER ETCHING IS MOUNTING COMPOUND

A813-15 10% OXALIC ACID ETCH

500X

Figure 6-7. Cladding Reaction and Fission Products in Fuel Rod F9C-31 Near the Top of the Fuel Column

Figure 6-8. Cladding Reaction Near the Top of the Fuel Column of Rod F9C-31

Figure 6-9. Cladding Reaction Near the Top of the Fuel Column of F9C-31

Figure 6-10. Etched Type-316 Stainless Steel Cladding of Fuel Rod F9C-31

6.2.7 Burnup Analysis

The burnup calculated from chemical analysis for the neodymium isotopes (6.26 at.%) is about 14 percent lower than the nominal* burnup (7.26 at.%). This discrepancy is consistent with the findings from previous experiments in the EBR-II.

6.3 ELEVATED CLADDING TEMPERATURE IRRADIATIONS

6.3.1 F10A, F10B Experiments (Cladding Performance Near Peak LMFBR Temperatures)

Irradiation of the F10A and F10B experiments in EBR-II is underway. These experiments, each containing 19-encapsulated fuel rods designed to determine the performance of several clad alloys at high temperature, will operate at ~16 kW/ft with ~1275°F cladding temperature to >11 at.% target burnup (nominal*).

The excessive force that was required to fit the channel over the fuel rods for these experiments is now believed to have been due to galling between the triangular flow reducers and an undersized channel. The across-the-inside-flats dimensions of the J19 subassembly provided by EBR-II for the F10A fuel rods were out of specification (both under and over), based on measurements using the GE Channel Gage.

6.3.2 F11A Experiment (Unencapsulated Fuel Rods Near Peak LMFBR Cladding Temperature in a Wire Spaced Subassembly)

The F11A experiment will determine the performance of unencapsulated fuel rods with Types-304, -316, -321 and -347 stainless steel in several structural conditions. The fuel rods will be irradiated in a wire spaced subassembly at ~1275°F cladding temperatures (nominal*).

* See general discussion, 6.1.

A814-02

A815-02

BRIGHTER BACKGROUND INDICATES
RELATIVELY HIGHER ALPHA ACTIVITY

A812-02

CENTER OF FUEL COLUMN

A813-02

TOP OF FUEL COLUMN

ALL 10X

Figure 6-11. Alpha Autoradiographs of Four Specimens from Fuel Rod F9C-31

A sample weld of a Type 321SS bottom end plug, EBR-II designation 66D1, was found to be questionable upon sectioning. Examination revealed lack of fusion to a far greater degree than observed previously in the GE program. Small defects of this type were observed during fabrication of the F11A Type 321 rods. A careful re-examination of the F11A qualification welds confirmed the earlier judgment that the presence of these small zones of lack of fusion did not jeopardize the integrity of the weld. However, the extensive lack of fusion in 66D1 is of more concern. It was determined that a lack of fusion to this extent could not be detected by X-radiography. It was therefore presumed that this type of defect could also be present in the fuel rods. The rods will be replaced. A complete review of the fuel rods with Type 321 cladding in other GE experiments indicates that this problem is unique to F11A. A program is underway to determine the cause and prevent recurrence.

6.4 IRRADIATIONS TO INVESTIGATE OPERATING LIMITS

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

The F20 experiment, containing 19-encapsulated fuel rods 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.

Work has stopped on this experiment, due to the greater priority of the F9F experiment.

Dosimetry capsules will replace some of the triangular filler rods normally inserted between shroud tubes in the MK-19A subassembly. These will contain complete spectral sets involving six nonfission and four fission monitors. ORNL standard capsules will be inserted as well as wires and glass fibers containing the isotopes to be activated. Several radial positions will be used and at least one complete axial characterization will be obtained.

The spacer wire and plenum tubes have been impregnated with Kr-85 to aid determination of the subassembly operating temperature.

6.5 SUBASSEMBLY IRRADIATIONS

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

The F9E experiment will study the performance of a 37 rod, grid spaced subassembly operating at $\sim 1100^{\circ}\text{F}$ cladding temperature to >11 at.% target burnup (nominal*). This experiment was designed under Task G6 and built and irradiated under Task F.

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

The F9F experiment, with 37-fuel rods prototypical of FFTF, will test grid spacer performance under near FFTF conditions. The experiment was designed under Task G6 and is being built and irradiated under Task F.

All the fuel rods have been loaded and welded. A new, simplified procedure for xenon tagging of fuel rods was developed, calibrated and qualified. Approximately 1.5 to 2.0cc of tag gas will be injected into each rod. The procedure has been reviewed and approved by EBR-II.

The Type 316 20% cold worked inner hexagonal channel has been received from Clark and Wheeler. Work is underway to fabricate the grid spacers.

EBR-II has completed review of the compatibility of F9F with the Mark S Test Vehicle. The upper and lower shields of the Mark S subassembly have been received from ANL for trial assembly of all F9F subassembly components prior to shipment to EBR-II.

* See general discussion under 6.1.

6.6 IRRADIATIONS EVALUATION AND SUPPORT

6.6.1 Cladding Carbon Analysis

Carbon analysis of irradiated stainless steel was investigated by BMI under the Task F program. The analytical system consisted of a laser microprobe, a furnace to oxidize the CO to CO₂ and a gas chromatograph. Lower limits of detection for CO₂ were calculated based on the analysis of standard steel samples (National Bureau of Standards) containing known amounts of carbon. These calculations indicated that the chromatograph could detect as little as 5.6 μ g of carbon. However, the technique was being developed to determine carbon gradients across a cladding wall of 15 to 20 mils in thickness. Assuming a dish-shaped sample 200 microns (8 mils) diameter X 7 microns deep, it was determined that only 1.4 μ g of carbon would be vaporized. It was therefore concluded that the technique is not sensitive enough to determine carbon gradients.

6.6.2 Xenon Tag

A simplified xenon tagging procedure has been developed, calibrated and qualified. This simplification was made possible because of the availability of the xenon tag gas in pressurized cylinders. A hypodermic needle is used to extract the correct amount of xenon tag gas from the septum chamber (Figure 6-12) and inject it into the fuel rod.

The calibration test consisted of seven gas tag ampoules with varying xenon contents. These ampoules were a mockup of the F9F plenum. The results from this test are tabulated in Table 6-2. The average dilution was ~7%. The dilution is due to the small amount of helium (glove box atmosphere) remaining in the syringe before insertion in the septum chamber.

The calibration curve was used to determine the amount of tag gas to add to obtain ~1.5 ml of xenon in the finished rods. This amount of gas (1.65 ml) was inserted into five gas tag ampoules and an end plug was then welded onto the open end. The results of this qualification test are tabulated in Table 6-3.

Table 6-2
XENON GAS TAG CALIBRATION

Milliliters At STP

Gas Injection	Actual Xenon	Dilution (%)
1.0	0.95	5
1.0	0.91	9
1.5	1.38	8
2.0	1.88	6
2.0	1.93	4
2.5	2.29	8
3.0	2.74	9
Average Dilution =		7%

Table 6-3
XENON GAS TAG QUALIFICATION

Milliliters At STP

Sample	Gas Injection	Actual Xenon	Dilution (%)
1	1.65	1.55	6
2	1.65	1.56	7
3	1.65	1.57	7
4	1.65	1.40 ⁽¹⁾	15
5	1.65	1.64	1
Average =		1.54	7.2%
$S_x =$		0.098	

(1) Welded three times.

Figure 6-12. Schematic Drawing of Gas Tag System

7. TASK G-6—CORE DESIGN ANALYSIS

Task Leader: P. R. Pluta*

7.1 GENERAL

The main purpose of this task is to develop a grid-spaced fuel assembly for the fast breeder reactor. The grid-spacer concept offers potential advantage over the present wire-wrap fuel assembly in:

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

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

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

7.2 F9E DIAGNOSTIC GRID SPACER EXPERIMENT IN EBR-II

The F9E group irradiation is the first in-pile test of a grid spaced fuel assembly utilizing the wave-beam grid-spacer concept, which was conceived by General Electric. F9E employs 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 to increase understanding of grid performance.

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

- Full Grid
- Staggered Grid
- Full Grid Modified Profile

The F9E experiment has been shipped to EBR-II and is expected to be inserted into core location 6E3 (or an equivalent position) at the beginning of reactor run 55 in February 1972. The assembly is designed to have a mixed mean coolant outlet temperature of 1000°F during most of the irradiation. If the proposed installation of a stainless steel reflector in the core is to be carried out in the near future, the assembly orifice will be resized to give this outlet temperature.

* Contributors are listed in Appendix B.

A topical report entitled "Design Basis of the F9E Grid-Spaced LMFBR Diagnostic Assembly for Irradiation in EBR-II" has been written and is undergoing final preparation as a GEAP.

7.3 F9F DIAGNOSTIC GRID SPACER EXPERIMENT IN EBR-II

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

The F9F assembly is designed primarily to determine:

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

In the F9E design, the principal modeling variation was to substitute annealed steel for cold-worked steel to obtain high relative rod-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 rods and the channel to cool the channel and obtain high beam-to-channel and rod-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 rods, 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 Subsection 7.4). All of the fuel rods 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.

The detailed fabrication drawings of F9F were completed, reviewed and the drawings released for fabrication. Refer to Subsection 6.5 for a discussion of the fabrication progress on the F9F assembly.

The specific power in EBR-II core position 5N2 (or equivalent), which is the design basis position for F9F, will increase ~8% after the proposed installation of the stainless steel reflector. If this change is to be made in the near future, the F9F orifice design will be changed such that the design basis mixed mean coolant outlet temperature (1012°F) is preserved. The design basis core location would not be changed.

Figure 7-1 (GE drawing 816E194) shows the complete F9F assembly. The fuel rod lower support was modified from that previously used to support the F9E bundle; the new support is compatible with the EBR-II irradiation subassembly MK-S1. Parts numbered 7 and 8 on Figure 7-1 replace the grid bar fuel rod support assembly used for F9E; the fuel rod lower end plug is also changed. The fuel rod end plug is engaged in a tapered hexagonal opening in the fuel rod support plate, positioning the end plug and providing an anti-rotation restraint. The hold-down comb, part number 8, engages the 37 rods, restricting the axial motion of the rods. Removal of the hold-down comb allows the removal of any fuel rod without removing and dismantling the inner channel, which contains the grid-spaced subassembly. The hold-down comb also secures the inner channel subassembly to the total assembly, allowing the inner channel subassembly to be attached to or separated from the total assembly without welds.

7.4 FFTF GRID-SPACED DRIVER FUEL ASSEMBLY

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

Design analysis of the reference FFTF grid spaced assembly continued. An important technical exchange between BRD, HEDL and ARD personnel on the aspects of the reference FTR wire-wrap driver fuel assembly design which

relates to the GE FTR grid-spaced driver assembly was held at Richland, Washington. This exchange set the foundation for the next phase of engineering analysis on the grid-spaced assembly.

