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GEAP-10028-31
AEC RESEARCH AND
DEVELOPMENT REPORT
AUGUST 1969

FAST REACTOR NO STOCK

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**SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR
DEVELOPMENT PROGRAM
THIRTY-FIRST QUARTERLY REPORT
MAY-JULY**

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

BREEDER REACTOR DEVELOPMENT OPERATION
GENERAL ELECTRIC
SUNNYVALE, CALIFORNIA

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SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

Thirty-First Quarterly Report

May - July 1969

CD Spalar for W. E. Bailey

W. E. Bailey, Project Engineer
Fast Ceramic Reactor Development Program

E. L. Zebroski

E. L. Zebroski, Manager
Sodium Reactor Technology

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Prepared for the
United States Atomic Energy Commission
Under Contract No. AT(04-3)-189
Project Agreement No. 10

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SECTION I

INTRODUCTION

The Fast Ceramic Reactor Development Program is an integrated analytical and experimental program directed toward the development of fast reactors employing ceramic fuels, with particular attention to mixed plutonium-uranium oxide. Its major objectives are:

- a. Development of a reliable, high performance fast reactor having nuclear characteristics which provide stable and safe operation, and
- b. Demonstration of low fuel cycle cost capability for such a reactor, primarily through achieving high burnup of

ceramic fuels operating at high specific power.

Progress during the period May 1 through July 1969 on the currently active tasks of this program is described in subsequent sections.

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

SECTION II

2.1 TASK B - DEFECTED FUEL BEHAVIOR

Neutron radiography of forced convection capsule B9A was completed. These radiographs showed significant fuel expansion in the top fuel pin [oxygen to metal ratio (O/M) 1.99] in the region of the cladding defect. Smaller dimensional changes were observed in the neutron radiographs of the bottom pin (O/M 1.95). Subsequent gamma scanning of the capsule indicated minor activity in the sodium annuli. This indicates that little erosion or dissolution of fuel and fission products occurred in the sodium.

Fabrication of accessory equipment required for the destructive post-irradiation examination of forced convection capsule B9A was completed. Installation of this equipment in the RML (Radioactive Materials Lab) is in progress.

An analysis of forced convection capsule capabilities with sodium inventories of 1 and 1.5 kg was completed. This analysis established the interrelationship of sodium inventory, fuel pin length, number of pins per capsule, pitch to diameter ratio, and sodium velocity.

Current Task B irradiation experiments are summarized in Table 2-1.

TABLE 2-1

TASK B—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

Designation	Purpose and Major Parameters	Design Parameters							Operating Conditions					Status
		Fuel						Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Rating (kW/ft)	Burnup (MWd/Te)		
		Diameter (inch)	Length (inch)	Density (Smear) (% TD)	Fuel-Cladding Gap (Diameter inch)	PuO ₂ (%)	O/M							
B3C	Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	88	0.003	25	2.00	Type-347 stainless steel	V-Raft GETR	1250	21	45,000	Examination completed.	
B3D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	85	0.006	25	1.97	Type-316 stainless steel	V-Raft GETR	1100	18	6,700	Stored	
B3E	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.213	23	83	0.008	25	1.98	Type-316 stainless steel	V-Raft GETR	1100	18	8,700	Stored	
B3F	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.212	23	84	0.008	25	1.98	Type-316 stainless steel	V-Raft GETR	1100	15	23,500	Examination completed.	
B4D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	84	0.007	25	1.98	Type-316 stainless steel	V-Raft GETR	1100	15	47,500	Examination completed.	
B8B	Fuel-sodium compatibility. Sodium-bonded fuel central temperature measured versus exposure time. Two pins.	0.216 0.216	2 2	80 83	0.010 0.010	25 25	2.00 1.93	Type-347 stainless steel	V-Raft GETR	800	14	14,500	Examination completed.	
B8C	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel-sodium-clad compatibility. Two pins.	0.144	2.2	86	0.006	25	1.97	Type-316 stainless steel	V-Raft GETR	1100	21	113,500	Irradiation completed	
		0.204	2.25	84	0.017*	25	1.97					57,000		
B8D	Fuel-sodium compatibility versus density and stoichiometry. Three pins.	0.214 0.214 0.214	1.75 1.75 1.75	80-88 80-88 80-88	0.0065 0.0065 0.0065	40 25 25	1.97 1.97 1.94	Type-316 stainless steel	V-Raft GETR	950	17.5	41,000 43,500 41,400	Irradiation completed.	
B9A	Defected fuel pin. Fuel swelling versus stoichiometry in flowing sodium. Two pins axially-aligned.	0.216 0.216	5 5	89 89	0.005 0.005	25 25	1.99 1.95	Type-316 stainless steel	Special V-Raft GETR	1019 971	20 20	7,300 7,300	Examination initiated.	

*Large gap required to accommodate sampling device

2.2 TASK C - SAFETY EXPERIMENTATION

Gross γ -scanning and neutrography of the first two fuel specimens (accumulated ~57,600 MWd/Te burnup in EBR-II), destined for TREAT (Transient Reactor Test Facility) transient irradiation, indicate that fuel pin integrity has been maintained throughout the EBR-II irradiation. The remaining four specimens of this test series have accumulated ~20,200 MWd/Te burnup in EBR-II.

The destructive examination of the failed specimen, C5A, was initiated. Gross cladding melting and extensive fuel

dispersal occurred with pin separation approximately midway in the fuel column. Profilometry of intact sections indicated cladding deformation of approximately 3 to 4.5 mils. The examination of the unfailed specimen, C5B, was completed. The molten fuel flow into the upper annular blanket appeared to have occurred in several stages.

The fuel pins (two powder and two pellet) for the C6 capsules were successfully filled with sodium. Pre- and post-Na-fill profilometry traces show no significant variations in diameter.

The current status of the in-pile experimental program for Task C is summarized in Table 2-2.

TABLE 2-2

TASK C—CURRENT EXPERIMENTAL TEST SERIES

		Design Parameters												TREAT- Transient Irr.		Non-Destructive Exam ⁺		
Capsule	Test Objective	Diameter (inch)	Fuel Length (inch)	Fuel		Fuel Composition				Cladding Material	Fabrication	Pre-Irradiated		Calibration	Test	Before Trans.	After Trans.	Destructive Exam.
				Pellet	Smeared	% UO ₂	% PuO ₂	% U-235	O/M			Yes	No					
C4A	Axial restraint at Fuel melting (~5000°F).	0.2159	14.23	93.3	90.4	80	20	0	1.983	316SS	C		X	C	C	C	C	C
C4B	Axial restraint similar to C4A; test to identify percent melting necessary to cause cladding failure.	0.2155	14.21	94.5	91.2	80	20	0	1.983	316SS	C		X	C	C	C	C	C
C4C	Axial restraint similar to C4A; test to observe effect of fuel density on incipient failure point identified in C4A.	0.2164	14.19	83.1	80.4	80	20	0	1.983	316SS	C		X	C	C	C	C	C
C4D	Axial restraint designed at ~3000°F - simulating average FCR fuel temperatures; test to determine if cladding deformation occurs in transient to fuel melting (~5000°F).	0.2154	14.22	83.7	80.1	80	20	0	1.983	316SS	C		X	C	C	C	C	C
C4E	Axial restraint designed to simulate the restraint of C4A after steady-state irradiation; specific test goals to be determined after nondestructive examination of pin following removal from EBR-11.	0.2159	13.46	84	80.6	72	28	92.7	1.974	316SS	C	I*		P	P	P	P	P
C4F	Identical to C4E (transient magnitude will be varied).	0.2160	13.51	84	80.5	72	28	92.7	1.975	316SS	C	I*		P	P	P	P	P
C4G	Axial restraint similar to C4A. This test (and C4H) will be compared to C4A and B to determine the effect of burnup on failure threshold.	0.2154	13.50	94.4	90.5	72	28	89.3	1.973	316SS	C	C*		P	P	C	P	P
C4H	Identical to C4G (transient magnitude will be varied).	0.2152	13.45	94.4	90.4	72	28	89.3	1.976	316SS	C	C*		P	P	C	P	P
C4I	Typical powder fuel specimen. Test to determine failure threshold relative to pellet pins.	0.2204	13.48	Powder	83.5	80	20	0	1.973	316SS	C		X	C	C	C	C	C
C4J	Identical to C4I (transient magnitude will be varied).	0.2205	13.45	Powder	83.5	80	20	0	1.973	316SS	C		X	C	C	C	C	C
C4K	This test (and C4L) will be compared to C4I and J to determine effect of burnup on failure threshold of powder fuel.	0.2205	13.50	Powder	84.1	72	28	93.0	1.972	316SS	C	I*		P	P	P	P	P
C4L	Identical to C4K (transient magnitude will be varied).	0.2206	13.47	Powder	84.2	72	28	93.0	1.972	316SS	C	I*		P	P	P	P	P
C5A	Test to study 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	100	0	12	2.000	316SS	C	C**		C	C	C	C	NC
C5B	Identical to C5A; except pressure relief provided by hole through blanket.	0.214	24	95	91	100	0	12	2.000	316SS	C	C**		C	C	C	C	C
C6A, -1	Test to evaluate transient performance of defected or sodium bonded fuel; determine 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.	0.212	24	86	80	100	0	10	2.000	316SS	I		X	P	P	P	P	P
-2		0.2203	24	Powder	80	100	0	10	2.000	316SS				P	P	P	P	P
-3		0.212	24	86	80	100	0	10	2.000	316SS								
C6B, -1	Identical to C6A (transient magnitude will be varied).	0.212	24	86	80	100	0	10	2.000	316SS	I		X	P	P	P	P	P
-2		0.2203	24	Powder	80	100	0	10	2.000	316SS								
-3		0.212	24	86	80	100	0	10	2.000	316SS								

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P - Planned
I - Initiated
NC - Nearly Completed
C - Completed

* - EBR-II fast flux, burnup goal ~50,000 MWD/Te

** - GETR Thermal flux, burnup goal ~20,000 MWD/Te

+ - Consists of x-rays, neutron radiographs, deencapsulation, cladding, diametral measurements, gamma transmission scan

2.3 TASK E - FUEL PROPERTIES AND PERFORMANCE EVALUATION

The experimental program to establish the compressive creep behavior of mixed oxide fuels was initiated. The installation of the test apparatus in the glove box was completed and fabrication of test specimens initiated.

The new radial heat flow furnace to measure thermal conductivity was installed in a glove box and preparation of test pellets initiated.

Capsules E2P and E2Q were removed from GETR (GE Test Reactor) for their second interim examination prior to reinsertion into GETR for further irradiation.

X-ray lattice parameters were measured on three unirradiated mixed oxide specimens with varying O/M ratios after exposure to sodium vapors at 760°C. Post-test fuel stoichiometries as inferred from these lattice parameters indicated that regardless of initial stoichiometry, all samples exhibited a uniform, slightly hypostoichiometric, O/M level.

A summary of Task E's current in-pile test program is shown in Table 2-3.

TABLE 2-3

TASK E—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

Capsule	Description	Design Parameters						Operating Conditions						Status
		Diameter (inch)	Fuel Length (inch)	Fuel		Composition		Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)		
				Density (% TD)	Pellet	Smeared	PuO ₂ (%)						O/M	
E1AA	Continuous in-pile thermal conductivity.	0.219	6.0	92.0	91.0	20	1.97	Type-316 stainless steel	GETR	1300	15	0	Report in process.	
E1B-1	Gap conductivity. Four pins with diametral gaps ranging from 2 to 10 mils.	0.218	1.25	95.9	92.5	20	1.98	Type-347 stainless steel	GETR	—	20	500	Report in publication.	
2		0.218	1.25	94.2	94.2	20	1.98	GETR	GETR	974	20	500		
3		0.218	1.25	96.1	85.0	20	1.98	GETR	GETR	956	20	500		
4		0.218	1.25	96.2	92.5	20	1.98	GETR	GETR	969	20	500		
E1E	Relative smeared conductivity at high burnup irradiation to	0.218	1.9	94.7	94.5	20	2.00	Type-347 stainless steel	MTR	875	20	85,000	Data analysis in progress.	
E1F	high burnup and rabbit test E1F.	0.218	1.9	94.6	94.8	20	2.00	GETR	MTR	654/875	20/28	77,000		
E1J-1	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.94	Type-316 stainless steel	GETR	1100	24	2,000	Irradiated	
E1J-2	Effect of restructuring on smeared thermal conductivity	0.217	5.0	92.0	90.5	25	1.94	Type-316 stainless steel	GETR	1100	24		Fabrication in progress.	
E1J-3	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.94	Type-316 stainless steel	GETR	1100	24		Fabrication in progress.	
E1J-4	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.97	Type-316 stainless steel	GETR	1100	24		Awaiting irradiation.	
E1J-5	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.97	Type-316 stainless steel	GETR	1100	24	2,000	Irradiated	
E1J-6	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	1.97	Type-316 stainless steel	GETR	1100	24		Fabrication in progress.	
E1J-7	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	2.00	Type-316 stainless steel	GETR	1100	24	2,000	Irradiated and examined.	
E1J-8	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	2.00	Type-316 stainless steel	GETR	1100	24		Fabrication in progress.	
E1J-9	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	90.5	25	2.00	Type-316 stainless steel	GETR	1100	24		Awaiting irradiation.	
E2C-1	High burnup. Reference fuel irradiated at near reference thermal conditions.	0.218	5	84.5	82.2	25	2.00	Type-347 stainless steel	GETR	1080	20	121,000	Report in process.	
2		0.218	5	94.3	91.4	25	2.00	GETR	GETR	1060	20	113,000		
E2D-1	High burnup. Reference fuel irradiated at near reference thermal conditions. Two pins in each capsule	0.150	5	90.1	87.5	25	2.00	Type-316A stainless steel	GETR	1040	17	125,000	Report in publication.	
2		0.150	5	91.6	88.3	25	1.96	GETR	GETR	1040	17	125,000		
E2E	Fuel swelling. Four separate pins at three power levels.	0.100	3	98.5	95.4	25	2.00	Type-304 stainless steel	MTR	785	11.6	199,000	Report in process.	
F		0.100	3	96.9	95.1	25	2.00	GETR	MTR	655	9.4	172,000		
G		0.100	3	97.0	95.2	25	2.00	GETR	MTR	560	7.9	255,000		
H		0.100	3	96.6	94.8	25	2.00	GETR	MTR	786	11.6	330,000		
E2J-1	High burnup. Annular versus solid pellet.	0.218	5	90.4	89.5	25	1.98	Type-316 stainless steel	GETR	1100	21	124,000	Report in process.	
2		0.218	5	96.0	88.5	25	1.98	GETR	GETR	1100	21	125,000		
E2P	Effect of radial restraint on fuel swelling. (15 mil tubing)	0.219	2.0	92.0	91.0	25	1.98	Type-316 stainless steel	GETR	1160	15	20,000	Interim exam in progress.	
E2Q	Effect of radial restraint on fuel swelling. (10 mil tubing)	0.229	2.0	92.0	91.0	25	1.98	Type-316 stainless steel	GETR	1160	15	20,000		
E5A-1	Powder and pellet fuel comparison at low burnup.	0.218	4.9	86.8	84.3	25	2.00	Type-347 stainless steel	GETR	1100	27	2,000	Report in review.	
2		0.218	4.9	84.1	84.1	25	2.00	GETR	GETR	1100	27	2,000		
E5B-1	Powder and pellet fuel comparison at high burnup.	0.218	4.9	85.8	84.1	25	2.00	Type-347 stainless steel	GETR	1400	20	138,000	Report in review.	
2		0.218	4.9	84.1	84.1	25	2.00	GETR	GETR	1400	20	138,000		
E5C-1	Powder, pellet, and annular fuel comparison at high burnup.	0.215	16.0	87.5	82.9	25	1.96	Type-316 stainless steel	GETR	1160	19.5	130,000	Awaiting examination.	
E5D-1	Powder, pellet, and annular fuel comparison at high burnup.	Powder	16.0	81.8	81.8	25	1.97	Type-316 stainless steel	GETR	1160	19.5	130,000	Awaiting examination.	
E5E-1	Powder, pellet, and annular fuel comparison at high burnup.	0.215 (Annular)	16.0	90.5	89.3	25	1.96	Type-316 stainless steel	GETR	1160	19.5	130,000	Awaiting examination.	
E5F	Linear fuel movement versus burnup.	0.214	10.0	92.0	87.1	25	1.98	Type-316 stainless steel	GETR	850	17	44,000	Awaiting examination.	

2.4 TASK F - INTEGRAL FAST FLUX IRRADIATIONS

The 16 unencapsulated Group 3A fuel pins were removed from the EBR-II core for an interim examination after ~30,000 MWD/Te. These are the highest burnup unencapsulated pins under test in the FCR Program.

Two experimental subassemblies were inserted into the EBR-II core; the 37 pin unencapsulated subassembly, Group F9B and the 19 pin high cladding temperature test, Group F5.

Analysis of bundle and channel clearances in the Group 9D subassembly* showed (1) the hexagonal channel did not meet specifications set by Argonne National Laboratory (ANL), (2) the bundle of wire wrapped fuel rods acts as a spring, and (3) the clearances within the subassembly indicate that from 1% to 2.85% of the cross-sectional area in the subassembly would be available to accommodate fuel pin swelling.

The current status of the Task F irradiation test program is given in Table 2-4.

*Channels are designed and built by ANL.

TABLE 2-4

SUMMARY OF TASK F FAST FLUX TESTING PROGRAM IN EBR-II

Group	Purpose and Major Parameters	S/A	Position	No. Pins	Exposure (MWd/Te)		Power kW/ft	Date In	Date Out	Fluence (nv x10 ⁻²²)	Irradiation	
					Target	Present					Finish	Status
F-0	Low power swelling rate of high density fuel at various burnup levels.	XG02	7A1	1	40K	40K	5-6	07/16/65	08/15/68		08/68	Post exam in progress
		XG03	7D1	2	60K	58.6K	5-6	07/16/65			10/69	In reactor
		XG04	7B1	2	100K	58.6K	5-6	07/16/65			05/72	In reactor
F-1	Short time performance and proof test.	XG01	4F2	6	2K	1.8K	14-16	05/06/65	05/23/65		05/65	Reported in GEAP-5570
F-2	Medium power scoping irradiations. Major parameters include smear density, stoichiometry, cladding-coolant interactions and burnup.	XG06	4E2	12	50K	51.1K	11-16	09/03/65	02/20/67		02/67	Post exam completed
		XG05	4C2	5	75K	74K	11-16	09/03/65	03/06/68		03/68	Post exam completed
		X050	4C2	4	100K	95.2K	11-16	02/23/69			05/70	In reactor
F-3A	Bare pin irradiations to evaluate fuel cladding and cladding-coolant interactions.	X040	5B2	16	100K	36.7K	15-18	08/15/68	06/30/69		03/70	Removed for interim exam
F-3B (E1H)	Density and stoichiometry effects for encapsulated fuel pins.	X027	4B3	18	80K, 120K	58.6K	13-16	11/21/67			09/70, 5/71	In reactor
F-4	Low density fuel behavior at high heat ratings.	X011	2F1	2	50K	35K	15-18	05/09/66	06/28/67		06/67	Post exam in progress
		X039	3C2	7	80K	56.9K	15-18	05/08/68	12/24/68		12/68	Discharged from EBR-II core 12/68 for interim exam
F-5	Fuel performance at peak FCR cladding temperatures.	X064	4C3	19	100K	8.6K	15-16	05/28/69			05/73	In reactor
F-6	Fuel-cladding interactions under accelerated swelling.	X010	7F3	4	50K	61.5K	7-9	03/24/66	04/24/69		04/69	Post exam in progress
F-7	Pre-irradiated axially restrained fuel prior to transient test. Main variable is fuel density.	X015	4A2	2	50K	59.8K	12-15	12/23/67	04/03/69		04/69	Post exam in progress
		X050	4C2	4	50K	20.3K	12-15	02/23/69			02/71	In reactor
F-8A	Effect of density on fuel swelling rate.	X019	6D2	7	50K	52.0K	10-11	01/13/67			07/69	In reactor
		X020	6B5	9	100K	53.0K	10-11	01/13/67			12/72	In reactor
F-8B	Effect of void deployment on fuel cladding interaction.	X036	7E1	19	50K	21.2K	6-9	05/08/68			05/72	In reactor
F-9A	High power, high burnup, unencapsulated fuel pin irradiations.	X043	4D2	37	100K	20.7K	11-16	02/20/69			05/73	In reactor
F-9B	Medium power, high burnup, unencapsulated fuel pin irradiations.	X062	6F3	37	100K	6.9K	10-13	05/23/69			08/73	In reactor
F-9C	High power, high burnup, unencapsulated fuel pin irradiations.	X056	5C2	37	100K	15.9K	13-16	04/03/69			06/73	In reactor
F-9D	Medium power, high burnup, unencapsulated fuel pin irradiations.	X058	6F1	37	100K	9.2K	8-10	04/24/69			08/73	In reactor
F-10B	Cladding performance at peak FCR cladding temperatures.	----	----	19	100K	-----	14-17	-----				Capsule fabrication in progress

2.5 TASK G - CORE NEUTRONICS, THERMAL HYDRAULICS, AND SAFETY ANALYSIS

Sets of resolved and unresolved resonance parameters for Pu-239 are being evaluated for use in the updated version of ENDF/B. Consistency has been found to be good between measurements of resolved resonances and unresolved parameters derived from the fission cross sections and alpha values up to 20 keV.

Specifications for the second version of the ENDRUN code, ENDRUN-2, have been set and code changes initiated. The TDOWN code has been used to successfully compute the ZPR-III, Assembly 48 benchmark problem.

Standard code interfaces for kinetics and shielding modules were proposed and suggestions made for modifications to the proposed standard interfaces for nuclear data processing and neutronics modules.

Check-out of the two-dimensional, transport-diffusion theory code, DOT2DB, has been completed and a user's manual written.

Both the machine language and all FORTRAN versions of the flux synthesis code BISYN have been converted to the UNIVAC 1108 computer.

A study of the effect of B₄C control rods on the Doppler coefficient in an LMFBR indicates that a homogeneous treatment under-computes the Doppler effect in fuel adjacent to the rod by approximately 10%.

The effect of mixing induced by wire-wrap on the circumferential cladding temperature distributions has been determined. An important conclusion is that interchannel mixing is an effective means of reducing the gradients around edge rods.

A simplified method was used to estimate the stresses and strains in a fuel channel and fuel pin as a result of irradiation-induced swelling.

A topical report describing the fuel swelling code and parameter study is in preparation.

A topical report covering the VELVET computer code and parameter study is being published.

The available experimental flow data for spiral wire wrapped rod bundles has been used to develop a correlation for pressure drop in wire wrapped bundles.

TABLE 2-5

SUMMARY OF TASK L MATERIALS IRRADIATED IN EBR-II

Series	Row-S/A	Capsules	Capsule Materials	Number of Specimens		Total Fluence			Projected Completion Date(b)
				Biaxial	Tensile	Target,	July 1969		
						10 ²³ nvt	10 ²³ nvt(c)	10 ²³ nvt(d)	
<u>Irradiation Completed</u>									
L-2'	4—XG06	5	One each of Types-304, -316, -321 and -347 stainless steels, and Incoloy-800.	15	80	0.38	0.34	0.25	February 1967
L-4	4—XO09	2	Type-316 stainless steel.	10	20	0.20	0.19	0.15	December 1966
L-4'	2—XO14	5	One each of Types-304, -321, and -347 stainless steels; and two of Incoloy-800.	25	50	0.20	0.17	0.10	April 1967
L-10	4—XO16	10	One each of Types -321 and -347 stainless steels; two of Type-304 stainless steel; three each of Type-316 stainless steel, and Incoloy-800.	70	60 ^(a)	0.20	0.29	0.22	October 1968
L-14	4—XO25	19	Five each of Type-316 stainless steel and Incoloy-800; four of Type-304 stainless steel; three of Type-321, two of Type-347 stainless steel.	133	114 ^(a)	0.20	0.31	0.23	December 1968
L-2I	4—XG05 & 4—XO39	1	Type-321	3	16	0.76	0.61	0.45	December 1968
<u>Irradiation in Progress</u>									
L-2	4—XG05 4—XO50	4	One each of Types-304, -316, and -347 stainless steels; and Incoloy-800.	12	64	0.76	0.63	0.41	September, 1969
L-6	2—XO18 4—XO18A	9	One each of Types -321 and -347 stainless steels; two each of Type-304 stainless steel and Incoloy-800; three of Type-316 stainless steel.	63	126 ^(a)	1.30	0.52	0.32	September 1970
L-15	4—XO39	4	All five alloys.	None	288	0.30	0.14	0.10	September, 1969
MT-3 ^(e) and 4	4—XA08	2	32 Incoloy-800, 32 Inconel-625 and 32 Hastelloy-X Tensile Samples.	None	96	1.00	0.72	0.43	December 1970

a. Includes specimens of experimental alloy of base compositions Fe, (9-14) Ni, (9-14) Cr, (0.0004 - 0.05) C.

b. Assuming EBR-II continues to operate at 45 MW with 50% load factor.

c. Fluence based on EBR-II core physics conversion factors.

d. Fluence based on flux wire analysis conversion factors.

e. Capsules from previous AEC program under contract AT(04-3)-189, P.A. 29.