7.5 SODIUM FLOW TEST

A six month sodium flow test is being conducted in the General Electric Breeder Test Facility Control Drive Test Loop to evaluate the endurance characteristics of the reference 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-rod dummy fuel elements positioned in series. The FFTF shield/orifice region has been simulated for both assemblies to generate prototypical inlet turbulence in each bundle. One 37-rod assembly utilizes staggered grids, the reference for the GE grid-spaced FFTF driver assembly. The other 37-rod assembly utilizes full grids, the backup concept for the FFTF driver fuel assembly.

The prime experimental objectives of this endurance test are to:

- a. Establish 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 cladding-grid interface.
- d. Obtain information that supports the F9E and F9F EBR-II irradiation experiments.

The secondary objectives are to determine:

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

A complete detailed description of the test parameters and test assemblies has previously been reported in the 40th Quarterly Report (GEAP 10028-40).

The test is scheduled for approximately 4000 hours of operation in sodium at 1000°F with a peak coolant 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.

Through the end of the quarter, approximately 2200 hours of operation at rated temperature and flow have been logged. The test continues to operate successfully in an uninterrupted manner. The loop pressure drop, measured continuously, has remained constant and is close to its predicted value.

At the present exposure rate, the 4000-hour test goal will be reached during April 1972.

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.

Flow-induced vibration of fuel rods using full grid spacers is characterized by a known set of support conditions, which is, in a relative sense, not difficult to analyze because of the numerous number of predictors who have in the past investigated this type of vibration. The reference staggered grid arrangement, however, represents a much more complex analytical task since each grid supports the rod on only one side at any given elevation. In many respects, the support supplied by the staggered grid support system is similar to that of a wire-wrap bundle where the degree of support is determined by rod flexure (rod bow).

In order to determine the flow-induced vibration response in the reference FFTF fuel bundle a two-point test program is in progress:

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

7.6.2 Single-Rod Tests

Single-rod flow tests have been conducted this quarter to provide preliminary vibrational information needed for the 217-rod bundle test. The single-rod flow tests have provided measurements of vibrational amplitudes and frequencies as functions of flow velocity for rods supported by staggered grids. This work will permit vibration sensors to be positioned at locations and in orientations corresponding to maximum displacements within the bundle. The single-rod tests have also provided an opportunity to use the accelerometers required to measure the vibrations of interior rods and to verify the operation of these accelerometers.

The single-rod test section consists of a cylindrical tube of plexiglass inside which the rod 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 beams, 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 rod is supported on the beams and fixed by screw supports on one end. A cylindrical accelerometer and copper pellets (to mock up the fuel mass) are mounted inside the rod. The rod 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 have been achieved, corresponding to test section velocities in excess of the maximum FFTF velocity.

The natural frequency of rod vibration with simulated staggered grid spacers on 3-inch spacings has been measured as 1250 ± 100 Hz. This is in excellent agreement with the value of 1243 Hz calculated for a rod simply supported at 3-inch intervals. This means that a staggered-grid spacer system induces the same fundamental vibrational mode pattern as a full-grid spacer system with the same spacing and that the staggered grid-to-rod contacts are true nodal points. The frequency of vibration was found to be identical at all flow rates and equal to that obtained by mechanically shocking the test channel. This is in agreement with theoretical calculations.

The vibrational amplitude was found to have a strong angular dependence. The maximum amplitude is in the plane of the joints between the two grid beams. The vibration at 90 degrees to this orientation is significantly less (see Figure 7-2).

Vibrational amplitudes were measured, for a number of flow velocities at the midspan location between grids 5 and 6, or 18 inches from the test section entrance. These are summarized in the following table and compared to the theoretical predictions of Reavis⁽⁷⁻¹⁾ and Chen-Weber⁽⁷⁻²⁾ for rods simply supported at 3-inch spans. The agreement with the Reavis correlation is excellent. It should be noted that the Reavis correlation is the reference correlation used for design of GE grid assemblies.

Flow Velocity ft/sec	Average Peak Amplitude		
	Experimental	Reavis Predictions	Chen-Weber Predictions
10	0.135×10^{-6} in.	0.1×10^{-6} in.	5.8×10^{-6} in.
16	0.325×10^{-6}	0.4×10^{-6}	14.6×10^{-6}
20	0.643×10^{-6}	0.7×10^{-6}	22.8×10^{-6}
22	0.700×10^{-6}	0.9×10^{-6}	27.5×10^{-6}

Figure 7-2. Angular Dependence of Vibrational Amplitude

The axial position of the accelerometer was shifted to grid midspan locations near the inlet and outlet of the test channel. Preliminary data reduction indicates an entrance region effect with vibrational amplitudes being greater near the test channel inlet. However, the effect is not as large as that predicted by Reavis.

A spectrum analyzer was used to investigate the detailed nature of the rod vibration. Figure 7-3 is a spectral plot for the grid 5-6 midspan location in the maximum amplitude angular orientation. It is observed that the frequency peak at 120 Hz is quite clear and pronounced; there are no other significant peaks.

The peak amplitudes measured for the staggered grids on 3-inch spacings are about one-fourth the amplitudes predicted for full grid systems with 6-inch spacings. Even these larger vibrations were small enough to indicate no potential flow-induced vibration problems. However, it must be verified that the rods in the full-size test bundle vibrate in a similar manner. In addition, the results of the sodium flow test must be evaluated before one can be assured there are no vibration problems associated with the staggered grid system.

7.6.3 217-Rod Flow Test

Some changes have been made in the FFTF flow test assembly as the result of the single-rod flow tests. The fixed accelerometer previously located between the third and fourth full grids and the proximity sensor between the sixth and seventh staggered grids have been interchanged. This was done because the expected vibrations in the staggered grid region are so low (based on the single-rod results) that they can not be accurately measured with the proximity sensors. In addition, care will be taken with the angular alignment of the accelerometers so that their orientation relative to the grid beams will always be known.

A pressure transducer will be installed flush with the channel wall between the first staggered grid support and the fuel rod tip. The purpose of this device will be to measure the static pressure fluctuations occurring in the bundle inlet region. In a very basic sense this quantity can be interpreted as a measure of how "noisy" the flow channel is. A

comparison will be made of the pressure fluctuations occurring with and without the fuel assembly nose piece to see what effect on flow "noise" this assembly has.

A spectrum analysis of the pressure fluctuations will be made and will be compared with the spectrum traces of the fuel rod vibrations to verify the strong correlation expected between the two values.

REFERENCES

- 7-1. Reavis, J. R., "Vibration Correlation for Maximum Fuel-Element Displacement in Parallel Turbulent Flow," *Nuclear Science and Engineering* **38**, pp. 63-69, (1969).
- 7-2. Chen, Y. N. and Weber, M., "Flow-Induced Vibrations in Tube Bundle Heat Exchangers with Cross and Parallel Flow," *Flow-Induced Vibrations in Heat Exchangers*, ASME, (1970).

Figure 7-3. Acceleration versus Frequency for Single Rod Flow Test with Staggered Grids

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 Modeling

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

Work was initiated on engineering specifications for a system of models to predict post failure events for subassemblies experiencing transient overpower fuel rod failure.

Equations have been developed which will allow for the volume change of the duct or subassemblies experiencing transient overpower fuel rod failure. In addition, a qualitative system for predicting fuel motion and the extent of molten fuel-coolant mixing following fuel rod rupture has also been developed. Data from these two developments will provide bases for a quantitative model for determining the extent of mixing.

8.2.1.1 Basic Mechanism of Fuel Expulsion from the Fuel Rod

Molten fuel can be expelled from a burst fuel rod in significant amount only if there is a sustained driving force to push it out. This driving force must exist because of the large quantities of fuel present in the coolant region which was revealed by the TREAT tests. Specific tests where this occurred are the Task C-1 series (4)⁽⁸⁻¹⁾ and (5)⁽⁸⁻²⁾ and ANL "S" series tests. This sustained driving force can only come from a medium that is highly compressible. Since liquid fuel is virtually incompressible, the compressible medium can only be one of the three following alternatives:

- a. Fuel vapor,
- b. Fission gas in the rod plenum or
- c. Fission gas which is released from the fuel as it melts.

Stress analysis calculations based on alternative c. above have shown that the molten fuel and fission gas pressure developed prior to cladding failure is high enough for the molten fuel temperature to be at least 6800°F before significant fuel vapor is formed. Thus, for moderate transients, as in the Task C series 4 tests, it is doubtful that fuel vapor is the compressible medium. For fast and large transients, fuel vapor may be formed due to very high fuel temperatures, and therefore be an additional compressible medium. Also, after failure the fuel rod pressure may fall rapidly enough that some fuel vapor may be formed. The rod plenum fission gas could provide a driving force if there is a clear connecting path between the plenum and the molten fuel zone. The gas must reach the molten fuel surface so that it can apply pressure to the fuel. The possible path, as has been proposed, is down the gap between fuel and cladding and horizontally inward between two pellets to the molten fuel surface; or, if there is an annular axial blanket and an extensive central void, directly down to the molten fuel surface. The following four points suggest that no such path exists. Firstly, owing to the volume increase of the fuel as it changes phase (10%) and as its temperature increases, the fuel is in intimate contact with the cladding; therefore, any flow down the fuel-to-cladding gap is eliminated because there is no gap. Secondly, post-test fuel examination has shown that the gap between the pellets has effectively disappeared, thus eliminating that flow path. Thirdly, post-test examination of fuel tested to destruction in TREAT has shown that fuel is missing from the region below the burst point. This denies the existence of a gas flow path between

* Contributors are listed in Appendix B.

the upper plenum and the molten fuel because such a path would lead to expulsion of fuel in the region above the burst point only. Finally, stress analysis shows that the molten fuel pressure must be sufficient to cause cladding yielding. This pressure is always greater than the plenum pressure in every fuel rod design. This last point was demonstrated experimentally in the C5B⁽⁸⁻²⁾ TREAT test for the special case of an annular blanket. In this test the molten fuel moved rapidly upward into the annular blanket and toward the gas plenum. The rod did not burst but did radially deform considerably. It appears, therefore, that in general the primary sustained driving force must be expanding fission gas which is mixed with the molten fuel, central void gas, and fuel vapor when temperature and pressure conditions permit.

8.2.1.2 Fuel and Rod Motions in the Coolant Channel

As the fuel is ejected from the rod two motions are possible, these being dependent on the type of cladding breach. If the cladding breach is transverse or circumferential, the molten fuel and gas produce an axial thrust on the two fuel rod parts. If the fuel rod hold down or clamping design is insufficient to withstand this thrust, the two fuel rod parts will begin to move apart until the thrust is exhausted or the pressure in the rod parts drops sufficiently for the end restraints to slow the motion. This phenomenon has been observed in the two FFTF transient overpower tests.⁽⁸⁻³⁾

If the cladding breach is longitudinal, then the thrust is at right angles to the rod hold down mechanism, and therefore causes no axial force on the rod. The molten-fuel fission gas mixture is then ejected as a jet into a fluid stream moving at right angles to it. This second motion is similar to the motion postulated for fast fission gas release from a ruptured gas plenum. Because the initial expulsion process and pressure pulse is very rapid (about 0.002 to 0.004 sec), there should be little heat transfer between the fuel and coolant during this time. Thus, this process can be treated from pure momentum and continuity considerations.