2.6 TASK L - CLADDING DEVELOPMENT

Irradiation of four L-2 materials capsules is continuing in S/A XO50. The L-6 series capsules were discharged after receiving additional irradiation in S/A XO18A. They are scheduled for further irradiation to reach their target exposure. The L-15 series capsules are waiting for reinsertion into a new subassembly. The total fluences, as of June 15, 1969, are 6.3, 5.2, and 1.4×10^{22} n/cm² for the L-2, L-6 and L-15 series, respectively.

Examination of 30 pairs of tensile specimens removed from the L-10 capsules revealed that 11 pairs were stuck together. Included in these specimens were all five commercial alloys and the high purity 14Ni-14Cr-Fe alloy. Metallographic examination revealed a strong metallurgical bond had occurred in these specimens during irradiation.

The post-irradiation examination of the L-14 capsules was initiated. High resolution gamma scans show movement of tantalum markers in temperature sentinels, indicating that the highest melting temperature, 1185°F (640°C), for the Al-Ni eutectic, was exceeded during the L-14 irradiation. Gross gamma scans of selected capsules indicate about a 14% drop in fast flux (>3MeV) across the subassembly. The

19 capsules in Group L-14 have been disassembled and tensile specimens and tubular test sections were cleaned.

A new remotely controlled balance has been made available for immersion density measurements on irradiated but non-alpha contaminated materials. The density variation to be expected in as-received stainless steel tubing was determined to be less than $\pm 0.15\%$. A density survey of L-10 tubular core sections revealed little if any cladding swelling had occurred on unstrained sections.

An alpha cleaning procedure has been demonstrated: alpha contaminated fuel cladding was removed from the alpha cell, allowing nondestructive examinations of fuel cladding to be performed using existing facilities in $\beta - \gamma$ cells.

Stress rupture unit No. 1 was completed and is operational. This unit is capable of testing the stress rupture properties of 54 tubular specimens (1/4-inch diameter) simultaneously at one temperature under one of six pressures.

The assessment of vendor capability to produce consistently high quality tubing was completed with the exception of the mechanical properties evaluation. The latter work is concentrating on long-term

testing of the materials in conjunction with a similar evaluation of the effect of local wall thinning on mechanical properties of thin wall tubing.

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

SECTION III

TASK B—DEFECTED FUEL BEHAVIOR

3.1 GENERAL

The purpose of Task B is to determine analytically and experimentally the manner and consequences of propagation of fuel defects, the behavior and performance of the fuel cladding interface in intimate contact with sodium, the distribution and rate of release of fission products from high performance mixed oxide fuel, and the identity of the fission product species which can escape to the primary coolant and cover gas.

The technical feasibility of fast reactors depends on quantitative knowledge of such release and on designs to limit the consequences of such release. It appears attainable to limit long-term activity release from vented fuel to essentially Kr-85 so that primary system maintenance conditions are equivalent to (or possibly better than) nonvented designs.

The technologies of defected, vented, and sodium-bonded fuels are closely related. Although directed primarily at studying the behavior of defected fuel, Task B also has included development of vented and sodium-bonded fuels. Although vented pins are a prospective reference

design for 300 MW and 1000 MW plants, funding limitations have made it necessary to terminate this aspect of the work.

3.2 DEFECTED FUEL TESTING

3.2.1 Forced Convection Capsule (B9A)

Fabrication of equipment required for the in-cell destructive examination of forced convection capsule B9A is complete. Fabrication of this equipment for the examination was necessitated because of the increased diameter and length of capsule B9A over previous capsules examined in these facilities. The capsule transfer cask was also modified during this period to handle long capsules of the B9A type.

Forced convection capsule B9A was removed from the GETR storage canal and transferred to the NTR [Nuclear Test Reactor (GE)] for neutron radiography of the fueled section of the capsule. Evaluation of the neutron radiographs show that significant expansion of the fuel in the top fuel pin occurred in the region of the cladding defect. The fuel appeared to have expanded sufficiently to touch the inner surface of the fluted liner surrounding the

fuel pin cluster. This is a distance of approximately 0.080 inch. Small dimensional changes were observed in the neutron radiographs of the bottom pin. Figure 3-1 shows the pre- and the post-irradiation appearance of the B9A fuel pins. Exact measurement of dimensional changes on the top and bottom fuel pins will be made during the destructive examination of the capsule.

The two fuel pins in this test were fabricated to the same dimensional specifications with the top pin containing mixed oxide of 1.99 stoichiometry and the bottom pin containing fuel of 1.95 stoichiometry. The pronounced difference in swelling between the two fuel pins appears to be directly related to stoichiometry. Both pins had pre-formed defects which insured entry of sodium.

Gamma-scanning of the capsule was also completed. These scans indicated that the capsule was correctly positioned during irradiation and that the two fuel pins operated at approximately the same power (see Figure 3-2). Only minor activity was detected in the primary sodium circuit of the capsule and no major areas of plate-out or deposition of fuel or fission products were detected.

These results imply that little erosion or dissolution of fuel and fission products occurred during operation with defected fuel pins.

3.2.2 "SODA POP" Tests

Design and fabrication of a new dummy forced convection capsule (FCC) for the SODA POP tests* was initiated. Some deformation of the capsule wall occurred during the last test which necessitated the fabrication of a new and better simulation of the capsule. This new capsule is being designed to permit replacement of specific sections without complete refabrication of the entire capsule.

In addition, disassembly of the SODA POP test rig revealed that the dummy control rod guide tube was deformed and control rod motion was inhibited. A more accurate simulation of this piece (using 3/32-in. aluminum rather than 1/16-in. thick aluminum) is scheduled.

Repair of the holddown supports for the reactor pressure vessel mockup was continued. These supports were weaker than those actually used in GETR and the damage to the holddown supports noted during the last test was due to vertical forces greater than anticipated.

Three additional capsule rupture tests are planned after test equipment

*The SODA POP test configuration is described in the 27th Quarterly report, and test results are indicated in later Quarterly reports. See appendix for document numbers.

TOP PIN - O/M 1.99

BOTTOM PIN - O/M 1.95

PRE-IRRADIATION NEUTROGRAPH

DEFECT AREA

DEFECT AREA

POST-IRRADIATION NEUTROGRAPH

**FIGURE 3-1. PRE- AND POST-IRRADIATION
APPEARANCE OF B9A FUEL PINS**

NOTE: PEAKS MARKED * ARE DUE TO
ACTIVATION OF CAPSULE COMPONENTS
NOT FISSION PRODUCT DEPOSITION

GEAP-10028-31

FIGURE 3-2. GROSS GAMMA SCAN OF CAPSULE B9A

repairs are complete. All three simulated capsule rupture tests employ the same liquid metal inventory and will be performed in the same simulated reactor environment.

Table 3-1 shows the test parameters for the projected tests. These tests are necessary to establish the reproducibility of the tests and to confirm the safety of 1500 gm sodium inventories in forced convection capsules in water-cooled test reactors.

3.2.3 Forced Convection Capsule Capabilities

A study was made of forced convection capsule test capabilities versus capsule sodium inventories and flow velocities in the test section. Using the pump characteristics of the currently available and proven helical induction pump used on capsule B9A, a three pin capsule with an average sodium velocity of up to 20 fps can be built within the 1 kilogram sodium inventory limit as shown in Figure 3-3. The increase in test capabilities available by increasing the capsule sodium inventory limit to 1.5 kilograms is also shown in this figure. This

increase to 1.5 kg. makes possible seven pin testing with sodium velocities of ~16 fps and three pin testing to ~25 ft/sec.

Figure 3-4 shows the capsule test capabilities possible using a new but undemonstrated (for this small size and head-flow characteristics) annular linear induction pump (ALIP) for 1 and 1.5 kg. sodium inventories. Using the ALIP and a capsule sodium inventory of 1.5 kilograms, three pin testing with sodium velocities of ~29 fps and seven pin testing with sodium velocities to ~19 fps are possible.

3.3 FISSION PRODUCT DISTRIBUTION: Miniloop (Out-of-Pile Radioactive Fission Product and Sodium Loop)

After completion of the experimental test in the Miniloop, the fuel sample was removed from the loop and encapsulated for shipment to the NTR for neutron radiography to determine its fuel washout occurred during the test. The neutron radiographs indicated little if any bulk fuel washout during the test. Figure 3-5 shows the appearance of the test sample.

TABLE 3-1

TEST PARAMETERS OF SODA POP SERIES 4, 5, AND 6

<u>Test</u>	<u>NaK Inventory</u>	<u>NaK Temperature</u>	<u>Reaction Orifice</u>	<u>Capsule Pressure</u>	<u>Reactor Pressure</u>	<u>Core</u>
4, 5, 6	1500 gm	1200°F	1.25 in.	~60 psig	~150 psig	Simulated

FIGURE 3-3. FCC CAPABILITIES USING HELICAL INDUCTION PUMP (7 1/2 in. o.d.)

FIGURE 3-4. FCC CAPABILITIES USING ANNULAR LINEAR INDUCTION PUMP (5 in. o.d.)

FIGURE 3-5. NEUTRON RADIOGRAPH SHOWING POST TEST APPEARANCE OF MINILoop TEST SAMPLE

Prior to loop disassembly, the activity in the loop piping was measured. Figure 3-6 shows the relative activity observed in the piping after the test. All loop sections show significant amounts of activity indicating radioactive material transfer from the fuel section to the coolant sodium and/or onto the loop piping surfaces. The loop was disassembled and removed from the containment box and sections are being prepared for leaching and visual examination.

A sample of the sodium was taken prior to insertion of the fuel and analyzed for specific isotopes. Isotopic analysis showed low level alpha activity ($4.4 \times 10^{-6} \mu \text{Ci/gm}$ of sodium) and the presence of Cs-134 and Cs-137 in the sample. This contamination of the sodium is believed to have occurred during the initial loop filling operation. Oxygen analysis of additional sodium samples must, therefore, be performed in a facility capable of handling radioactive contaminated sodium.

NOTE: TEST FUEL SAMPLE REMOVED PRIOR TO ACTIVITY MEASUREMENT
VALUES NORMALIZED TO HIGHEST ACTIVITY

FIGURE 3-6. RELATIVE ACTIVITY IN MINILOOP PIPING

SECTION IV

TASK C—SAFETY EXPERIMENTATION

4.1 GENERAL

The purpose of Task C is to establish the behavior of fuel and cladding under various casualty and accident situations. Both analytical techniques and an experimental program of testing the response of fuel to severe transients are used. These investigations include a study of the performance of fuel and cladding under transient conditions as a function of axial restraint, density, performance of fuel in the presence of a defect, and comparison of the behavior of power and pelleted fuel. Additional tests are conducted to determine the potential for molten fuel movement and to observe the effect of the prototypic range of fuel and blanket lengths on movement. Future experiments involve multiple-pin assemblies to study reduced flow conditions and energy partition data needed for the analysis of large terminal accidents.

During the current contract period, a final report will be issued on transient irradiation of zero-burnup, axially-restrained fuel as a function of density and fuel form (pellet and powder). Design and fabrication of TREAT capsules for transient irradiation (axially-restrained fuel series)

of EBR-II irradiated fuel specimens will be initiated.

Experiments designed to study relocation of molten fuel during transients in pre-irradiated long pins will be completed. Fabrication of capsules for transient tests of defected or sodium-bonded fuel will be initiated. Scoping studies on out-of-pile flow blockage experiments will be performed.

4.2 AXIALLY-RESTRAINED FUEL
EXPERIMENTS4.2.1 EBR-II Irradiations

The maximum burnup of specimens C4G and C4H (EBR-II capsules F7C and F7D), following completion of EBR-II irradiation, has been adjusted to a value of ~57,600 MWd/Te based on the final EBR-II physics data describing the irradiation history of the two specimens.

EBR-II irradiation has continued for the remaining four specimens C4E, F, K, and L (EBR-II capsules F7A, B, E, and F), with an accumulated burnup of ~20,200 MWd/Te as of July 11, 1969.

4.2.2 Post EBR-II Irradiation Examinations

The F7C and F7D capsules have been gross γ -scanned, profilometered, and neutrographed. The gross γ -scans of these two capsules show no significant indication of loss of integrity of the fuel pin. A similarly located peak in the gross γ -scan of each specimen (See Figure 4-1) and adjacent typical regions of each specimen are to be spectrographically γ -scanned to differentially determine the localized composition of material causing the peaking. It is believed that the peaks represent, in both cases, a concentration of fission products, possibly a fission-product ingot located in the void formed in the center of the fuel column.