8.2.1.3 Differences Between Tests and Reactor Conditions

Most TREAT test capsules have a large heat sink used for temperature limitation purposes. That is, heat from expelled fuel is stored in the heat sink and dissipated slowly to the surrounding containment. The rapid heat removal from the expelled molten fuel may permit it to partially freeze and partially plug the cladding breach. This could slow the molten fuel expulsion rate considerably. In the C4B⁽⁸⁻⁴⁾ test, the estimated fuel expulsion time exceeded 0.1 sec. This is approximately 10 times slower than the expulsion time assuming fully molten fuel. Thus, ignoring heat transfer between molten fuel and containment walls may be adequate for reactor calculations but inadequate for test analysis. Heat transfer between molten fuel and containing walls must therefore be accounted for as an extension to the momentum models.

8.2.1.4 Fuel-To-Coolant Heat Transfer

During the fuel expulsion into the coolant stream, mixing might be expected between the molten fuel, fission gas, and sodium. This could result in a fuel-coolant interaction zone as defined by Cho⁽⁸⁻⁵⁾. Correct energy release prediction by Cho's model depends on the correct selection of the mixing zone geometry. The arbitrary selection of a mixing zone geometry may be removed by considering the molten fuel coolant motions. Taylor's^(8-6, 8-7, 8-8) theory on the interfacial stability between two liquids, one accelerating the other, states that if the heavier fluid accelerates the lighter fluid then their interface is stable and mixing does not occur. Conversely, if the lighter fluid accelerates the heavier fluid, the two will mix. Using the theory together with expected fuel rod failure conditions, fuel coolant interactions can be qualitatively estimated. The results are summarized in Figure 8-1. The initial failure conditions can be subdivided into coherent and non-coherent fuel rod failures in a reactor subassembly or a transient overpower test capsule. In this discussion, coherence or incoherence can apply to both space and time. For instance, fuel rods in a subassembly could fail at the same axial location but at different times; this is one incoherent form. Molten fuel temperature and pressure adjacent to the cladding breach allow further classification since these two parameters determine the initial fuel vapor formation. If the coolant pressure adjacent to the cladding breach is 75 psia, then the fuel temperature must be approximately 6800°F for fuel vapor flashing to occur. Using these parameters and Taylor's theory, four basic accident sequences can be deduced. Figure 8-1 has been subdivided into eight classifications to illustrate the difference between reactor conditions and TREAT tests.

Figure 8-1. Molten Fuel-Coolant Mixing Logic

8.2.1.5 Accident Classification

(1.) Coherent Rod Failure with No Fuel Vapor Flashing [(Chart Classifications (1) and (2) in Figure 8-1)]

If all the fuel rods fail simultaneously with respect to axial position and time, then the expelled molten fuel motion ultimately becomes axial. The heavy molten fuel-fission gas mixture is therefore accelerating the lighter coolant, the density ratio varying with the fission gas void fraction. Thus, the interface between the fuel and coolant is effectively stable, resulting in negligible mixing of fuel and coolant. Fuel to coolant heat transfer takes place at the interface only with stable coolant film boiling being the most likely energy transfer mechanism. If a large radial heat sink is present, porous breach plugging is possible followed by a delayed fuel expulsion.

(2.) Coherent Rod Failure with Fuel Vapor Flashing [(Chart Classification (3) and (4) in Figure 8-1)]

If the fuel vapor flashing occurs as the first fuel is expelled from the rods, then the lighter fuel vapor is axially accelerating the heavier coolant. For these conditions, the vapor will mix with the coolant and exchange heat very rapidly. The rapid heat exchange rate produces coolant vapor and a rapid pressure rise adjacent to the cladding breach area. This slows the fuel expulsion and prevents further fuel vapor formation. The remaining fuel motion and fuel-to-coolant heat transfer follows as in (1.) above. Fuel cooling is more rapid if there is a large radial heat transfer rate due to a large radial heat sink. Partial breach plugging is still possible after the fuel vapor is condensed to liquid.

(3.) Incoherent Rod Failures without Fuel Vapor Flashing [(Chart Classifications (5) and (6) in Figure 8-1)]

With incoherent failures, molten fuel expulsion from the rods can occur at different times and at different axial locations. This may lead to a fuel-coolant-fuel "sandwich." In this case, the lower fuel slug accelerates the sodium slug which then accelerates the upper fuel slug. Thus, molten fuel and coolant mixing occurs between the upper fuel slug and the sodium beneath. Sodium vaporizes causing a pressure rise, and greatly reduces the fuel expulsion rate. The lower fuel slug then accelerates the upper mixed slug, and the motion then proceeds as in (1.) above. A large radial heat sink could cause partial breach plugging and slow the fuel expulsion. The intermediate sodium slug could then be heated from the fuel slug surfaces, vaporize, pressurize, and possibly remove the blockage. The fuel expulsion would then resume.

(4.) Incoherent Rod Failures with Fuel Vapor Flashing [(Chart Classification (7) and (8) in Figure 8-1)]

Assuming the same fuel-coolant-fuel "sandwich," as in (3.), the lower, lighter fuel vapor slug accelerates the heavier sodium. Mixing and a fuel vapor/coolant interaction takes place between the lower fuel slug and the sodium slug. This leads to sodium vapor/fuel interaction, a pressure rise, and a fuel expulsion slow down. The remaining expulsion will proceed as in (3.) if the pressure does not fall below the fuel vapor flash point. A large radial heat sink would enhance the cooling rate. Partial breach plugging is possible, but unlikely due to the larger pressure forces generated in the vapor coolant interaction.

8.2.1.6 Axial Flow Path Plugging

When the fuel expulsion is nearly complete, the molten fuel column may reach into the blanket area. The molten fuel in the blanket area may then freeze by giving up heat to the cladding and blanket UO_2 . This could plug the axial flow path and cause a pressure rise in the molten fuel below through continued heat generation. The pressure rise could either damage the bundle containment walls or blow away the blockage. This mechanism might explain the ANL S-4⁽⁸⁻⁹⁾ delayed pressure pulse.

8.2.1.7 Pressure Rise Correction and Bundle Wall Failure

During the expulsion process, allowance must be made for radial movement of the bundle wall due to pressure forces in the molten fuel-gas bubble and sodium. This is achieved by a stress analysis of the containment wall coupled with a constituent mass balance. Such an allowance has two purposes: a more accurate pressure pulse prediction, and a failure prediction for the bundle walls in reactor calculations.

8.2.1.8 Current Status

To date, the equations for two phase compressible flow have been developed and programmed. Some equations for the pressure rise correction have been developed. The plastic range analysis for hexagonal shaped bundle wall is in the process of development.

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/BUBBLE code has been extended to model various pipe break accidents for loop systems (pump in cold leg, or pump in hot leg). In addition, new transient pump characteristic parameters have been added.

Pipe breaks can be modeled at the following reactor primary system locations:

1. Pump outlet
2. Core inlet
3. Core outlet
4. IHX outlet (available only for case with pump in cold leg).

The location of the break is specified by the input parameter IPB which is set to a value of 1 to 4 corresponding to the above list. (Omittance of IPB or setting IPB to zero bypasses the pipe break option.) Three additional input values are used to specify the break:

POS = Pressure outside the break (lb/ft²)
 ABK = Area of break (ft²)
 CBK = Effective orifice coefficient for flow through break.

A reasonable approximation to the instantaneous velocity of flow through the break is given by Bernoulli's equation:

$$P_{BK} - P_{OS} = (1 + C_{BK}) \frac{\rho V_{BK}^2}{2g_o} \quad 8-1$$

P_{BK} = Pressure in piping at point of break
 P_{OS} = Pressure outside of break
 C_{BK} = Orifice coefficient at pipe break
 V_{BK} = Flow velocity at break

If P_{OS} and C_{BK} are input values and P_{BK} can be identified with a known pressure, the break flow velocity can be explicitly calculated:

$$V_{BK} = \frac{P_{BK} - P_{OS}}{|P_{BK} - P_{OS}|} \sqrt{\frac{|P_{BK} - P_{OS}|}{1 + C_{BK}} \frac{2g_o}{\rho}} \quad 8-2$$

The flow velocity at the break will vary with time as the local piping pressure P_{BK} varies. For a break at the reactor inlet,

$$P_{BK} = P_{PPOUT}$$

The effect of the break on the system is transmitted through the flow downstream (or upstream depending on the location of the nearest sodium surface). The equations and quantities to perturb the system properly are being revised. The previous assumption of perturbing only a single continuity equation was found to be inadequate.

The code now provides for the pressure rise of the two pumps to be specified in three ways, according to the input parameter FPI:

1. FPI = 0.0 Pumps 1 and 2 follow characteristic curve
2. FPI = 1.0 ΔP , pump 1, specified function of time
Pump 2 follows characteristic curve
3. FPI = 2.0 ΔP , pumps 1 and 2, specified function of time.

The basic pump characteristic curve is defined by the following equations:

$$\Delta P = \Delta P_{N, \text{ initial}} \left[A + B \left(\frac{V_{PN}}{V_{PN, \text{ initial}}} \right) + C \left(\frac{V_{PN}}{V_{PN, \text{ initial}}} \right)^2 \right] \quad 8-3$$

where

$$A = HS$$

$$B = \frac{FD_i^2 \left(\frac{HS}{HS_i} \right) + HS (1 - FD_i)^2}{FD_i (FD_i - 1)}$$

$$C = \frac{\frac{HS}{HS_i} - HS \left(1 - \frac{1}{FD_i} \right)}{\left(\frac{FD_i}{FD_i} \right)^2 (1 - FD_i)}$$

HS = Fraction of pump pressure head over design head at stopped flow

FD = Fraction of pump design flow at zero pressure.

New inputs to VOIDUM/BUBBLE are tables of HS, FD and ΔP as functions of time.

TIME = $t_1, t_2, t_3, \dots$

HSI = $HS_1, HS_2, HS_3, \dots$

FDI = $FD_1, FD_2, FD_3, \dots$

PRESS = $\Delta P_1, \Delta P_2, \Delta P_3, \dots$

Values at arbitrary times are determined within the code by linear interpolation. For example,

$$t_2 < t < t_3$$

$$FDI = FD_2 + (FD_3 - FD_2) \left(\frac{t - t_2}{t_3 - t_2} \right) \quad 8-4$$

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 studied quantitatively through probability modeling to form the basis for evaluating the worth of proposed research on key safety questions.

The Fortieth Quarterly Report⁽⁸⁻¹⁰⁾ describes the effect and incentives of added instrumentation in the plant protection system (PPS). Work this quarter focused on the impact of fission gas accommodation and plant protection system features on the frequency and severity of accidents caused by fuel failure propagation.

Selected results of the analysis are shown in Table 8-1. Open communication was assumed between the central void and the vent or plenum. This assumption maximizes the differences between plenum and vented fuel and is not representative of solid fuel and blanket design as discussed in Subsection 8.2.6.1 above. For these assumptions, vented fuel appears to have a significantly lower frequency of power excursions compared to plenum fuel for equal instrumentation. In particular, vented fuel with open communication eliminates the possibility of a power excursion due to fission gas voiding.

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

Debugging of the FREADM-VENUS interface is 95% completed. Data have been picked up from FREADM and successfully passed to VENUS. No anomalous behavior has been detected. A more taxing test problem is being prepared for execution through the coupled FREADM-VENUS programs and for direct execution through the unmodified version of VENUS.

A version of TOFF-II for the FREADM program has been coded and is being debugged. Compatibility with FREADM is obtained by extending TOFF-II capability to two or more fuel rods whose failure thresholds are calculated simultaneously. The maximum number of rods allowable is 17. The same TOFF-II program has also been made compatible with the heat transfer program THTE.

8.2.5 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.

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.

A preliminary PISCES IDL problem has been completed to investigate the response of fuel-sodium interaction during core disassembly. This work is intended to lead to improved methods for analyzing the hypothetical core disassembly (HCD) when there is sodium in the core. It is anticipated that estimates of the "sodium-in" HCD damage potential can be reduced using more refined models of the disassembly region. The investigations may also lead to reduced estimates of HCD damage potential in the dry core case. The methods of analysis and the information on equations of state (EOS) may also be useful in establishing disassembly thresholds for initiating core disassembly shutdown with computer codes such as VENUS.

The major input information required for the code is the equation of state of the material used. This information is input as a subroutine to the main program and, to be compatible with the main program, it must compute pressure from specific volume and internal energy. These requirements have been satisfied by using the equation of state model developed for DASA⁽⁸⁻¹¹⁾. There are 19 material parameters required by the EOS subroutine. The values used in the example problem are given in Table 8-2; they were discussed in the Fortieth Quarterly⁽⁸⁻¹⁰⁾. The major source of data for these parameters was the Liquid Metals Handbook. The values in Table 8-2 supersede similar parameter values in Table 8-4 of the Fortieth Quarterly⁽⁸⁻¹⁰⁾. The major change from the previous values is the value for the parameter "G."

Table 8-1
PROBABILITY MODELING RESULTS

	Vented Fuel With Flux Trips (PPS)	Plenum Fuel With Flux Trips (PPS)	Plenum Fuel With Flux Plus Bundle Temperature Trips (PPS)
Initiating event frequency, per year, ^(a)			
Crack failures	80	80	80
Burst failures	20	200	200
Frequency of power reduction due to fuel failures, per year	2.5	10.7	10.7
Frequency of shutdown due to single bundle boiling and meltdown, per year	0.09	2.29	0.024
Frequency of a power excursion due to fission gas voiding, per year	0	0.034	0.033
Frequency of a power excursion due to whole core boiling and meltdown per year	3×10^{-11}	2.4×10^{-9}	7.5×10^{-12}

(a) These numbers are chosen arbitrarily but are judged to be reasonable.

Figure 8-2 presents the geometry of the example problem which is a sodium column approximately the height of the sodium pool in the reactor vessel of an LMFBR. The column is assumed to be rigidly constrained in the transverse direction and at one end. Energy is added uniformly over section B-C of Figure 8-2 at a rate indicated in Figure 8-3. The energy addition rate is based on the results of a computation made with the ANL molten fuel-coolant interaction (MFCI) model described by Cho, Evans, and Wright⁽⁸⁻¹²⁾ for 200 micron particles uniformly mixed at a sodium/fuel mass ratio of 0.07. The average pressure in the heated region and the specific kinetic energy of the sodium slug C-D (Figure 8-2) are shown in Figure 8-4. These macroscopic values are consistent with each other but both the pressure and specific kinetic energy are considerably higher than expected based on using ANL MFCI models. The ANL models predict a factor of three lower pressure and a factor of 100 lower specific kinetic energy compared to this computation with PISCES IDL. These differences have not yet been explained but effort will be directed toward identifying reasons for such large differences in the two models. Differences in the sodium equations of state in the two models is the most likely source of the differences.

Figures 8-5 and 8-6 present some of the microscopic results. The pressure and displacement at the interfaces of the heated zone (B-C) are plotted. The oscillations in the pressure are not completely understood, but are thought to be the result of pressure waves in the fluid. The displacement of the interfaces B and C are consistent with the boundary conditions on either end of the sodium column. It is interesting to note the very large liquid sodium expansion. Although interface C has been displaced 200 cm, no sodium vapor has formed because the liquid expansion has been able to accommodate the volume increase in the heated zone. This result is felt to be due in part to the assumption of a flat axial energy addition profile for the heated zone.

Future effort will be directed toward some of the differences between the results of the PISCES model and the ANL MFCI model and improving the current simplistic and extremely conservative model. Additional analysis will be made to determine the importance of spatial variation in the heating rate on macroscopic parameters such as pressure and slug kinetic energy.

Table 8-2
PARAMETERS USED FOR MODELING SODIUM

Reference density at 0°C	0.9725 gm/cc
Solid density at melting temperature	0.9516 gm/cc
Liquid density at melting temperature	0.9275 gm/cc
Melting temperature	370.82°K
Boiling temperature	1154.4°K
Critical temperature	2803.0°K
Molecular weight	26.5 gm/mole
Solid enthalpy at melting temperature	0.7982 Kcal/mole
Liquid enthalpy at melting temperature	1.515 Kcal/mole
Vapor enthalpy at melting temperature	30.44 Kcal/mole
Liquid enthalpy at boiling temperature	8.644 Kcal/mole
Vapor enthalpy at boiling temperature	33.93 Kcal/mole
Solid equation-of-state coefficients:	
C	64.773 Kbar
D	149.188 Kbar
S	738.236 Kbar
G	0.0285 Kbar/cal/gm
Parameter in vapor equation-of-state (K°)	5.5
Liquid density equation coefficients:	
C1	1.8
D1	2.8

**SODIUM COLUMN
INITIAL CONDITIONS:**

TEMPERATURE = 812°K
 DENSITY = 0.825 gm/cc
 PRESSURE = 0.011 BARS
 ENERGY = 0.95 kJ/g

Figure 8-2. Example Hydrodynamic Problem of Rapid Sodium Heating

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Figure 8-3. Energy Addition Rate

Figure 8-4. Average Pressure in Heated Zone and Kinetic Energy of Sodium Slug

Figure 8-5. Pressure and Displacement History of Interface B

Figure 8-6. Pressure and Displacement History of Interface C

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

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

9.1 GENERAL

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

9.2 LIAISON

The liaison function between ANL and GE carried out under Task G-12 has continued to grow and to become more directed. The services of the GE resident engineer at ANL were sought more frequently by both GE and ANL personnel. Inquiries and requests for technical information were responded to in a routine fashion and reports and comments on recent ANL development program results of importance to current GE PA-10 analytical work were issued on a regular basis. In particular, information was transmitted on Fuel Element Failure Propagation Loop containment analysis, application to the ANL parametric Fuel/Coolant Interaction model in determining the envelope source term for closed loops and the use of reactor containment analytical tools such as REXCO-H for the design of in-pile test loops.

The TREAT test program was followed and commented upon as soon as preliminary results became available. The tests performed during this reporting period were S-8 (piston autoclave), I-3 (impact column capsule) and D-1A (repeat of D-1, release of small amounts of fuel from an enriched pellet in a fuel stack).

Test data from the E-1, E-2, E-3, H-1 and H-2 TREAT test runs were requested from ANL/RAS and have been received for GE's use to check out failure prediction models.

A complete set of Mark II-C integral TREAT test loop shop drawings, along with a component description has been received and was transmitted to Task C-2.

Progress on the R-series test vehicle design, associated hazards analysis and preliminary test plan has been reported. Summary descriptions of RAS computer program developments and evolutions were reported, covering REXCO-H, FCI, NaHAMMER, SAS and THTB. The many faceted Post Accident Heat Removal Program with its intensive out-of-pile testing program was also reported in some detail. Contacts were also established with the EBR-II Project, the Material Science Division, and the Chemical Engineering Division to obtain information relating to LMFBR core and structural design and instrumentation.

In December the resident GE engineer visited the BRD site in Sunnyvale, California. There he participated in a seminar held on a GE study on probabilistic analysis of LMFBR safety and gave an account of ANL safety-related research and development work which is addressed to the same failure sequences and which will eventually yield an independent determination of event frequencies.

While at BRD he met individually with PA-10 staff members and commented on material he obtained from ANL which had been forwarded or discussed in his periodic reports. An important objective of the visit was to determine how effective his liaison activity has been and where additional needs had developed.

9.3 PARTICIPATION IN ANL ACTIVITIES

Approximately one-half of the resident engineer's time has been spent in engineering work in support of the R-series loss-of-flow and flow blockage TREAT test series. Activities included: containment analysis for the test vehicle, formulation of design requirements, and evaluation of instrumentation needs.

* Contributors are listed in Appendix B.

10. TASK L—CLADDING DEVELOPMENT

Task Leader: H. J. Busboom*

10.1 GENERAL

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

The objectives of the cladding materials development program are:

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 core materials and alloys which are potential future candidates for fuel rod cladding and core structural materials.

10.2 MATERIALS IRRADIATION PERFORMANCE

The objective of this subtask is to provide and follow capsules used as vehicles for irradiating clad alloy specimens in EBR-II, to document the irradiation history of the capsules, and to prepare specimens for evaluation of mechanical and physical properties. A summary of this effort is shown in Table 1-4.

10.2.1 Series L-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.

Destructive examination of these capsules was completed. As reported previously⁽¹⁰⁻¹⁾, the core region specimens were severely bowed and in some cases self welded together rendering them useless for post irradiation testing. The blanket tubular specimens, however, were intact and were profiled. The profilometry showed less than 1% change in diameter of these test sections. These sections were irradiated in the temperature region 426 to 704°C (800 to 1300°F) to a fluence of 4×10^{22} n/cm², E>0.1 MeV-GE. Selected blanket specimens will be used to obtain additional clad alloy swelling data. Future data on the L-6 capsule specimens will be reported under clad alloy swelling.

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) which had been discharged from S/A X080 were reconstituted into S/A X148 for continued irradiation in Row 6 of EBR-II. The total fluences accumulated to date are 2.9×10^{22} n/cm²-GE and 8.4×10^{22} n/cm²-GE respectively.

10.2.3 Series L-16 Materials Capsules

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

* Contributors are listed in Appendix B.

10.2.3.1 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 prior to the first interim examination is scheduled for a total fluence of $3 \times 10^{22} \text{ n/cm}^2 \cdot \text{GE}$. The total fluence accumulated to date is $2.9 \times 10^{22} \text{ n/cm}^2 \cdot \text{GE}$.

10.2.3.2 L-16D Capsules

These three capsules were designed to accommodate highly irradiated ring specimens and unirradiated controls for continued irradiation to higher fluences. Selected clad specimens are being taken from the high burnup Task F fuel rods and Task L materials capsules to investigate swelling saturation. In addition, samples have been obtained from tubing after exposure to high temperature flowing sodium.