The neutrographs of the capsules show no evidence that the enclosed fuel specimens have failed or have suffered any significant localized change from the as-built geometry. A small accumulation of fuel, probably in the form of small chips, in the region of the stainless steel spacers was noted directly below the UO_2 insulator at the bottom end of the fuel column in both specimens. These chips are within the fuel specimen cladding in both cases and were introduced either during pellet loading or were formed as a result of thermal cracking of the fuel during startup or cooldown.

4.3 AXIAL FUEL RELOCATION EXPERIMENTS

The destructive examination of specimen C5A was initiated. The fuel pin was removed and rough-cut sectioning completed. Various sections from specimen C5A* are viewed in Figures 4-2 through 4-4. Some significant features of these sections are the clean surfaces of the fuel pellets (pellets in Figure 4-5 were broken during disassembly), extensive cladding melting (occurring at a definite pitch angle in some areas), and the fine particles of fuel dispersed into the coolant annulus.

Sections of the fuel pin, where the cladding had not failed, were profilometered. Using the upper blanket area as a reference base, results for an 8-inch section from the lower portion of the fuel pin (see Figure 4-5) indicated cladding deformation at ≈ 3 to 4.5 mils.

Detailed examination (metallography) of several samples was initiated. The destructive examination of C5A is expected to be completed early next quarter.

*Refer to Figure 4-5 (neutron radiograph of C5A, first shown in the 28th Quarterly) for section locations.

FIGURE 4-1. TASK C SERIES IV GROSS δ -SCANS OF THE FUEL REGION OF EBR-II
CAPSULES F7C AND F7D (CONTAINING FUEL SPECIMENS C4G AND
C4H RESPECTIVELY)

NEG. #54110
(d)

NEG. #54198
(c)

NEG. #54123
(a)

NEG. #54181
(b)

FIGURE 4-2. SPECIMEN C5A ROUGH-CUT SAMPLES

NEG. #54205
270°

NEG. #54204
180°

NEG. #54202
0°

NEG. #54203
90°

FIGURE 4-3. SPECIMEN C5A ROUGH-CUT SAMPLE, FOUR VIEWS

NEG. #54208
270°

NEG. #54210
180°

NEG. #54206
0°

NEG. #54207
90°

FIGURE 4-4. SPECIMEN C5A ROUGH-CUT SAMPLE, FOUR VIEWS

FIGURE 4-5. SPECIMEN C5A NEUTRON RADIOGRAPH WITH SAMPLE LOCATIONS

The destructive examination of specimen C5B was completed during this quarter. Initial results were reported in the 29th Quarterly. Figure 4-6 is of a longitudinal section located ≈ 5 inches above the fuel and blanket interface. The fuel plug which fills the center hole of the blanket appears to be composed of three layers: (1) a thin dense layer adjacent to the blanket i. d., (2) a thicker layer with small to medium size pores, and (3) a center core which has a spongy structure (also note the separation in this core). A large crack in one of the blanket pellets that was open to the center hole is also filled with melted fuel. At a position 8 inches above the fuel and blanket interface (total molten fuel travel was ≈ 10 inches), the fuel plug structure is very porous and the layering effect is no longer evident.

It is postulated that the "layering effect" occurred as a result of molten fuel, driven initially by expanding gas bubbles (i. e., fuel and gas slugs), leaving thin layers of fuel as the slugs traveled up the blanket. This was followed by a general movement of a molten fuel and fission gas mixture (porous structure seen in center core, Figure 4-6). The formation of a fuel and gas slug can be seen in Figure 4-7*. Note the fuel bridge across

the void and the sharp corners at the ends, presumably indicating upward movement.

The cladding was etched and examined in several areas, but no significant discrepancies were observed, except for fine intergranular cracks at the i. d. of the cladding in the area of the large metallic particles shown in Figure 4-8 and 4-9. A Beta-gamma autoradiograph of this sample shows the activity of these particles to be much less than that for the fuel. The particles may be conglomerates of metallic impurities introduced during fuel pin fabrication rather than solid fission products.

4.4 DEFECTED OR SODIUM-BONDED FUEL EXPERIMENTS

The high pressure sodium fill system installation was completed and is now operational. Four fuel pins (two powder and two pellet) have been successfully filled with sodium. Two sodium-filled pins and one gas-bonded pin were delivered to the development shop for assembly of capsule C6A. A comparison of pre- and post-Na-fill profilometry records of these pins revealed no significant differences. Two of the inner capsule assemblies have been completed and the third is nearing completion. A schematic of the C6 capsule cross section is shown in Figure 4-10. Capsule C6A is tentatively scheduled for completion in October 1969.

*Additional slugs were noted in other micrographs not included in this report.

FIGURE 4-6. SPECIMEN C5B LONGITUDINAL SAMPLE UPPER BLANKET REGION ≈ 5 inches ABOVE FUEL BLANKET INTERFACE

**FIGURE 4-7. SPECIMEN C5B LONGITUDINAL, SAMPLE SHOWING FUEL BRIDGE
≈ 1 1/2 inches BELOW FUEL/BLANKET INTERFACE**

250X, POLISHED

NEG. #A296-06

250X, POLISHED

NEG. #A296-02

FIGURE 4-8. LARGE METALLIC PARTICLES FROM SPECIMEN C5B LONGITUDINAL SAMPLE, $\approx$ 6 inches ABOVE FUEL/LOWER BLANKET INTERFACE

FIGURE 4-9. LARGE METALLIC PARTICLE FOUND IN FUEL PELLET INTERFACE BONDED TO CLADDING i.d., SPECIMEN C5B LONGITUDINAL SPECIMEN 6 inches ABOVE FUEL/LOWER BLANKET INTERFACE

FCR TREAT CAPSULE

TOP

MIDDLE

TYPICAL CROSS SECTION

BOTTOM

FIGURE 4-10. SERIES C6 CAPSULE ASSEMBLY SCHEMATIC 4-13

VAL6823

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The preliminary study to determine the feasibility of using the Series 6 multipin capsule for TREAT irradiation of EBR-II specimens has been completed. Using the C6 capsule (Figure 4-10) as the reference design, power distributions and power normalization factors for irradiation of fully enriched UO_2 - 20% PuO_2 in TREAT were computed. Two types of flux filters, nickel and B-10, were considered as a means of reducing the test pin power density relative to TREAT and reducing the power peaking in the test pin.

In the filter design, calculations indicated the filter material must be placed outside of the capsule region containing the three encapsulated test pins to avoid skewing of the power in this region. Replacement of the aluminum heat sink surrounding each test pin with the above materials resulted in flux skewing which would probably be unacceptable. On this basis, the annular space outside the heater was modified to accommodate the flux filter.

Computed power distributions and power normalization factors with flux

filters are given in Figures 4-11 and 4-12. The power normalization factors and, hence, the power distributions shown are not highly reliable because of the sensitivity of the normalization technique to the spatial self-shielding of the thermal neutron flux in the test pin, as well as to neglected perturbations in the epithermal flux. These factors are particularly important in highly enriched test pins. It is important, therefore, that experimental verification of the power normalization factors precede actual transient testing of fuel and that test capsules be designed with sufficient flexibility to permit further reduction in test pin power should it become necessary. Results from the current Series 4 program (see subsection 4.1) of TREAT transient irradiation of high burnup test pins should provide a check on the calculation method.

4.5 LOSS OF FLOW EXPERIMENTS

Work on Mockup 1A has been restarted. Components which have been in storage are being prepared for assembly. A minor redesign has been made to simplify assembly.

FIGURE 4-11. POWER NORMALIZATION AND DISTRIBUTION IN FULLY ENRICHED TEST PINS

FIGURE 4-12. POWER NORMALIZATION AND DISTRIBUTION IN FULLY ENRICHED TEST PINS

SECTION V

TASK E—FUEL MATERIAL PROPERTIES AND PERFORMANCE EVALUATION

5.1 GENERAL

The purpose of this task is to establish physical, mechanical, and thermal properties of ceramic fuels which are needed for the design of fast reactor fuel elements and which are used in fuel behavior calculations under both steady state and casualty conditions. Ex-reactor experiments are conducted (as well as in-reactor) to provide property measurements or metallographic observations which aid in the interpretation of the behavior of irradiated specimens.

5.2 HIGH BURNUP IRRADIATIONS

5.2.1 Capsule E2P-Q (Relative Restraint of 10 and 15 mil Cladding)

Capsule E2P-Q was removed from GETR on June 22, 1969 for an interim examination. The fuel burnup was approximately 20,000 MWd/Te. The fuel pins will be neutrographed and diametral measurements taken using an in-cell profilometer prior to reencapsulation and reinsertion in GETR for further irradiation.

5.2.2 Capsules E5C, E5D, E5E

(Comparison of 16-Inch-Long Solid Pellet, Annular Pellet, and Powder Fuel)

These capsules are awaiting hot cell examination after being irradiated in the GETR to their design burnup of 130,000 MWd/Te. These pins contain physically blended mixed oxide fuel irradiated to the highest burnup yet attained under the PA-10 program.

5.3 PHYSICAL PROPERTY STUDIES

5.3.1 Mechanical Properties

A program to study the compressive creep behavior of mixed plutonium-uranium oxide fuels was initiated. The experiment is designed to yield creep data on mixed oxide fuels as a function of temperature and stress to determine the effect of stoichiometry, density, and plutonium content. The results of this study will be of value in the studies of fuel swelling. Seven batches

of sample pellets (0.375-inch diameter x 0.375-inch length) have been prepared with different stoichiometry, density, and plutonium content. The sample pellets will be used in a parametric study to correlate the effects of these physical properties on the creep deformation rate of mixed oxide fuel. The nominal values of density and composition for each batch of the sample pellets are given below:

<u>Batch</u>	<u>Composition</u>	<u>Density</u>
A	(Pu _{0.2} U _{0.8})O _{1.97}	96
B	(Pu _{0.2} U _{0.8})O _{1.97}	93
C	(Pu _{0.2} U _{0.8})O _{1.97}	90
D	(Pu _{0.1} U _{0.9})O _{1.97}	93
E	(Pu _{0.3} U _{0.7})O _{1.97}	93
F	(Pu _{0.2} U _{0.8})O _{1.93}	96
G	(Pu _{0.2} U _{0.8})O _{2.00}	96

Determination of the microstructure and chemical analysis of each batch of pellets was initiated.

Installation of the test apparatus in a plutonium glovebox was completed. The testing of the plutonium-uranium oxide samples will be initiated in the next quarter. Test temperatures will vary from 1000°C to 1600°C with compressive loads from 1000 psi to 10,000 psi.

A topical report covering the creep behavior of stoichiometric UO₂ was completed.

5.3.2 Thermal Conductivity

Preparations for experiments to determine the effects of density and stoichiometry on the thermal conductivity of mixed plutonium-uranium oxides were started. The radial heat-flow test apparatus was installed in a plutonium glovebox and sample fabrication initiated.

Sample pellets with the following nominal values of density and composition are being fabricated:

<u>Composition</u>	<u>Theoretical % Density</u>
(U _{0.2} Pu _{0.8})O _{1.97}	96
(U _{0.2} Pu _{0.8})O _{1.97}	93
(U _{0.2} Pu _{0.8})O _{1.97}	90
(U _{0.2} Ox _{0.8})O _{1.93}	96
(U _{0.2} Pu _{0.8})O _{2.00}	96

5.4 FUEL IMPURITY CONCENTRATION LIMITS AND STANDARDS

In setting meaningful limits on impurity concentrations, the effects of impurities on fuel performance and the

economic factors associated with the manufacture of the fuel, as well as the effect on the entire fast reactor core should be considered. A conservative approach was used to appraise the effect of impurities from a core physics standpoint. Concentration limits were established by the amount of the impurity associated with a reactivity worth of 4 cents. Table 5-1 is a tabulation for 25 impurity elements producing a combined reactivity worth of 1 dollar. Also considered in setting the impurity limits were the decay chains of activated impurities and their effect on fuel reprocessing shielding and handling. Comparison of the impurity levels detected in production mixed oxide fuel with these values shows that the impurity levels are much lower than the arbitrary 4-cent reactivity worth and nearer the limits of analytical detection. The problem of hydrogen impurity levels had not previously been considered from a physics standpoint. The proposed FCR impurity limits have taken into consideration hydrogen contents above the arbitrary 4-cent reactivity effect by lowering the limits of other impurities to levels comparable to typical production values. The sum of the impurities still results in a total reactivity worth of less than 1 dollar. The impurity concentration limits are not rigid numbers; rather the

limits are subject to periodic revision as new data become available.

A comparison of the residual hydrogen found in production fuel sintered in He-6% H₂ and N₂-6% H₂ atmospheres is being made to evaluate the effect of sintering atmosphere composition on residual hydrogen concentration in the fuel. The results of this program will be reported when all data have been obtained.