Pre-irradiation data on all of the specimens to be used in these capsules has been obtained prior to capsule assembly. This included complete dimensional measurements, immersion density determinations, metallography and transmission and scanning microscopy examinations where necessary.

10.2.3.3 Pre-Irradiated Specimens

The transmission electron microscopy (TEM) examinations of irradiated Type 316 and 347 fuel rod cladding from the F2 series of fuel rod tests have been described previously.^(10-2, 10-3) After initial sectioning and alpha-decontamination, the length of the specimens was between 1 inch and 1-1/4 inch long. Previously reported immersion density measurements of void swelling⁽¹⁰⁻³⁾ showed volume increases up to 10%. Subsequently, rings have been cut from the end nearer the bottom of the EBR-II core to provide TEM discs and samples for elastic modulus measurements. The fluence and temperature gradients along these specimens were significant and therefore further density determinations were made on each specimen prior to assembly in the L16D capsules. These data together with the original results (on the 1 to 1-1/4 inch long specimens) are listed in Table 10-1. The new density measurements on the shorter sections are generally within 2 to 3% of the original values, showing excellent agreement. All of the samples from the lowest locations in the original fuel rods (~2 inches from the bottom of the EBR-II core) show further decreases in density (i.e., increased swelling) on re-measurement after removing the lower half of each specimen. In the case of specimens F2X-D and F2G-D the new swelling values are 11 to 20% higher than the initial values. These differences are likely due to the combined effects of fluence and temperature which both increase from bottom to top of the specimen. Since the temperatures are all below the peak swelling temperature for these alloys, 500 to 550°C (932 to 1022°F), these combined effects would be expected to cause a swelling gradient in the specimens in the direction observed. A discussion of the sodium exposed specimens is presented in Subsection 10.5.1.

10.3 CLAD ALLOY SWELLING

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

10.3.1 TEM Examination of Fuel Rod Cladding Samples

Immersion density measurements of void swelling in 18 cladding specimens from seven mixed oxide experimental fuel rods irradiated in EBR-II were reported and discussed in detail previously.⁽¹⁰⁻¹⁾ Further examination of these specimens using transmission electron microscopy techniques is in progress. To date, TEM foils have been examined from rods F3B-2, E1H-26 and F5F (solution annealed Type 316 cladding), E1H-29 (13% cold worked Type 304 cladding), F5M (solution annealed Type 321 cladding) and F5Q (50% cold worked Type 316 cladding). The analysis of the micrographs is continuing.

10.3.1.1 50% Cold Worked Type 316

Immersion density measurements on specimen F5Q-B irradiated at a nominal mean mid-wall temperature of 526°C (979°F) to a fluence of $3.08 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV} \cdot \text{GE}$ indicated that the cladding had undergone a slight densification (0.07%) during irradiation. The TEM results show that this specimen is essentially free of voids. Close examination of a large foil area revealed only a single cluster of four voids. As anticipated, the microstructure contains a

Table 10-1
DENSITIES OF CLAD SECTIONS FOR RE-IRRADIATION IN L16D TEST

Sample	Cladding	Fluence 10^{22} n/cm ² E>0.1 MeV-GE	Temperature (°C) (°F)		Density ^(a) (g.cm ⁻³)	$\frac{-\Delta\rho^{(a)}}{\rho}$ (%)	Density ^(b) (g.cm ⁻³)	$\frac{-\Delta\rho^{(b)}}{\rho}$ (%)
F2G-D	347	6.40	487	909	7.4886	5.49	7.4034	6.58
F2G-H	347	7.00	526	980	7.2007	9.12	7.1842	9.34
F2G-M	347	5.50	541	1007	7.5114	5.21	7.5060	5.28
F2T-9	347	3.76	543	1009	7.7488	1.99	7.7432	2.00
F2V-C	316	6.15	471	881	7.6354	3.80	7.6253	3.93
F2V-E	316	7.03	495	923	7.3834	6.96	7.3936	6.85
F2V-F	316	7.32	508	946	7.3651	7.20	7.3605	7.26
F2V-J	316	7.32	523	975	7.3031	7.98	7.2696	8.41
F2X-D	316	5.58	460	861	7.7159	2.78	7.6902	3.10
F2X-J	316	6.56	505	941	7.4964	5.54	7.4871	5.66
F2X-M	316	5.60	516	960	7.6770	3.26	7.6763	3.27
F2H-7A	316	3.77	542	1007	7.8197	1.38	7.8208	1.36

Controls: F2G 7.9245 g.cm⁻³
 F2T 7.9014 g.cm⁻³
 F2V 7.9369 g.cm⁻³
 F2X 7.9362 g.cm⁻³
 F2H 7.9288 g.cm⁻³

(a) Mean of six measurements of original 1 to 1-1/4 in. long sections.

(b) Mean of six measurements of same specimens after removing 0.25 to 0.5 in. from lower end.

large number of dislocations indicating a high degree of retained cold work. However, the defect structure differs substantially from the unirradiated structure of an archive sample in that extensive precipitation occurred during irradiation. Electron diffraction analysis suggests that this precipitate is sigma phase.

10.3.1.2 13% Cold Worked Type 304

The sample examined (E1H-29-H) was irradiated at 593°C (1100°F) to 4.19×10^{22} n/cm², E>0.1 MeV-GE and showed a volume increase of 0.21%. Transmission electron microscopy examination of this specimen revealed a highly non-uniform void distribution (Figure 10-1). In regions of the foil showing high densities of dislocations on slip planes, virtually no voids have appeared. Whereas in the less deformed regions between these areas, considerable numbers of voids are apparent which have contributed to the overall swelling of the specimen.

10.3.1.3 Solution Annealed Type 316

Foils cut from three solution-annealed Type 316 cladding fuel rods (F3B-2, E1H-26 and F5F) have been examined. Void size and number density measurements are complete for two sections of rod F3B-2. Specimen F3B-2D2 irradiated at 468°C (875°F) to a fluence of 3.16×10^{22} n/cm², E>0.1 MeV-GE contained 7.8×10^{14} voids/cm³ with an average void diameter of 190 Å. The other specimen F3B-2G irradiated at 494°C (922°F) to 3.26×10^{22} n/cm², E>0.1 MeV-GE contained 3.5×10^{14} voids/cm³ with an average void diameter of 250 Å. These measurements are in agreement with void formation observations in solution annealed Type 304⁽¹⁰⁻⁴⁾ which showed that at a given fluence level void size increases with temperature of irradiation while the void concentration decreases. The void size distribution histograms for these specimens show a large proportion of voids with average diameters below 80 Å indicating that void nucleation is continuing to occur at these temperatures and fluence levels.

Figure 10-1. 13% Cold Worked Type-304 Fuel Cladding (E1H-29) Irradiated to $4.2 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$ at 1100°F (593°C). Swelling = $0.21\% \Delta V/V$

The section examined from rod F5F was from near the top of the fuel column where the mean mid-wall temperature was 625°C (1157°F) and the accumulated fluence $2.36 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. The void concentration is small and the voids are frequently associated with precipitate particles.

The fuel rod section E1H-26I (also solution annealed Type 316 cladding) is of high interest. This section was irradiated at 573°C (1065°F) to a fluence of $4.8 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. During irradiation 0.59% $\Delta D/D$ plastic deformation occurred⁽¹⁰⁻¹⁾. The immersion density measurement showing a 4.84% decrease was considered anomalously large and was speculated to be an effect of stress on void formation or void growth. The TEM examination results gave a void concentration of $1.4 \times 10^{14} \text{ voids/cm}^3$ with an average void diameter of 830 Å. The voids are shown in Figure 10-2 together with the extensive precipitation and dislocation formation. Direct comparison with void formation in solution annealed Type 316 irradiated in a truly unstressed condition is not possible because of the lack of data at this fluence and temperature. The closest applicable data are those obtained by HEDL from unstressed cylindrical specimens irradiated in S/A X021C in EBR-II⁽¹⁰⁻⁵⁾. These data, together with the data obtained to date from the F-2⁽¹⁰⁻³⁾ and F3A series fuel rod claddings, are shown in Figures 10-3 and 10-4, which depict the irradiation temperature dependence of void size and number density. The numbers beside the data points are the fluence ($\times 10^{22} \text{ n/cm}^2$) for $E > 0.1 \text{ MeV-GE}$. These fluences are important because it has been shown⁽¹⁰⁻⁴⁾ that during irradiation void nucleation eventually ceases and then further swelling occurs by void growth only. Thus, detailed comparisons should ideally be made only at the same fluence level. The trend of increasing void size and decreasing void concentration with increasing temperature of irradiation are readily apparent in Figures 10-3 and 10-4. Whether void morphologies in the stressed and unstressed specimens are really different is difficult to discern. Apart from the single unstressed datum point at 615°C (1140°F) the void size data (including E1H-26I) appear to show a uniform trend of increasing size with temperature. UK solution annealed Type 316 fuel cladding data⁽¹⁰⁻⁶⁾ showing 1000 to 1500 Å diameter voids at temperatures of 575 to 650°C (1065 to 1202°F) support the trend exhibited by the U.S. data. Considering these data, it is still not possible to discern unequivocally whether or not there is a stress effect on void growth at the higher irradiation temperatures.

10.3.1.4 Solution Annealed Type 321

To date only one TEM foil of solution annealed Type 321 has been examined. The foil is from specimen F5M-J irradiated at 627°C (1160°F) to $2.3 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$. The microstructure and void morphology are quite different from those in the companion solution annealed Type 316 specimen F5F-H. The Type 321 foil contains a small number of relatively large voids (700 to 800 Å in diameter) which are very non-uniformly distributed, as well as small needle-like precipitate particles which may be TiC. By contrast, a comparable solution-annealed Type 316 specimen contains a uniform distribution of smaller voids often located in conjunction with larger precipitate particles.

10.3.2 Measurements of Elastic Moduli of Irradiated Fuel Cladding

Twelve samples were selected from alpha-decontaminated fuel rod cladding sections from the F2-series fuel rod irradiations. The samples are of two cladding types (solution annealed Types 316 and 347) with accumulated fluences up to $7.3 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$ and volume increases of up to 10.0% $\Delta V/V$ as measured by immersion density. The purpose of these measurements is to determine if volume changes of this magnitude cause large changes in the elastic properties of the material. Elastic moduli have been assumed to be constant in LMFBR core designs.

Accordingly work is in progress to measure the Young's modulus of elasticity, E , of these specimens using a dynamic resonance technique. Rings 0.2 inch in length have been cut from each cladding section and have been used to demonstrate that fundamental resonant frequencies of vibration can indeed be obtained on irradiated cladding samples allowing the calculation of Young's modulus to be made. Additional room temperature measurements are in progress which, if significant modulus changes are observed, will be extended to high temperature measurements.

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 property evaluations.

→ | ← 0.1 μ NOTE: AVERAGE VOID DIAMETER = 830 $\text{\AA}$, VOID CONCENTRATION = $1.4 \times 10^{14}/\text{cm}^3$, SWELLING = 4.7% $\Delta V/V$

Figure 10-2. Solution Annealed Type-316 Fuel Cladding (E1H-26) Irradiated to $4.8 \times 10^{22} \text{ n/cm}^2$, $E > 0.1 \text{ MeV-GE}$ at 1065°F (573°C)

Figure 10-3. Void Size in Solution Annealed Type-316 versus Temperature of Irradiation

Figure 10-4. Void Concentrations in Solution Annealed Type-316 versus Temperature of Irradiation

10.4.1 Burst Testing of F3A Fuel Rods

Burst tests performed at 593°C (1100°F) have been completed on 36 three-inch long sections from six F3A fuel rods and reported previously⁽¹⁰⁻¹⁾. The sharp reduction in strength properties of both Type-304L and Type-316 stainless steels was attributed to a combination of radiation damage, annealing, and intergranular attack, both directly dependent on temperature of irradiation. Metallography has now been completed on five transverse sections selected from the F3A rods used in the burst test program. Additional burst tests have also been performed on one F9C fuel rod (F9C-31) clad with Type-316 stainless steel. The irradiation history of this rod is reported with the results from the post-irradiation examination in the Task F section of this report. The results from the F9C-31 burst tests are shown in Table 10-2.

After completion of the F3A burst testing, transverse metallographic sections were taken through the burst regions of fuel rod sections F3B-9G, -15G and -16G which were at the highest irradiation temperatures. In addition, transverse sections, -9H and -15H, from the region immediately adjacent to the -9G and 15G burst specimens, were also examined. These sections allow a comparison to be made of the fuel cladding before and after burst testing. The extent of intergranular attack seen in the five F3A metallography sections is listed in Table 10-3. The intergranular attack in the unstressed specimens -9H and -15H is manifested by a delineation of the grain boundaries in the as-polished condition, which would suggest the presence of a grain boundary phase, Figure 10-5. In some instances complete boundary separation has occurred and entire grains have been removed. In contrast the stressed specimens from the burst sections F3B-9G, -15G and -16G display considerably more grain boundary separation, indicating that the weakened boundaries have opened under stress. Section F3B-15G is shown in Figure 10-6.

Table 10-2
RESULTS FROM F9C-31 FUEL ROD BURST TESTS

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

Specimen	Number	Alloy	0.2% Yield Strength (psi)	Burst Strength (psi)	Elongation (%)	Calculated Irradiation Temperature Cladding Mean		Axial Location From Core Bottom (in.)
						(°F)	(°C)	
Control	1	316	23,810	56,700	16.0			
Control	2	316	23,950	54,600	12.0			
F9C	31-A	316	75,700	75,700	0.52	700	371	-2.5 → -0.5 Lower Insulator
F9C	31-F	316	80,700	80,700	0.48	835	446	2.5 → 5 Fuel
F9C	31-L	316	64,400	72,100	5.4	910	488	7.5 → 9.5 Fuel
F9C	31-M	316	56,500	67,400	5.9	945	507	9.5 → 11.5 Fuel
F9C	31-R	316	48,300	63,450	7.5	950	510	14 → 16.5 Upper Insulator
F9C	31-S	316	47,450	63,450	7.6	950	510	16.5 → 19 Plenum

Peak fluence of F9C-31 was 4×10^{22} n/cm², E > 0.1 MeV-GE

Examination of the fracture regions indicated that the fracture originated in an intergranular manner by opening grain boundaries already weakened by intergranular attack but that the fracture propagated in a predominantly transgranular manner, see Figure 10-6. This type of fracture would appear to be inconsistent with grain boundary embrittlement by helium during irradiation. However, in the burst test, the strain rate after fracture initiation is increased considerably with resulting rapid crack propagation which results in a transgranular fracture mode. Additional features observed during the metallographic examination are:

- A continuous intergranular-carbide precipitate was observed in fuel rod section F3B-16G (in contrast to specimens from -9G and -15G which showed discontinuous grain boundary carbide precipitates.

Figure 10-5. Metallography of Unburst Sample F3B-15H Showing Attack on Cladding Inside Surface (7 mils) and Intergranular Phase

Table 10-3
EXTENT OF INTERGRANULAR ATTACK EXISTING
IN SELECTED F3A FUEL RODS

Specimen Number	Alloy	Irradiation Temperature At Cladding (i.d.)		Percent of i.d. Circumference	Extent of Intergranular Attack	
		°F	°C		Depth (mils)	
					Average	Maximum
F3B9G	304L	1050	565	80 ^(a)	4	5
F3B9H	304L	1062	572	80 ^(a)	3.7	7.5
F3B15G	316	1059	570	100	2	7
F3B15H	316	1071	576	100	3.7	7.5
F3B16G	316	1032 ^(a)	555	10	1.5	4.8

(a) Approximately 40 of the 80% reported is characterized by the existence of the unidentified reaction phase.

Figure 10-6. Burst Test Specimen F3B-15G Illustrating Intergranular Corrosion and Cracking of Weakened Boundaries on the Inside Surface of the Tube

- b. The weld zones were not found to be associated with the burst fractures.

The results of the burst tests on both the F3A and F9C fuel rod sections have been plotted against temperature of irradiation in Figures 10-7, 10-8 and 10-9. Metallographic examination of sections from the F9C rod (see Task F, Section 6) did not reveal any appreciable intergranular attack. It was therefore considered reasonable to use the F9C-31 burst test data to establish the burst properties of irradiated Type 316 fuel cladding without intergranular attack. This fuel rod received a slightly higher fluence than the F3A rods.

Although metallographical examination of the burst sections had revealed up to 7 mils of intergranular penetration in the higher temperature regions of the fuel rods, this attack did not proportionally decrease the burst rupture strength, Figures 10-7 and 10-8. The burst strength values for specimens from the regions which exhibited intergranular attack are very similar to the values obtained from burst tests on the unattacked F9C fuel rod. With as much as 7 mils attack one would expect the strength to drop dramatically in the high temperature regions where intergranular attack has taken place. This is not the case.

The effect of the intergranular attack is more readily apparent in Figure 10-9 where ductility values ($\Delta D/D_0$) are plotted against temperature of irradiation. The ductility values for both core region and plenum region specimens can be seen to increase with increasing temperature of irradiation. Normally an improvement in ductility would be expected up to about 621°C (1150°F) due to the increased annealing of radiation damage. The burst values show this trend, except for those specimens taken from the core regions, subject to intergranular attack. The ductility values in these regions are very low particularly when compared to the F9C-31 values, and are attributed to a notch effect created by intergranular attack. From the above observations, it appears that the most serious aspect of the intergranular attack is the resultant notch effect rather than direct contribution to wall thinning. The wall thinning would proportionally reduce the burst stress to much lower values than measured in these tests. Burst test results on fuel rods at ANL⁽¹⁰⁻⁷⁾ have shown a similar trend as was observed in the F3A and F9C tests data.

10.5 SODIUM EFFECTS

10.5.1 Characterization of Stainless Steel After Long-Term Exposure to Flowing Sodium

To fully assess performance of cladding materials it is necessary to determine the combined effects of fast neutron radiation damage in conjunction with exposure to flowing sodium. Since results of exposures in reactor at temperatures which cause major and minor element compositional changes in stainless steel will not be available in the near future, an interim set of experiments will be employed. Specimens which have been pre-exposed to flowing sodium at 704°C (1300°F) will be irradiated in EBR-II to determine the effect of sodium exposure on irradiation-induced void formation.

AISI Types-316, -321 and -347 stainless steel were exposed to flowing sodium at 704°C (1300°F) for various times up to 15,726 hours. The materials were exposed in identical sodium test loops constructed of Type-304 stainless steel with the exception of heat transfer surfaces which were Type 316. The sodium flow rate was 304 to 518 cm/sec (10 to 17 feet/second) and the maximum temperature drop was 278°C (500°F). Oxygen level in the sodium was maintained at 1 to 4 ppm by cold trapping.

Cladding specimens exposed to flowing sodium on their outside surface were subjected to non-destructive examination to characterize the post-exposure condition of individual tubes. The examination included measurements on inside and outside diameter plus wall thickness and ultrasonic determination of defects.

The data generated from dimensional measurements are summarized in Table 10-4. The diameter measurements are calibrated against a known standard and accurate to ± 0.0001 inch. One trend is immediately apparent in the data. The outside diameter of the tubes decreases with increasing exposure to sodium for all materials. However, for the Type-321 and -347 stainless steels, there was no further decrease in diameter from 10,352 hours to 15,726 hours after an initial decrease of 0.9 mil for the Type 347 and 0.6 mil for the Type 321. Wall thickness measurements agreed quite well with the diameter measurements.

NOTE: BURST TESTS USING 1000 psi/min PRESSURIZATION RATE

Figure 10-7. Effect of Irradiation Temperature on the Rupture and 0.2% Yield Strength of Type-304L Stainless Steel Cladding Tested at 1100°F (593°C)

NOTE: BURST TESTS USING 1000 psi/min PRESSURIZATION RATE

Figure 10-8. Effect of Irradiation Temperature on the Rupture and 0.2% Yield Strength of Type-316 Stainless Steel Cladding Tested at 1100 °F (593 °C)

IRRADIATION TEMPERATURE (CLADDING MEAN)

NOTE: BURST TESTS USING 1000 psi/min PRESSURIZATION RATE

Figure 10-9. Effect of Irradiation Temperature on Ductility of Types-304L and -316 Stainless Steel Cladding Tested at 1100°F (593°C)

Table 10-4
SUMMARY OF CLADDING SPECIMEN DIMENSIONAL MEASUREMENTS

Material	Time (h)	Exposure Temperature (°F)	o.d.	i.d.	Wall Thickness		Sample
					Average	Range	
316 S Ann. ^(a)		Control	0.2502	0.2197	0.0154	0.0149-0.0159	0031
316 S Ann.	11,379	1150	0.2500	0.2196	0.0152	0.0149-0.0154	6451
316 S Ann.	10,352	1300	0.2496	0.2195	0.0151	0.0147-0.0153	5531
316 S Ann.	15,726	1300	0.2495	0.2195	0.0150	0.0148-0.0153	4231
347 S Ann.		Control	0.2501	0.2192	0.0156	0.0149-0.0161	0051
347 S Ann.	11,379	1150	0.2500	0.2200	0.0149	0.0148-0.0151	6431
347 S Ann.	10,352	1300	0.2492	0.2199	0.0147	0.0145-0.0148	5551
347 S Ann.	15,726	1300	0.2492	0.2198	0.0147	0.0142-0.0153	4251
321 S Ann.		Control	0.2498	0.2202	0.0150	0.0147-0.0151	0061
321 S Ann.	11,379	1150	0.2497	0.2203	0.0147	0.0143-0.0151	6441
321 S Ann.	10,352	1300	0.2492	0.2200	0.0144	0.0139-0.0148	5561
321 S Ann.	15,726	1300	0.2492	0.2200	0.0146	0.0141-0.0150	4261

(a) S Ann. means solution annealed

The surface finish of the tubes was changed. In general the surface appeared to be roughened after exposure to sodium at 704°F (1300°F) while there was little roughening at 621°C (1150°F). However, excluding local variations in diameter, there appeared to be consistent trends in the diameter measurements over several inches of tubing: The unexposed tubing and the tubing exposed at 621°C (1150°F) on the other hand usually varied less than ±0.00005 inch from the mean.