5.5 FUEL-SODIUM BEHAVIOR

X-ray lattice parameter measurements were made on three unirradiated samples of Incoloy-800 clad (U-25 wt% Pu)O_{2±x} after a 1000 hour exposure to sodium vapor at 1400°F. The specimens had initial O/M ratios of 1.97, 2.00, and 2.04. These measurements showed that unit cell sizes were essentially uniform and were slightly larger than that of the stoichiometric compound despite initial differences in O/M ratios. On the basis of this study, the lattice parameters of hypostoichiometric fuel were found to decrease to that of the stoichiometric compound after isothermal exposure to 1400°F sodium vapor for periods up to 1000 hours as shown in Figure 5-1. Similarly, the lattice parameters of the

TABLE 5-1

IMPURITY CONCENTRATION LIMITS^a

Element	Arbitrary 4-Cent Reactivity Worth (ppm) ^d	Proposed FCR Limit (ppm)	Worst Case for Production Order (ppm)	Limit of Detection (ppm) ^b	Limit of Efficient Analysis (ppm) ^c
H	1.0	5	3	7 μ l total gas 1 ppm H ₂ O	---
B	15	10	0.3	0.5	10
C	1000	150	188	50	--
N	250	---	20	10	--
F	50	25	15	5	--
Na	500	200	55	40	100
Mg	500	100	5	10	100
Al	1000	500	200	10	100
Si	1000	200	99	5	100
Cl	250	25	13	5	--
Ca	650	100	150	50	100
Ti	3800	---	< 250	--	--
V	5300	200	< 5	20	100
Cr	5800	250	150	5	100
Mn	300	50	8	1	20
Fe	3900	500	160	40	100
Ni	6100	500	92	20	100
Cu	700	100	10	10	5
Zn	250	400	41	50	100
Mo	400	100	12	10	100
Ag	200	20	0.5	1	5
Cd	1200	100	0.2	2	100
Sn	1300	300	< 2	10	20
Pb	2300	50	< 5	1	20
Be	15	15	--	1	10

^aAll elements except H, C, N, F, Cl based on emission spectrograph results.^bAs of 3/12/69, accuracy +100% -50%; limits constantly being lowered.^cLimits allowing easy direct comparison to standards as of 6/16/69.^dppm = parts per million parts of uranium-plutonium metal by weight.

FIGURE 5-1. LATTICE PARAMETER MEASUREMENTS OF (U-25 wt% Pu) $O_{2\pm x}$ EXPOSED TO SODIUM

hyperstoichiometric compound increased under these conditions to a common equilibrium value. These results imply that oxygen migration occurred. The development of a stoichiometry close to 2.0 in a hypostoichiometric compound exposed to sodium had previously been noted in sample B8B containing fuel with O/M ratios of 1.94 and 2.00*. That sample had failed, but

irradiation was continued to 14,000 MWd/Te. These values are also plotted in Figure 5-1.

References

*"Sodium Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-Sixth Quarterly Report, February-April 1968," June 1968, (GEAP-5631).

SECTION VI

TASK F—INTEGRAL FAST FLUX IRRADIATIONS

6.1 GENERAL

The purpose of this task is to apply the more promising results of the basic fuel technology developments to the design, fabrication, and integral testing of fuel elements in fast fluxes. Important fuel performance characteristics, such as fission gas release, fuel-cladding interactions, and fuel thermal performance, are determined under conditions simulating, as closely as possible, those expected in a fast ceramic reactor. Testing of fuel pins in a fast flux, in principle, will permit testing of fuel bundles under fair approximation of prototypical conditions.

The test parameters for Task F are selected by a combination of design analysis (which relies on basic properties data) and on the results of selected experiments (such as those in Task E and Task L). In addition, the results of prior fast-flux irradiation tests are interpreted as quantitatively as possible to guide the selection of future tests. The selection of test parameters is

also influenced and guided by the work under Task C, which focuses on the preferred range for safety considerations of LMFBR designs. Demonstration of high fuel burnup (50,000 to 100,000 MWd/Te) is required over a range of power levels (4 to 24 kW/ft). Fuel parameters that are under study in a fast environment include density, configuration, fabrication process history, stoichiometry, diametral gap, axial gap, and external restraint. Fast flux tests are performed with both encapsulated and unencapsulated pins.

6.2 MEDIUM POWER, LONG TERM
IRRADIATIONS (GROUP 2: SCOPING
IRRADIATIONS)

Irradiation of the remaining four Group-2 capsules was continued in EBR-II subassembly XO50. Calculated peak burnup accumulated on the fuel in these capsules was 95,200 MWd/Te. Target burnup for these capsules is 100,000 MWd/Te.

6.3 LOW POWER FUEL SWELLING IRRADIATIONS

6.3.1 Group 0 (Low Power Swelling of High Density Fuel)

The post irradiation examination of capsule FOE was continued. Table 6-1 summarizes the FOE test parameters. The pin was removed from its capsule and diameter profiles obtained at 0°, 45°, 90°, and 135° orientations around the pin. The profilometer traces are shown in Figure 6-1. The maximum diametral increase measured was 0.0014 inch (0.56%). The four profiles were very similar and indicated that ridges had developed at locations corresponding to fuel pellet interfaces. Excellent correspondence was observed between the pellet interfaces visible in neutron-radiographs of the fuel column and the ridges indicated on the profilometer traces. Maximum height of the ridges observed was ~0.0008 inch.

The fuel pin was sectioned at several positions along its length. Examination of the "as-cut" sections showed that no central void had formed anywhere along the length of the fuel column.

Metallographic examination of two transverse sections from the peak power region of the fuel pin (Figures 6-2 and 6-3) showed the following:

- No fuel-cladding gap was present in the sections examined; however, voids originally present as a result of chips in the "as-fabricated" pellets were not filled with fuel or fission products.
- No central void was formed in the fuel. Calculated peak temperature at the center of the fuel was 1100°C to 1200°C.
- No reaction was observed to have occurred between the fuel and cladding. The calculated peak cladding i. d. temperature was 950°F.
- A high density region was observed in one of the fuel sections examined (Figure 6-2). It is postulated that this high density region was formed during the fuel fabrication process since the fuel irradiation temperatures were not high enough to cause fuel densification in-pile.*

Only equiaxed grains were observed in the two fuel sections examined. The sizes of the grains and the distribution of grain sizes observed in each specimen were distinctly different. High magnification

*Pellets were handmade in 1964. Uneven forming pressures were eliminated by new automatic equipment.

TABLE 6-1

CAPSULE FOE TEST PARAMETERS

Fuel Parameters

Fuel Type	solid pellet
Pu	20%
U-235	93.7%
O/M	1.999
Length	14.25 in.
Density	
Pellet	96.5% theoretical
Smeared	93.6% theoretical

Cladding Parameters

Material	Type-347-Stainless steel, annealed
Wall	0.015 in.
Outside Diameter	0.250 in.

Fuel-Cladding Gap	0.002 in.
-------------------	-----------

Calculated Peak Fuel Burnup	~40,000 MWd/Te
-----------------------------	----------------

Peak Linear Power	~6 kW/ft
-------------------	----------

Peak to Average Power Ratio Over Fuel Length	~1.14
--	-------

pictures showing the change in grain size across the radius of the two sections are shown in Figure 6-4. The fuel in the pellets at the two sections examined was fabricated from different batches of coprecipitated powder. Different temperatures were also used to sinter the two batches of fuel powder. Metallography of archive pellets made from each batch of powder showed that pellets made from the

powder sintered at a temperature of 1430°C (powder batch No. 55) had an average grain size of about 3μ . The powder sintered at a temperature of 1550°C (powder batch No. 39) had grains ranging in size from 25μ to 35μ . The irradiated fuel from powder batch No. 55, sintered at 1430°C, exhibited the 3μ "as-fabricated" grain size adjacent to the cladding; however, an increase in grain size occurred uniformly and continuously

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FIGURE 6-1. PROFILOMETER TRACES OF F0E FUEL PIN

FIGURE 6-2. METALLOGRAPHIC CROSS SECTION OF FUEL PIN F0E TAKEN ~7 inches FROM BOTTOM OF FUEL PIN. FUEL WAS FABRICATED FROM POWDER BATCH No. 55. HIGH DENSITY REGION NEAR CENTER OF FUEL IS POSTULATED TO HAVE BEEN FORMED DURING FABRICATION (SEE SECTION 6.3.1)

FIGURE 6-3. METALLOGRAPHIC CROSS SECTION OF FUEL PIN F0E TAKEN $\sim 9 \frac{1}{2}$ inches FROM BOTTOM OF FUEL PIN. FUEL WAS FABRICATED FROM POWDER BATCH 39.

FUEL CENTER

FUEL CENTER

MID RADIUS

MID RADIUS

FUEL SURFACE

FUEL SURFACE

CLAD

FUEL FABRICATED FROM POWDER BATCH No. 55. FUEL WAS SINTERED AT 1430°C FOR 3 1/2 hours. ETCHANT $\text{HNO}_3 - \text{H}_2\text{O}_2$

FUEL FABRICATED FROM POWDER BATCH No. 39. FUEL WAS SINTERED AT 1550°C FOR 2 hours. ETCHANT $\text{HNO}_3 - \text{H}_2\text{O}_2$

FIGURE 6-4. HIGH MAGNIFICATION (250X) PHOTOGRAPHS ILLUSTRATING THE RADICALLY DIFFERENT MICROSTRUCTURE OBSERVED IN EQUIVALENT POWER REGIONS OF FUEL PIN F0E

across the radius of the fuel. In the center of the fuel the grain size was 25μ to 35μ . The grain size of the fuel sintered at 1550°C remained in the range from 25μ to 35μ . No significant variation in grain size across the fuel radius was observed.

Based on the observed differences in fuel grain size between the two cross sections, it appears that either (1) grain growth in mixed oxide fuel can occur at temperatures below 1000°C if the initial assintered grain size is sufficiently small or (2) the calculated fuel temperatures were low. At the present it is not clear which of these hypotheses is correct.

A typical photomicrograph of the etched Type-347 stainless steel fuel cladding is shown in Figure 6-5. Slightly enhanced grain boundary precipitation was observed at the cladding i.d. The calculated temperature of the inside surface of the cladding at the position shown in the photomicrograph was 900°F .

A transverse metallography specimen from the peak temperature region of the F0E capsule tube was examined. The polished specimen of 304 stainless steel contained an isolated region of continuous intergranular precipitates near the tube o.d. (the flowing sodium surface). In Figure 6-6 is shown polished and etched 250X micrographs of this area. The affected grain boundaries were 1 to 2 grain

diameters from the tube o.d. (~ 1.5 mils) and roughly parallel to the tube surface. The affected area was about 5 mils in length. The seam weld in the tube was remote from this area. A single small grain had fallen out of the polished matrix near the precipitates; however, a 10% oxalic acid electrolytic etch (of sufficient severity to delineate the other grain boundaries in the sensitized structure) resulted in the loss of several other grains in the affected area.

Material for the F0E capsule tube was supplied by ANL and came from heat No. 136272. The analysis of this heat is shown in Table 6-2.

TABLE 6-2
CERTIFIED CHEMISTRY OF AISI 304,
HEAT NO. 136272

	<u>Weight %</u>
Carbon	0.07
Manganese	1.55
Phosphorus	0.027
Silicon	0.48
Chromium	18.46
Nickel	9.35
Iron	balance
Sulfur	0.019

The exposure history of capsule F0E is tabulated in Table 6-3. The F0E capsule had the longest residence time in the EBR-II of any capsule examined to date on

FIGURE 6-5. TYPICAL APPEARANCE OF F0E TYPE 347 S/S FUEL PIN CLAD. SECTION SHOWN WAS LOCATED IN PEAK FLUX REGION OF THE PIN. THE CALCULATED i.d. TEMPERATURE WAS 900°F

ETCHED

250X

POLISHED

250X

FIGURE 6-6. METALLOGRAPHY OF F0E CAPSULE SHOWING REGION OF CONTINUOUS INTERGRANULAR PRECIPITATES OBSERVED NEAR THE o.d. OF THE TYPE-304 S/S TUBE

TABLE 6-3

EXPOSURE HISTORY—CAPSULE F0E

EBR-II Core Position	Row 7, XG02
Insertion Date	7-16-65
Removal Date	8-15-68
Months in Core	37
Peak Fluence (nvt > 1 MeV)	1.3×10^{22}
Top of Core Fluence (nvt > 1 MeV)	0.6×10^{22}
Average Capsule Temperature at Core Centerline (°F)	825
Average Capsule Temperature at Core Top (°F)	910

the FCR Program (37 months). Previous examinations of FCR capsule tubes have shown no sign of anomalous grain boundary precipitates or cracks which might be attributed to caustic stress corrosion*.

Irradiation of the two Group-0 capsules in subassembly XG03 and the two capsules in subassembly XG04 was continued. Calculated peak burnup on the fuel in each subassembly is 58,600 MWd/Te.