It was found that a meaningful examination for defects by ultrasonic testing was impossible since the equipment used did not distinguish between surface and subsurface defects. The system was calibrated to set off an alarm if a defect exceeded 10% of the tubing wall thickness based on a standard with an outside diameter and inside diameter notch. In some instances the alarm was on constantly while the probe traversed the tube. Other tubes exhibited a constant noise level which was below the alarm threshold, but this did not appear to be related to microstructure or amount of sodium exposure. Some of the tubes (i.e., Types 321 and 347) which were exposed 15,726 hours at 704°C (1300°F) and examined by microscopy were known to have cavities under the outside surface. However, ultrasonic testing indicated that the tubes were defect-free with no observable level of noise. All of the control samples were found to be free of defects.

Chemical analysis of the alloys showed that all had experienced selective removal of both major and minor alloying elements by the sodium. These changes in composition are reflected in changes in both microstructure and mechanical properties.

X-ray fluorescence analyses were performed on the surfaces of specimens exposed for times ranging from 538 to 15,726 hours. All specimens showed progressive losses of chromium, nickel, and manganese with increasing exposure time. However, the change in composition between 10,000 and 15,000 hours was very slight so it would appear that equilibrium in the transfer of elements has been reached after about 10,000 hours. During the 15,726 hours exposure the specimens lost 65 to 80% of the initial manganese, 50 to 65% of the initial chromium, and 55 to 90% of the initial nickel. Such depletion leaves an iron-rich layer which transforms to the commonly observed ferrite surface layer. An

electron microprobe analysis was also performed on the specimens which showed depletions ranging up to 130 microns below the surface. The depth of depletion was strongly dependent on alloy type⁽¹⁰⁻⁸⁾.

The effect of sodium exposure on minor alloying elements was also determined. It was found that unstabilized alloys (Types-304 and -316 stainless steel) lost carbon while the stabilized alloys acquired carbon during exposure*. Table 10-5 shows the bulk carbon content of specimens after various exposure times. There were some factors which caused run-to-run variations in the rate of carbon loss or gain which are presently unexplained; however, the general trend is apparent. Specimens of Types-304 and -316 stainless steel which were located nearer the sodium inlet (therefore purer sodium) contained only 0.005 weight per cent carbon after exposure of 15,726 hours at 704°C (1300°F). Nitrogen was also lost from the specimens nearest the sodium inlet, but farther downstream the stabilized alloys showed increases in nitrogen content at 704°C (1300°F). At 621°C (1150°F) all alloys increased in nitrogen content substantially more than at 704°C (1300°F). Substantial amounts of boron were lost from both the stabilized and unstabilized alloys, but no variation was observed in the sulfur or phosphorus content.

Evidence of changes in the chemical composition due to sodium exposure is evident in the microstructure of the alloys. Optical metallography showed that with increasing time a ferrite layer formed and increased in thickness in the Type-316 stainless steel. At long exposure times there was some pitting in the ferrite because of the loss of major alloying elements.

Table 10-5
EFFECT OF FLOWING SODIUM ON CARBON CONTENT OF STAINLESS ALLOYS

Exposure Time (h)	Temperature (°C) (°F)		Weight Percent Carbon		
			316	321	347
0	Unexposed		0.060	0.074	0.076
538	704	1300	0.054	ND ^(a)	0.083
1768	704	1300	0.058	0.087	0.097
4838	704	1300	0.039	ND ^(a)	ND ^(a)
10352	704	1300	0.036	0.210	0.075
15726	704	1300	0.013	0.160	ND ^(a)
11379	621	1150	0.057	0.079	0.085

(a) ND = Not Determined

The amount of sigma phase precipitation increased with exposure time. In the stabilized alloys a ferrite layer also formed, but gross sigma phase formation was not evident in the Type-321 stainless steel. The most significant difference between the stabilized and unstabilized alloys was the formation of subsurface intergranular cavities in the Types 321 and 347. Figure 10-10 shows a comparison of the three alloys after exposure to flowing 704°C (1300°F) sodium for 15,726 hours. The ferrite surface layer is readily apparent in the Types-316 and -321 stainless steel, but is also present in the Type 347 even though it did not etch dark enough to contrast with the austenite. The large intergranular dark areas in the stabilized alloys are believed to be cavities in the steel. The intergranular nature of the cavities is better shown by Nomarski Contrast of the as-polished surfaces of the specimens (see Figure 10-11). The grain boundaries near the surface are delineated due to alloy depletion, but there is an absence of sigma phase in this region. In the bulk of the material it may be seen there are many instances of large sigma particles formed at grain boundary triple points.

Further examinations were made to determine the nature of the cavities in the stabilized alloys. It was thought that the cavities may be the result of high concentrations of Kirkendahl porosity caused by losses of major alloy

* Type-321 stainless steel contains titanium addition as a carbide former while Type 347 contains niobium for the same purpose.

Figure 10-10. Microstructure of Stainless Steel Exposed to Flowing Sodium at 1300°F (704°C) for 15,726 Hours

Figure 10-11. Nomarski Contrast of Stainless Steel Exposed to Flowing Sodium at 1300°F (704°C) for 15,726 Hours

elements. However, when the materials were examined by thin foil transmission electron microscopy very few small voids were found. The voids found were not sufficient in size or number to account for the large cavities. In a further attempt to find microporosity in the Type-321 stainless steel, a metallographic specimen was reground and polished for examination by a scanning electron microscope (SEM). A Type 316 sample was included for comparison. Figure 10-12a shows the Type 316 sample which had been lightly etched with glyceric acid. There are extensive amounts of sigma phase visible in the micrograph and a few small pits. (The pits were quite rare and possibly a result of the metallographic preparation.) Figure 10-12b shows the Type-321 stainless steel at the same magnification and distance from the outside surface as Figure 10-12a. The large cavity is typical of the many to be found in this sample. No microporosity or precipitates were found.

Additional work was performed using the SEM to examine the surface of the Type 321 tubing which was exposed to flowing sodium. Figure 10-13 shows one of the pits which were found scattered across the surface. The pits are apparently connected to the subsurface cavities forming a network of porosity along the grain boundaries of the material. When examined in cross section, this three-dimensional network appeared to be a collection of discrete cavities. Apparently in the stabilized alloys (Types-321 and -347 stainless steel) the sodium is able to attack along the grain boundaries, dissolving whatever phase is present after long exposure at 704°C (1300°F).

Immersion density measurements were performed to determine the density of the stainless steel prior to irradiation. Density will again be measured after irradiation to determine the amount of irradiation induced swelling in the specimens. The density of each sodium-exposed specimen in L16-D was individually determined (see Table 10-6) along with unexposed control specimens. It was found that the Type-316 stainless steel slightly densified during exposure to flowing sodium at 704°C (1300°F) for 10,352 and 15,726 hours. The stabilized alloys on the other hand which had exhibited subsurface porosity decreased in density ~0.4% after 10,352 hours and ~0.8% after 15,726 hours. This corresponds to cavity volume fractions measured from the micrographs of 0.59% and 0.9% respectively. This independent measurement indicates that all of the observed change in density can be accounted for by the volume of the cavities observed in the micrographs.

Table 10-6
L16D SPECIMENS UNIRRADIATED DENSITIES

PA-15	Sample Number		Pre-Exposure Time		Density (g/cm ³)	$\Delta\rho/\rho$ (%)
	L16D Designation	Alloy	In 1300°F (704°C) Na (h)			
0031	3K	316	0 (Control)		7.9642	Control
0051	3G	347	0 (Control)		7.9318	Control
0061	3L	321	0 (Control)		7.8910	Control
5531A	3F	316	10,000		7.9561	-0.10
5531B	3O	316	10,000		7.9671	+0.04
5551A	3N	347	10,000		7.8971	-0.44
5551B	3E	347	10,000		7.9013	-0.39
5561A	3M	321	10,000		7.8579	-0.42
5561B	3D	321	10,000		7.8476	-0.55
4231A	3J	316	15,000		7.9820	+0.22
4231B	3Q	316	15,000		7.9665	+0.03
4251A	3H	347	15,000		7.8768	-0.69
4251B	3C	347	15,000		7.8662	-0.83
4261A	3I	321	15,000		7.8210	-0.89
4261B	3P	321	15,000		7.8353	-0.71

Figure 10-12. Cross Sections of Specimens Exposed to Flowing Sodium at 1300°F (704°C) for 15,726 Hours

Figure 10-13. Pit in Surface of Type-321 Stainless Steel Exposed to Flowing Sodium at 1300°F (704°C) for 15,726 Hours

10.6 MATERIALS MANAGEMENT

The objective of this subtask is to provide high quality tubing to experimenters on all tasks of PA-10 programs, and to characterize this tubing.

10.6.1 Type 321H Air-Melted Tubing

Three thousand feet of Type 321H air-melted tubing including annealed 10 percent and 20 percent cold worked tubing has been received and is undergoing incoming inspection and characterization. The check analysis of the chemistry, Table 10-7, shows that this tubing meets both ASTM-A-213 and the GE-BRD fuel clad tubing specifications used to order the tubing with one exception—a higher cobalt content, 0.13 wt%, compared to the 0.1 wt% maximum allowable by the GE specification. This difference is considered minor and was noted earlier, before the tube fabrication began. Ten percent of the tubing has been processed through non-destructive testing, including o.d., i.d. and wall thickness measurements. Thirty-eight out of the 40 tubes inspected met the dimensional requirements. Ultrasonic testing for surface defects has been postponed until a calibration procedure using a multinotch standard is developed. Mechanical property measurements, including tensile, burst, and stress-rupture testing, have been initiated on this tubing.