6.3.2 Group 6 (Fuel Cladding Interactions Under Accelerated Swelling)

Post-irradiation examination was initiated on the four Group-6 capsules. The Group-6 capsules were removed from

EBR-II subassembly XO10 on April 24, 1969. Calculated peak burnup on the fuel in the capsules was 61,500 MWd/Te. Each capsule was visually examined and gamma scanned. All four capsules were found to be bowed about 1 inch over a length of 3-1/4 feet. The bow prevented the insertion of the capsules into the standard neutron-radiography fixture. As a result, neutron radiography of the capsules is being delayed until a special fixture is fabricated. The extent of the capsule bow is illustrated in Figure 6-7. The two Group-F7 capsules (see subsection 6.7.1) are shown in Figure 6-7 for comparison. The four capsules were covered by red surface deposits over much of their surface. The surface deposits were more abundant toward the top end (higher temperature end of the capsules).

The gamma scans of the capsules did not show any unusual activity peaks which would indicate failure of the fuel pins.

*"Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-Fourth Quarterly Report, August-October 1967, December 1967," (GEAP-5541), p. 116.

FIGURE 6-7. PHOTOGRAPH OF GROUP F6 CAPSULES SHOWING BOW OF EACH CAPSULE. THE TWO CAPSULES ON THE RIGHT ARE GROUP F7 CAPSULES

Gamma scans of each capsule are shown in Figure 6-8 through 6-11.

6.3.3 Group 8A (Effect of Fuel Density on Fuel Swelling Rate)

Irradiation of the seven Group-8A capsules in EBR-II subassembly X019 and nine capsules in subassembly X020 was continued. Calculated peak burnups on the fuel in subassemblies X019 and X020 was 52,000 MWd/Te and 53,000 MWd/Te respectively. Target burnups for fuel in subassemblies X019 and X020 are 50,000 and 100,000 MWd/Te, respectively. Subassembly X019 is presently scheduled for removal from the EBR-II at the end of Run36 in July 1969.

6.3.4 Group 8B (Effect of Void Deployment on Fuel Cladding Interactions)

Irradiation of the 19 capsules in Group 8B was continued toward a target burnup of 50,000 MWd/Te. Calculated peak burnup on the fuel in these capsules is 21,200 MWd/Te. The Group 8B capsules are being irradiated in EBR-II subassembly X036.

6.4 HIGH POWER INTERMEDIATE BURNUP IRRADIATIONS

6.4.1 Group 3B (Density and Stoichiometry Effects)

Irradiation of the 18 Group 3B capsules was continued in EBR-II subassembly

X027. Calculated peak burnup on the fuel in these capsules is 58,600 MWd/Te. Target burnup for the fuel in these capsules is 120,000 MWd/Te. An interim examination is scheduled when a burnup of 80,000 MWd/Te has been attained.

6.4.2 Group 4 (Low Density Fuel Behavior at High Heat Ratings)

Approval-in-principle to continue the irradiation of the seven Group 4 fuel pins to higher burnups was granted by the EBR-II Project. In addition, the EBR-II Project requested that an addendum to the Group-4 hazards analysis be submitted (by GE) to support continued irradiation of the Group-4 fuel pins to higher burnups. The preparation of the addendum was initiated.

6.5 ELEVATED CLADDING TEMPERATURE TESTS

6.5.1 Group 5 (Fuel Performance at Peak FCR Cladding Temperatures)

Approval for insertion of the Group-5 capsules into EBR-II was obtained and the capsules were loaded into EBR-II subassembly X064. The subassembly was inserted into the reactor on May 28, 1969. The peak burnup on the fuel in the Group-5 capsules is 8,600 MWd/Te.

6.5.2 Group 10B (Cladding Performance at Peak FCR Cladding Temperatures)

Fabrication of the 19 Group 10B fuel pins was completed. Two vibratory

FIGURE 6-8. GAMMA SCAN OF GROUP F6 CAPSULE FOJ

FIGURE 6-9. GAMMA SCAN OF GROUP F6 CAPSULE FOK

FIGURE 6-10. GAMMA SCAN OF GROUP F6 CAPSULE FOL

FIGURE 6-11. GAMMA SCAN OF GROUP F6 CAPSULE FOM

compacted fuel pins were rejected because of intermixing of the powder $(\text{PuU})\text{O}_2$ fuel with the UO_2 insulator and blanket material. Refabrication of these two fuel pins was initiated.

The bottom end plugs were welded into all 19 capsule bodies and the welds were inspected and hydrostatically pressure tested for leaks. Seventeen of the fuel pins were wire wrapped, inserted into their capsules, and prepared for sodium filling.

6.5.3 Group 10A (Cladding Performance at Peak FCR Cladding Temperatures)

A proposal for approval-in-principle for the Group-10A subassembly was prepared and submitted to AEC-RDT (AEC-Reactor Development and Technology).

6.6 UNENCAPSULATED FUEL PIN IRRADIATIONS

6.6.1 Group 3A (Unencapsulated Fuel Pin Irradiation to Evaluate Fuel-Cladding and Cladding-Coolant Interactions)

Irradiation of the 16 Group-3A fuel pins in EBR-II subassembly X040 was terminated on June 30, 1969 for purposes of an interim examination. Calculated peak burnup on the fuel in these pins was 36,700 MWd/Te. The subassembly was removed

from the reactor for cleaning and disassembly on July 14, 1969. Following the interim examination the fuel pins are to be reinserted into the EBR-II and their irradiation continued to a fuel burnup of 50,000 MWd/Te.

No comments were received on the addendum to the Group-3A hazards analysis requested by the EBR-II Project prior to the removal of the subassembly from the EBR-II core. The purpose of the addendum was to provide the EBR-II Project with information to evaluate whether irradiation of the Group 3A pins to burnups greater than 3 at. % was justified without an interim examination.

6.6.2 Group 9A (High Power, High Burnup, Unencapsulated Fuel Pin Irradiations)

Irradiation of the 37 Group-9A fuel pins in EBR-II subassembly X043 was continued. Calculated peak fuel burnup in the subassembly is 20,700 MWd/Te.

No comments were received concerning the addendum to the Group-9A hazards analysis which was submitted to the EBR-II Project in May 1969. The purpose of the addendum was similar to that submitted for the Group 3A, i.e., supply the EBR-II Project with information for evaluating whether irradiation of the Group-9A

subassembly to burnups in excess of 3 at. % is justified without performing an interim examination.

6.6.3 Group 9B (Medium Power, High Burnup, Unencapsulated Fuel Pin Irradiations)

The 37 fuel pin Group-9B irradiation was continued in EBR-II subassembly X062. Calculated peak burnup on the fuel in the subassembly was 6,900 MWd/Te. Target burnup for the fuel in the Group-9B subassembly is 100,000 MWd/Te.

6.6.4 Group 9C (High Power, High Burnup, Unencapsulated Fuel Pin Irradiations)

Irradiation of the 37 Group-9C fuel pins was continued. The peak burnup on these fuel pins is 15,900 MWd/Te. Target burnup for the Group-9C fuel pins is 100,000 MWd/Te.

6.6.5 Group 9D (Medium Power, High Burnup, Unencapsulated Fuel Pin Irradiations)

The 37 pin Group-9D (subassembly X058) was inserted into the EBR-II on April 24, 1969. After one cycle of operation the subassembly was moved from row seven to row six of the EBR-II core because of concern over flux peaking in the EBR-II blanket (row eight). The flux peaking

calculations are being checked by EBR-II, and if possible, the subassembly will be returned to row seven of the core; if not, irradiation will continue in row six at ~10% higher power than originally requested.

An analysis of the dimensional measurements taken on the Group-9D bundle and hexagonal channel during assembly was completed. The results indicate the following:

- The inside dimensions of the hexagonal channel did not meet the tolerances specified on ANL drawing EB-1-3997-C*. On the drawing the channel dimensions are 2.006 ± 0.003 inches across the flats. Measurements on the channel showed the dimensions ranged from 1.991 to 2.017 inches.
- Calculated clearances between the hexagonal channel and rod bundle based upon measurements of the channel and as-built rod diameters and wire-wrap diameters indicated clearances ranged from zero to 0.026 inch along the length of the subassembly. The subassembly design specifies a clearance ranging from 0.004 to 0.010 inch.
- Actual measurement of the clearances between the channel and rod bundle were made by inserting shims between the fuel pin bundle and the inside of

*Channels were designed and built by ANL.

the channel. These measurements showed that the clearances were distributed throughout the bundle, in contrast to accumulating along any particular side of the channel. This is believed to result from bowing of the rods. As larger shims were inserted, no abrupt change in the force required to remove the shims was observed; even when the calculated clearances were exceeded. This is believed to result from the elimination of the as-fabricated bow from the pins and the channel to bundle clearances. Therefore, the bundle of wire wrapped fuel rods acts as a spring.

- When even larger shims were inserted the fuel rods bowed toward the direction where there was no wire-wrap support.
- Based on the as-built dimensions of the the fuel pins, wire wrap, and channel, the clearance available for swelling of the fuel rods was calculated to be 1%. Calculations based on measurements using the shims indicate that the area available for swelling is as great as 2.85%.

Calculated peak burnup on the fuel in these fuel pins was 9,200 MWd/Te.

6.6.6 Group 11A (Cladding Performance at Peak FCR Cladding Temperatures)

A proposal for approval-in-principle for the Group-11A subassembly was prepared and submitted to AEC-RDT.

6.7 PRE-IRRADIATION TRANSIENT TESTS (Group 7: Effect of Irradiations, Density, and Axial Restraint)

Irradiation of the four Group-7 capsules was continued in EBR-II sub-assembly XO50. Calculated peak burnup on the fuel in these capsules was 20,300 MWd/Te.

6.8 BURNUP ANALYSIS OF MIXED OXIDE FUEL

Work was initiated to improve the accuracy of the present capability for determining burnup of FCR fuels. The following work was begun:

- Modification of the burnup computer code. The code was changed so that it now considers U-235 as the base isotope for burnup calculations instead of U-238. This change was required because of the high U-235 enrichments used in FCR fuel tests to obtain the desired power ratings during irradiation in the EBR-II.

- Procurement of additional samples of standard isotopes for burnup "spike solutions" for calibration of the mass spectrometer.
- Arrangement of a burnup sample exchange with other laboratories doing burnup analysis on irradiated mixed oxide fuel.

6.9 MICROPROBE EXAMINATION OF IRRADIATED MIXED OXIDE FUELS

A sample from ANL fuel pin SOV-6 was prepared for examination in the GE shielded microprobe. Analysis of the sample will be completed when a set of standards for the microprobe is obtained and calibration completed. The SOV-6 sample is being examined as part of a microprobe sample exchange program with ANL.

SECTION VII

TASK G—CORE NEUTRONICS, THERMAL
HYDRAULICS AND SAFETY ANALYSIS

7.1 GENERAL

The purpose of Task G is to develop and use computational analysis to provide guidance for effective planning and coordination of the experimental programs within P. A. 10, and to contribute to the basic analytical and design tools for the sodium-cooled fast breeder program. This requires integration of results of several disciplines, including core neutronics, thermal hydraulics, and safety and economic analysis. To accomplish this, improved analytical methods and computer programs are being developed and used in analytical studies of fast reactor neutronics, safety characteristics, and both analytical and experimental thermal hydraulics. These studies, together with a continual evaluation and surveillance of the results of experimental programs, form a basis for establishing current design and safety criteria. These criteria, in turn, define the range of reactor characteristics and test parameters upon which the experimental programs must focus.

Methods development and parametric analyses are being focused primarily on

the areas of: (1) nuclear data and updating of group constants, with emphasis on processing of data for use in neutronics computations, (2) nuclear core analyses, including the evaluation of approximations used in neutronics computations, (3) transient and safety analysis, including accident studies and improvement of FORE II and sodium voiding models, and (4) thermal-hydraulic and fuel analysis, including fuel swelling model development and coolant mixing studies.

7.2 NUCLEAR DATA AND GROUP
CONSTANTS7.2.1 Cross Section Files

The analysis of Pu-239 resonance parameters has been extended to include p-wave contributions. Theoretical p-wave cross sections, based on Otter's analysis,^{(1)*} have been factored into the computations by subtracting the p-wave contribution from different experimental values and deriving the s-wave, J=1 values

*References are in subsection 7.6

of $\langle \Gamma_f \rangle$ from the residual cross sections. Several values of the p-wave strength function were considered, ranging from 1×10^{-4} to $2 \times 10^{-4} \text{ ev}^{-1/2}$. The analysis was concentrated in the energy range from 1 to 20 keV, where the alpha value is particularly important to LMFBR design, and fits were made to the fission cross-section data of James,⁽²⁾ Shunk, et al.,⁽³⁾ and Gilboy and Knoll;⁽⁴⁾ and to the values for alpha of Gwin, et al.⁽⁵⁾ Values of $\langle \Gamma_f \rangle_{J=1}$ were also obtained for the recommended fission cross sections of James and Patrick.⁽⁶⁾ In deriving $\langle \Gamma_f \rangle_{J=1}$ from fission cross-section data, the other s-wave parameters were given the values stated in GEAP-10028,⁽⁷⁾ and $\langle \Gamma_f \rangle_{J=1}$ adjusted to exactly produce the experimental cross section under consideration. To fit $\langle \Gamma_f \rangle_{J=1}$ to the Gwin alpha data, a trial and error approach was used, since the p-wave contribution could not be directly subtracted from the measurements.