Table 10-7
CHEMISTRY OF TYPE 321H TUBING MATERIALS IN WEIGHT PERCENT

Element	ASTM	Vacuum Melt Number 93132		Air Melt Number M1462		
	A213-65	Ladle	Ingot Check	Ladle	Ingot Check	Tubing Check
Nickel	9 to 13	9.58	9.23	11.12	10.49	10.44-10.98
Chromium	17 to 20	17.48	17.07	17.59	17.60	17.41-17.55
Carbon	0.04 to 0.10	0.062	0.06	0.06	0.062	0.07-0.08
Titanium	>4XC, <0.6	0.50	0.46	0.51	0.46	0.44-0.49
Manganese	2.0 max	1.69	1.64	1.73	1.74	1.75-1.80
Silicon	0.75 max	0.54	0.59	0.55	0.53	0.52-0.53
Phosphorous	0.03 max	0.012	0.008	0.021	0.022	0.024-0.025
Sulphur	0.03 max	0.006	0.007	0.011	0.009	0.008
Copper	0.1 max ^(a)	0.02	0.01	0.12	0.10	0.02-0.05
Cobalt	0.1 max ^(a)	0.02	<0.01	0.08	0.13	0.13
Boron	0.003 max ^(a)	<0.001	0.000015			0.0003
Nitrogen	0.04 max ^(a)	0.011	0.018 (0.008) ^(b)		0.012	0.015 ^(b)
Nb			<0.01		<0.02	<0.02
Pb			<0.005		<0.02	<0.02
Al			<0.01		0.16	0.16-0.17
Mo			0.05	0.15	0.26	0.26-0.34
Ta			<0.01		<0.02	<0.02

(a) Per GE Specifications 22A1756 and 22A1793

(b) Determined by Leco Combustion Technique

Initial 704°C (1300°F) biaxial stress rupture results show that the Type 321H tubing processed using a 1148°C (2100°F) final anneal exhibits higher short term stress-rupture properties than similar tubing (same heat) with a 1065°C (1950°F) final anneal. Specimens given the lower temperature anneal are failing at stress levels above 9000 psi in times less than 600 hours. However, specimens with the high temperature anneal are unfailed after 600 hours at levels to 15,000 psi. In short term burst tests both materials exhibit similar ultimate burst strengths at 704°C (1300°F), ~45,000 to 47,000 psi.

10.6.2 Type 321H Vacuum-Melted Tubing

An order has been placed for 1500 feet of Type 321H vacuum-melted tubing. The chemical analysis of this heat has been checked and meets GE specifications. The tubing will be supplied in three different conditions: (1) annealed, (2) 10% cold worked, and (3) 20% cold worked; one-half of the total supplied to each of the above conditions will be solution annealed at 1204°C (2200°F) for 15 minutes prior to the last draw-anneal cycle (the last anneal being 2000°F) and the other half will be given a more conventional solution treatment of 1093°C (2000°F) prior to the last draw-anneal cycle. The basis for the higher temperature solution anneal can be found in work by White and Freedman⁽¹⁰⁻⁹⁾ who reported that complete solution of Ti and C is necessary for maximum high temperature creep strength. They further reported that for this carbon content, complete solution will not occur below 1177°C (2150°F)⁽¹⁰⁻¹⁰⁾. Current GE creep rupture tests on another heat of air-melted Type 321H are consistent with these observations.

10.6.3 NDT Development

Work has begun to develop a statistical basis for non-destructive testing of fuel cladding. Initial efforts have involved design and procurement of multiple notch standards and equipment calibration.

One multiple notch standard has been received made from 20% cold worked Type 321 stainless steel tubing of 0.250 inch diameter and 0.015 inch wall. Figure 10-14 shows the ultrasonic test response to nominal 0.001, 0.0015 and 0.002 inch deep notches which are 0.003 inch wide and 0.030 inch long with sharp corners. Both longitudinal and transverse notches for inside and outside are included. Pulse eddy current responses to the same standard are shown in Figure 10-15 where a nominal 0.0005 inch deep notch is also included.

Standards such as this will be used to establish the response of test equipment and reveal any tendency to drift.

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Figure 10-14. Ultrasonic Response to Multiple Notch Standards Type-321H Tubing

LONGITUDINAL o. d.
NOTCHES

TRANSVERSE o. d.
NOTCHES

LONGITUDINAL i. d.
NOTCHES

TRANSVERSE i. d.
NOTCHES

Figure 10-15. Pulse Eddy Current Responses to Multiple Notch Standards Type-321H Tubing

11. TASK P—FUEL ELEMENT MODELING

Task Leader: F. E. Gelhaus*

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

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

The LIFE-2 code was received and is being placed in production status through use of an IBM computer. The program has successfully been compiled and portions of the supplied sample problem have been run. As an initial step in the evaluation of the mechanical model in LIFE-2, the code will be used to model the F12P,Q (radial restraint) experiment.

The documentation of the BEHAVE-2 fuel rod performance code is being reviewed. The following description is excerpted from this documentation. Further details have been published by S. Oldberg⁽¹¹⁻¹⁾.

In the BEHAVE-2 model, the fuel is divided into two major zones as illustrated in Figure 11-1. Within the structural zone the viscosity of the fuel is within a few decades in magnitude of the viscosity of the cladding. Cracks are simulated in planes normal to the rod axis or along radially oriented planes. The fuel is assumed to have no tensile strength, and thus is assumed to crack whenever tensile stress exists normal to either cracking plane.

Inward of the structural fuel zone the fuel viscosity is negligibly small at high linear powers, and the correspondingly high fuel temperatures produce a zone of virtually constant hydrostatic pressure termed the inviscid zone. The high temperatures within the inviscid zone promote crack healing by diffusional or plastic flow processes and the assumption is made that no cracks exist there.

The inviscid zone is further subdivided into two regions. In the non-transport (outermost) region, porosity is reduced by a mechanical hot pressing process after the fuel has expanded enough to fill the central void. In the transport region, porosity above a residual 1% is eliminated by diffusion controlled processes.

The temperatures which delineate the boundaries between the structural and inviscid zones and the non-transport and transport regions are functions of the characteristic time of a particular event. For instance, for a month of steady operation, the structural-inviscid zone and the non-transport/transport region boundary temperatures are currently held at 1300°C and 1600°C respectively. For events lasting a fraction of a second, such as a scram or transient over-power condition, the temperature delineating the inviscid boundary occurs at or close to the temperature of fuel melting; the temperature delineating the transport region is in excess of the center temperature, and hence the transport region does not exist for these very rapid events.

Another assumption of the model concerns the disposition of the as-fabricated fuel-cladding gap. At the first application of a small amount of power to a fresh fuel rod, thermal stresses crack the fuel into a large number of pie-shaped segments. The exact position in space of each segment at low linear power and the start of life is indeterminate; any segment can be touching the cladding, can remain inward toward the fuel center, or can be at some intermediate radial position.

* Contributors are listed in Appendix B.

Figure 11-1. Division of Fuel into Structural and Inviscid Zones; Subdivision of Inviscid Fuel into Non-Transport and Transport Regions

BEHAVE-2 makes the assumption that the fuel segments are as far out as possible, thus producing a configuration in which there is no fuel-cladding gap. At the start of a BEHAVE-2 problem (before going to power), the fuel is translated radially outward without deformation to the inside surface of the cladding. This movement produces a pattern of radial cracks and a small central void. It should be noted that the assumption of closure of the fuel-cladding gap, made for purposes of assigning a determinate position to the fuel, does not preclude the assumption of the existence of a gap in the code's heat transfer section.

In order to permit numerical solution of the force-displacement rate equations in BEHAVE-2, the cladding and structural fuel regions are divided radially and axially into a number of rings (Figure 11-2). The assumptions of negligible shear stresses between rings (constant properties within a ring) and isotropic swelling rates permit an exact solution for the force displacement rate relations for a ring as shown in Figure 11-3. Deformations are produced in the fuel and cladding rings by a combination of elasticity, stress induced plasticity, and thermal and irradiation creep under applied loads. The matrix equation shown in Figure 11-3 relates the nodal forces F to the nodal displacement rates, U . In general, the matrix coefficients, K_{ij} , are functions of the effective strain rate and the pattern of cracking within the ring. Since strain rates and crack patterns are unknown at the start of the calculation, iteration is required for solution.

The multiplicity of fuel and cladding rings are coupled together by the direct stiffness method, maintaining radial and axial force equilibrium and displacement compatibility between rings. The artificial restriction is imposed on the structural fuel and on the cladding that no relative axial movement can occur between the rings within each of these regions. However, relative axial movement is permitted between the fuel and the cladding which can yield a stepped axial velocity profile as shown in Figure 11-2. Interposed between the fuel and cladding regions are "shear elements" which are mathematical devices analogous to a layer of viscous lubricant and which transmit shear forces between the fuel and cladding proportional to their relative axial velocity. BEHAVE-2 simulates either static or sliding friction by continually adjusting the viscosity of each shear element in response to changes in fuel-cladding normal force. A non-sliding or "stick" condition is simulated by a very high viscosity. A rigid axial fuel column restraint which is employed in some experimental pins can also be simulated by the code.

Figure 11-2. Idealization of Cladding and Structural Fuel Regions

Figure 11-3. Formulation of Force-Displacement-Rate Relations for Fuel or Cladding Ring

The inviscid fuel region interacts with the structural region as a constant pressure boundary condition (Figure 11-4). Where voidage exists within the inviscid region, the region is compressible and its pressure known at the beginning of a timestep. The inviscid region pressure may, at the option of the user, either be set equal to the plenum pressure or the fission gas pressure may be allowed to build up above the plenum pressure to reflect restriction of access of fission gas to the plenum.

Fuel movement within the inviscid region is described by a set of "flow rules." Under an expansionary potential, fuel flows inward filling up the central void. Upon exhaustion of the central void, porosity in the non-transport region is eliminated starting with the hottest innermost fuel in this region and working outward. When all porosity in the non-transport region has been eliminated, further expansion causes fuel in the innermost ring to be extruded out axially into regions having remaining available voids. Finally, when all voidage within the inviscid region has disappeared, the region assumes an incompressible nature and imposes its expansion rate on the structural region boundary.

Since the boundaries between the fuel zones and regions employed in BEHAVE-2 are isotherms, they move radially with changes in linear power. This movement necessitates a redefinition of the nodal grid with each timestep (Figure 11-5). If in the process the identity of a ring changes, the new ring's properties are calculated as a weighted average of the properties of the old rings from which it is made.

When the linear power is increased in a fuel rod simulation run, the boundary between the structural fuel zone and the non-transport region moves outward which moves fuel from the structural zone into the non-transport region. Cracks which formerly existed in the structural zone are then considered to be eliminated. However, the program currently considers the entire former crack volume to be converted into spherical porosity volume.

The model of fuel swelling presently in BEHAVE-2 treats the phenomenon as an isotropic dilation of the material proportional to burnup. The rate is currently assumed to be independent of temperature. Fission gas constrained to occupy its atomic volume at low temperatures is released from surface tension binding at temperatures above a "gas release temperature." The gas release temperature is currently set equal to the transport region boundary temperature. Porosity, cracks and the central void are assumed to be in pressure equilibrium with a fission gas plenum.

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Figure 11-4. Idealization of Inviscid Fuel Zone

Figure 11-5. Redefinition of Inter-Ring Boundaries in Fuel

APPENDIX A. PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Nos. 1 through 113 from July 1959 through December 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
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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

GEAP-10028-40

Fortieth Quarterly Report
August—October 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
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K. M. Horst
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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

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Fabrication Cost Estimate
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The Post-Irradiation Examination
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J. M. Gerhart
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K. P. Cohen, M. J. McNelly and B. Wolfe
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Plutonium Fuel Processing and Fabrication
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H. W. Alter, G. D. Collins, and E. L. Zebroski
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Comparative Study of PuC-UC and PuO_2 - UO_2
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Part I - Technical Considerations;
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P. Greebler and D. B. Sherer
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Effects of Improved Methods
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P. Greebler and E. Goldman
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