Figure 7-1 shows the resulting sets of $\langle \Gamma_f \rangle_{J=1}$ as functions of energy. The values reported by Pitterle⁽⁸⁾ are also shown for comparison. Above 8 keV the inelastic width is non-zero and presents a significant effect on the derived values of $\langle \Gamma_f \rangle_{J=1}$. The results shown in Figure 7-1 have been corrected for the effects of Γ_{in} by applying ratios of the $\langle \Gamma_f \rangle_{J=1}$ values which include Γ_{in} to those which exclude Γ_{in} , as obtained by Pitterle. Also, for all but Pitterle's

results, the p-wave strength function was taken to be $1.5 \times 10^{-4} \text{ ev}^{-1/2}$, and the reduced neutron width was held constant at $0.94 \times 10^{-3} \text{ ev}^{+1/2}$.

Comparison of the curves of $\langle \Gamma_f \rangle_{J=1}$ in Figure 7-1 indicates the following:

- The fission data agree quite well between 3.5 and 20 keV.
- The Gwin alpha values give generally lower $\langle \Gamma_f \rangle_{J=1}$ below 8 keV. Below 3.5 keV, agreement is good with the Shunk data, but the drop above 4 keV has no support in the fission results. The low value between 5 and 6 keV for the alpha data is caused by a sharp peak in Gwin's alpha at this energy. While some other experimental results indicate the same peak, Gwin's 10 gram sample and corrected measurements of Czirr⁽⁹⁾ do not show it. This analysis also indicates that it is not there.
- The rise in $\langle \Gamma_f \rangle_{J=1}$ at 8 to 9 keV is indicated by all but the Gilboy and Knoll results. This corresponds to a low value of alpha.
- From 10 to 20 keV, the fission cross section results are again in good agreement, with the higher values from the Gwin data indicative of low alphas in this range.

FIGURE 7-1. DERIVED $\langle \Gamma_f \rangle_{J=1}^{l=0}$ VALUES

- At higher energies the results become tenuous because of the uncertainty in the p-wave contributions and the treatment of the Γ_{in} effects, both of which become increasingly important with energy. It might be expected that $\langle \Gamma_f \rangle_{J=1}$ will average somewhere between 30 and 50 MeV at 20 keV and follow statistical theory above that, with a fission threshold somewhere above 100 keV.

A review has been made of resolved resonance parameters for Pu-239. The parameters of Derrien, et al.,⁽¹⁰⁾ have a higher level density up to 400 eV than the present ENDF/B file and are more consistent with the statistical parameters derived in the analysis described above. Two sets of parameters have been derived from Derrien's tables. The first has values of $J=1/2$ given to resonances which have no definitive experimental spin assignment. For the second, all resonances have been given assignments of either $J=0$ or 1, based upon experimental assignments and the magnitudes of Γ_f . Both sets are being evaluated for how well they agree with cross-section measurements.

The detailed cross sections and secondary energy distributions for the (n, 2n) reaction in Be-9 given by Perkins⁽¹¹⁾ have been put into the ENDF/B format. The secondary distributions have been described as tabulated

functions and are now being evaluated by generating transfer matrices with the ENDRUN code.

Cross sections and resonance parameters for Pu-239, U-238, and Pu-240 which are consistent with the CROF04 values have been put into the ENDF/B format. This will enable a GMUG data tape to be generated which is consistent with the cross sections currently in use for LMFBR physics computations at GE-BRDO.

7.2.2 Multigroup Cross Section Averaging

Specifications for the second version of ENDRUN, ENDRUN-2, have been listed and details are being prepared for programming. ENDRUN-2 will include an improved numerical integration technique for resolved resonances, better ordering of input, a more accurate computation of the elastic scattering matrix, and changes to handle modifications to the ENDF/B format.

The ENDRUN-1 code is checked out and will be sent to the Argonne Code Center when documentation is completed. The first draft of the topical report on ENDRUN-1 is being reorganized for clarification purposes. Two reports will be issued: one which provides user information and the other which describes the programming details.

Check-out of the TDOWN code has progressed through a successful

computation of ZPR-III, Assembly 48, using one-dimensional diffusion theory. Figure 7-2 schematically shows the procedure used in this computation. The GMUG file had been previously computed with the ENDRUN code in 29 groups (1/2 lethargy group widths) and for all Assembly 48 materials. TDOWN was used to compute reactor dependent (self-shielding) microscopic cross sections, which were put into a MUG file. A spherical, one-dimensional diffusion theory problem was then run with the DEMON code using the MUG file.

Table 7-1 shows a comparison of the results obtained with this procedure and those obtained at Argonne National Laboratory using the MC² and MACH-1 codes,⁽¹²⁾ and the earlier GE results in which the FCC code was used in place of TDOWN.⁽¹²⁾ The comparison is good enough to conclude that the TDOWN computations do not contain gross errors. Earlier comparisons of cross sections generated with TDOWN and FCC using the same generalized file also showed good agreement. The higher value of k_{eff} obtained with TDOWN is probably caused by a downward correction, by a factor of 10, in the ENDF/B capture cross section of Cr. In general, agreement between the TDOWN calculation and the MC²-MACH-1 calculations and experiment is better than in the previous calculation (with FCC), particularly for the central worths of light

materials. The improvement in the U-238 capture and fission ratios may be due to a better interpolation scheme between σ_0 values in TDOWN. The difference in the ENDRUN generated fission and capture ratios for Pu-239 and those obtained with MC² may be because of a more detailed treatment of fission width distributions in ENDRUN, where 10 representative values were used rather than the 5 in MC².

Further check-out of TDOWN will be performed by computing the 10 benchmark critical assemblies specified for ENDF/B testing. Check-out of TDOWN in 60 groups is also underway in order to determine the accuracy of the rapid cross-section adjusting technique as a function of number of groups.

7.2.3 Working Committee Activities

Lists of interface data in standard format for shielding and point kinetics computational modules have been distributed to members of the ad hoc committee on Computer Code Coordination. Following the procedure prescribed by the committee for arriving at standardized interfaces lists, the following suggested modifications have been sent to the appropriate sub-committee chairmen:

- Nuclear Data Processing:
Corrections to errors and
limitations in definitions.

FIGURE 7-2, COMPUTATIONAL PROCEDURE FOR CHECKING OUT TDOWN CODE

TABLE 7-1

COMPARISON OF ENDRUN-TDOWN-DEMON RESULTS WITH THOSE FROM
MC²-MACH-1 AND ENDRUN-FCC-DEMON

	Experimental Measurement	ENDRUN TDOWN DEMON	MC ² MACH-1*	ENDRUN FCC DEMON*
k_{eff}	--	0.9851	0.9783	0.9756
$\sigma_f^{238}/\sigma_f^{235}$	0.0307	0.0305	0.0304	0.0300
$\sigma_f^{239}/\sigma_f^{235}$	0.976	0.897	0.919	0.904
$\sigma_f^{240}/\sigma_f^{235}$	0.243	0.228	0.226	0.223
$\sigma_f^{241}/\sigma_f^{235}$	--	0.186	--	0.180
$\sigma_c^{235}/\sigma_f^{235}$	--	0.277	0.282	0.280
$\sigma_c^{238}/\sigma_f^{235}$	0.138	0.136	0.142	0.141
$\sigma_c^{239}/\sigma_f^{235}$	--	0.196	0.197	0.191
$\sigma_c^{240}/\sigma_f^{235}$	--	0.229	0.219	0.218
$\sigma_c^{241}/\sigma_f^{235}$	--	0.209	--	0.204
Central Worths ($10^{-5} \Delta k/\text{mole}$)				
C	-0.05	+0.02	-0.015	-0.20
Na	-0.14	-0.08	-0.07	-0.22
Fe	-0.69	-0.79	-0.84	-0.88
Ni	-1.07	-1.20	-1.10	-1.33
U-235	79	101.6	102	103.4
U-238	-5.9	-7.54	-7.88	-7.6
Pu-239	106	130	134	132
Pu-240	19.4	19.8	20.6	20.4

*Hess, A. L., "Summary of Phase IIB Analyses of ZPR-3 Assemblies 48 and 11 with ENDF/B-Based Cross Sections," ZPR-I Memo No. 1 sent to members of CSEWG, June 25, 1969.

- Neutronics: Expansions and corrections to the finite difference flux interfaces. Major modifications and additions to interfaces for variational and nonvariational synthesis.
- Kinetics: Modifications to the point kinetics interface.

Evaluations have begun for arriving at updated fuel isotopic files for ENDF/B by September 1969. This activity is coordinated through an informal committee formed by the National Neutron Cross Section Center at Brookhaven National Laboratory. GE-BRDO has been given the responsibility of providing new resonance parameters and smooth cross sections for the resolved and unresolved energy range up to 10 keV. The evaluations associated with this activity are described in Subsection 7.2.1.

7.3 CORE ANALYSIS

7.3.1 Two-Dimensional Methods

The new code, DOT2DB, which incorporates the diffusion theory subroutines of the BNWL code 2DB⁽¹³⁾ into the two-dimensional transport theory code DOT*, has been checked out, including the recently added triangular mesh option for diffusion theory. By refining the programming of the main transport and diffusion flux

calculation subroutines, the code calculational efficiency was increased by approximately 15%. A first draft of the users manual for DOT2DB has been written.

A (r, z) diffusion representation of EBR-II was calculated using DOT2DB in 10 groups. The same calculation was made using the synthesis code BISYN with five shape functions calculated through Rows 1 and 2, Row 3, Row 4, Row 5, and Row 6 of EBR-II. Because of the control and safety rod positions in Rows 3 and 5, these shape functions should differ substantially from the others, making this problem a difficult test of BISYN. The results of the calculations with DOT2DB and BISYN are compared in Figure 7-3 showing the average radial power distribution. The error in the power distribution as calculated by BISYN is as high as 10 percent.

7.3.2 Methods Evaluation

The investigation of the magnitude of the error introduced in LMFBR calculations through the approximation of anisotropic scattering by a transport cross section has been completed. Twenty-four group cross sections, with anisotropic scattering represented by a Legendre series through P_3 , were used in BRDO's version of the one-dimensional transport/diffusion code ANISN⁽¹⁴⁾ for calculations of spherical model LMFBRs. The transport cross section, σ_{tr} , used in this analysis was

*From F. R. Mynatt in "A Users Manual for DOT, A Two-Dimensional Discrete Ordinates Transport Code With Anisotropic Scattering." Work is in progress.

FIGURE 7-3. EBR II RADIAL POWER DISTRIBUTION AS COMPUTED WITH THE DOT2DB AND BISYN CODES

defined for group g as

$$(\sigma_{tr})_g = (\sigma_t)_g - \sum_{g'} (\sigma_{sl})_{g \rightarrow g'}$$

where $(\sigma_{sl})_{g \rightarrow g'}$ is the P_1 anisotropic scattering matrix (with the $2\ell+1$ factor) and $(\sigma_t)_g$ is the total cross section.

The diffusion option of the code ANISN allows for the use of $\ell=0$ and $\ell=1$ P_ℓ scattering matrices. When both are used, the theory is essentially "consistent P_1 " theory. Some of the calculations were made in this mode, with the results tabulated under diffusion theory in Table 7-2.

The results of the k_{eff} calculations using the P_ℓ scattering matrices through P_0 , P_1 , P_2 , and P_3 and for the P_0 scattering matrix with the transport cross section approximation are given in Table 7-2. The following conclusions, some of which were preliminarily made in the previous quarterly, (7) can be drawn from these results:

- Anisotropic scattering is important in FCR calculations. The P_0 results without the transport cross-section approximation are very poor.
- Using the P_0 scattering matrix with the transport cross-section approximation reduces the error in the treatment of anisotropic scattering to about 20 percent of the error

introduced by using diffusion theory in place of transport theory. Therefore:

- When diffusion theory is used, it is unnecessary to consider any treatment of anisotropic scattering more accurate than that provided by the transport cross section.
- When transport theory is used, P_ℓ scattering matrices should be used, through P_1 for large reactors and P_2 or P_3 for small reactors.
- The error in the treatment of anisotropic scattering is not entirely independent of the error due to the use of diffusion theory.

A draft of the report to be titled, "Effects of Nuclear Methods Approximations on Fast Spectrum Power Reactor Analysis," has been completed.

7.3.3 Fuel Cycle Methods

The startup fuel cycle code, FUMBLE, has been successfully run on a simple problem for the 1000 MWe LMFBR design. Checking of the results and check-out of the important options is continuing.

Check-out runs with the BICYCL code are continuing in order to cover more of the many options and correction routines in the code. The first draft of the report describing BICYCL and its application is being

TABLE 7-2

DEPENDENCE OF k_{eff} ON THE TREATMENT
OF ANISOTROPIC SCATTERING

Core	Radius	Method	$k(P_0)$	$k(P_1)$	$k(P_2)$	$k(P_3)$	$k(\sigma_{\text{tr}})$
Base ↓ ↓ Blanketed ↓	45.8 cm	Diffusion	1.08247	1.00339	--	--	1.00439
		S_4	--	1.01110	(1.01304)*	(1.01305)	1.01438
		S_6	--	1.00984	1.01178	1.01179	1.01297
		S_8	--	1.00953	(1.01147)	(1.01148)	(1.01266)
	↓	S_{16}	--	1.00930	(1.01124)	(1.01125)	(1.01243)
	65.0 cm	Diffusion	1.06431	1.01201	--	--	1.01285
	96.3 cm	Diffusion	1.04825	1.01941	--	--	1.01996
		S_4	--	1.02018	(1.02042)	(1.02042)	1.02103
		S_6	--	1.02004	(1.02028)	1.02028	(1.02089)
	↓	S_{16}	--	1.02001	(1.02025)	(1.02025)	(1.02086)
Blanketed ↓	47.4 cm	Diffusion	1.06318	1.00289	--	--	1.00390
		S_4	1.06863	1.01072	1.01367	1.01272	1.01406
		S_6	--	1.01045	(1.01340)	(1.01245)	(1.01379)
	104.3 cm	Diffusion	1.04091	1.02038	--	--	1.02079
	↓	S_4	1.04148	1.02119	(1.02124)	1.02129	1.02189

*The data given in parenthesis has been estimated from the other data.

revised to clarify the choices and uses of the code options.

7.3.4 Code Conversion to FORTRAN IV

The BISYN code has been successfully executed on the UNIVAC 1108 computer using both the all-FORTRAN and the machine language I/O routines. The all-FORTRAN version required twice as much central processor time as the machine language version. Only one small problem with the 1108 operating system was encountered in using the all-FORTRAN version. The FORTRAN I/O subroutines should be directly applicable to the BICYCL, BISYN-PERT, and TDOWN codes.

It is planned that the GE 635 and UNIVAC 1108 machine language versions, as well as the FORTRAN I/O routines, will be included in future code packages sent to the Argonne Code Center. The all-FORTRAN version of BISYN will be sent shortly.

7.3.5 Design Effects Studies

A series of one-dimensional, 60 energy group, diffusion theory calculations were performed to investigate the U-238 Doppler effect near B_4C control rods in an LMFBR. The normal design methods of treating control rods is to compute spatially self-shielded cross sections for the control material by running a cell

calculation and homogenizing the control with the fuel regions for subsequent calculations. In such cases, it was expected that the Doppler coefficient was under-computed.

In Table 7-3 is listed the U-238 Doppler change in reactivity computed for control cell problems in which the control, surrounding structure and coolant, and fuel regions were first treated heterogeneously and then treated homogeneously with the usual control self-shielding correction. It is seen that with the control rod in, the Doppler coefficient for the cell is nearly 10% greater in the heterogeneous case. With control out it is only 1.5% larger, but with control out the Doppler coefficient is 2.5 times larger than with control in. The effect this has on the total reactor Doppler coefficient will be less in percentage, depending on how much of the core is represented by the unit control cell.

7.4 TRANSIENT AND SAFETY ANALYSIS

7.4.1 Transient Analysis

Major portions of the FREADM-1 code have been combined and check-out of the logic for performing multi-bundle heat transfer calculations has been completed.

Check-out of the kinetics portion of the code was performed by running similar transients with FREADM-1 and FORE-II. A 10 cent/sec ramp insertion was input with

TABLE 7-3

COMPARISON OF DOPPLER COEFFICIENT FOR HOMOGENEOUS
AND HETEROGENEOUS CELLS

<u>Cell Treatment</u>	<u>Control Rod</u>	<u>U-238 Doppler Coefficient ($T \, dk/dT$)</u>
Homogeneous*	In	-0.00293
Homogeneous	Out	-0.00726
Heterogeneous	In	-0.00320
Heterogeneous	Out	-0.00737

*Control material cross sections corrected for spatial self-shielding.

starting power density at 10^{-5} of full power density for a large LMFBR. Doppler feedback was unimportant over most of the transient; hence, the test provides a check of the kinetics calculations which is nearly independent of differences in the feedback models employed. Results are shown in Figure 7-4. The small differences in relative power shown in Figure 7-4 may be due to time step differences, since the FREADM-1 calculation required only $1/3$ as many time steps as the FORE-II case.

A feature of FREADM-1 which provides a large degree of flexibility is the triggering capability. Accident initiating and propagation events, as well as scram, may be triggered when preset levels of any one of a number of different variables are reached. The triggering capability is summarized in Table 7-4.

Specification of a subroutine for computing reactivity feedback under such severe conditions that the core motion can be considered to be hydrodynamic has been completed for use in the second code version, FREADM-2.

7.4.2 Properties Sensitivity Study

The objective of this work is to determine whether improved information about various physical, nuclear, and mechanical reactor parameters is needed to predict more accurately reactor response to accidents, such as larger flow transients or reactivity insertions. The study is being conducted by examining two accidents with the FORE-II computer code and varying each of the desired parameters to determine the sensitivity of the transients to them.

FIGURE 7-4. COMPARISON OF FORE II AND FREADM-1 RESULTS

TABLE 7-4

SUMMARY OF ACCIDENT INITIATING AND PROPAGATING
EVENTS AND TRIGGERING VARIABLES

Event	Triggered when:
Reactor Scram	<ol style="list-style-type: none"> 1. Elapsed time from start of transient exceeds input level. 2. Power level exceeds input level. 3. Reactor period is less than input value. 4. Rate of power increase exceeds input level. 5. Elapsed time from fuel failure by bundle-type exceeds input level. * 6. Elapsed time from void initiation by bundle-type exceeds input level. * 7. Coolant velocity by bundle-type is less than input level. * 8. Coolant outlet temperature by bundle-type exceeds input level. *
Programmed Reactivity Insertion	<ol style="list-style-type: none"> 1. Elapsed time from start of transient exceeds input level. 2. Elapsed time from fuel failure by bundle-type exceeds input level. * 3. Elapsed time from void initiation by bundle-type exceeds input level. *
Fuel Failure in a Bundle-Type	<ol style="list-style-type: none"> 1. Elapsed time from start of transient exceeds input level. 2. Average fuel temperature at an axial section of a bundle exceeds input level. * 3. Temperature of a specified radial fuel node exceeds input level. * 4. Cladding temperature exceeds input level. * 5. Elapsed time from failure in another bundle exceeds input level (propagation). *

TABLE 7-4

SUMMARY OF ACCIDENT INITIATING AND PROPAGATING
EVENTS AND TRIGGERING VARIABLES (Continued)

Event	Triggered when:
Voiding of a Bundle-Type	<ol style="list-style-type: none"> 1. Elapsed time from start of transient exceeds input level. 2. Cladding temperature exceeds input level. * 3. Coolant temperature exceeds input level. * 4. Temperature of a specified radial node exceeds input level. * 5. Average fuel temperature of an axial section exceeds input level. * 6. Elapsed time from void initiation in another bundle-type exceeds input level (propagation). *

*Trigger levels input by bundle-type.

Although many of the FORE-II cases have been run, the detailed examination of the results is incomplete, and therefore, only a few preliminary observations will be reported here.

In the case of the flow reduction accident, the choice of the model used for the gap coefficient between the fuel and the cladding appears to be the most significant single factor. If a variable gap model is used, in which the cladding can grow due to thermal expansion, the rate of temperature rise in the sodium is considerably reduced, and the possibility of sodium boiling is made much smaller for a given flow reduction and control response.

In the case of a large, step reactivity insertion, several parameters have significant effects on the resultant transient.

- The use of a variable fuel-cladding heat transfer coefficient appears to increase the severity of the transient. The oxide heats up faster than the cladding, reducing the gap size, and increasing the gap thermal conductivity. The rate increases at which heat is transferred to the sodium, and the probability of sodium boiling also increases.
- The use of new specific heats for the mixed oxide, which are based on the NMPO (Nuclear Materials and Propulsion Operation) and ANL

measurements for UO_2 , reduce the rate of fuel melting. The probability that the initial failure mechanism is the pressure pulse, caused by the rapid release of entrained gaseous fission products from the molten fuel, is reduced.

- The assumption is made that during the transient, the mixed oxide is restrained from growing axially by the fuel cladding. If the oxide-cladding "bond" is weak in the axial direction during the transient, and the oxide can grow axially to some extent because of its own temperature rise, the resultant negative feedback is then large enough to cause a considerable reduction in the severity of the transient.

7.5 THERMAL HYDRAULICS AND FUEL ANALYSIS

7.5.1 Temperature Distribution Around an Edge Fuel Pin

The effect of wire-wrap induced flow sweeping on circumferential cladding temperature distributions has been determined for edge rods in fuel bundles of 1.2 P/D (pitch to diameter ratio).

Previous predictions of cladding temperature variations for edge rods ⁽¹⁵⁾ were based on a slug-flow model and neglected flow mixing between adjacent channels. In

addition, the temperature fields were allowed to develop axially for only 12 inches, the length of the wire pitch, rather than for the full core length. Both of these assumptions were recognized as being very approximate, however, it was believed that the reported temperature variations were conservative (greater than the true maximum variations), because the neglect of inter-channel mixing more than counterbalanced the reduced thermal development length.

A more sophisticated solution to the temperature distribution around an edge rod has been provided by a three-dimensional analysis including the effects of coolant mixing from flow channel to adjacent flow channel. The method used does not predict circumferential temperature variations. However, temperatures of cladding locations directly facing each coolant channel were deduced from the results. The complete cladding circumferential temperature distributions were then inferred by comparing these results with the detailed distributions reported in the earlier study.

In wire wrapped bundles, interchannel mixing is primarily the result of induced flow sweeping from channel to channel. The thermal effect is to increase the effective sodium thermal conductivity between channels. Flow sweeping in a 1.2 P/D wire wrapped bundle can be characterized by $0.074 \leq Kr \leq 0.140$ which is the COBRA mixing parameter defined by Rowe. ⁽¹⁶⁾ These limiting levels will be

referred to as minimum and maximum mixing, respectively.

Results are summarized in Table 7-5 and Figures 7-5 through 7-7. Figure 7-5 shows cladding temperatures at the core exit for edge rods with 50 mils (full wire wrap diameter) wall spacing. The dashed line added to the slug-flow model curve represents the localized effect of the wire wrap at the 180° angular position. With the relatively large wall spacing of 50 mils, the slug-flow results are good approximations to the temperatures including mixing. The thermal effect of interchannel mixing is visible as a reduction in temperature near the 60° position and a general smoothing of the temperature distribution.

As the wall spacing is reduced, the disagreement between the results of the two models becomes more pronounced.

The variations at 25 mils and 16 mils wall spacing (Figures 7-6 and 7-7 are less severe than the slug-flow predictions. In each case, the temperatures are averaged across the cladding thickness, rather than being local values.

An important conclusion of this study is that the cladding temperature gradients reported in the preliminary study are conservatively high. Interchannel mixing is effective in reducing the gradients around edge rods. Even in the cases where the total variation is not reduced, the distribution is significantly smoothed.

A second conclusion is that a reduced wire-wrap thickness at the wall is an effective means of reducing the cladding temperature variations caused by coolant channel asymmetry. The preliminary slug-flow analysis indicated the minimum temperature

TABLE 7-5

MAXIMUM CIRCUMFERENTIAL CLADDING TEMPERATURE VARIATIONS (°F)

<u>Edge Rod</u>	<u>Slug Flow Model</u>	<u>With Interchannel Flow Mixing</u>	
		<u>Minimum Mixing</u>	<u>Maximum Mixing</u>
50 mils wall spacing	115	125	105
25 mils wall spacing	76	40	30
16 mils wall spacing	76	47	35

Diameter = 0.250 inches, P/D = 1.2
Average Linear Power = 11.5 kW/ft

Wire Wrap Pitch = 12 inches
Peak Linear Power = 13.8 kW/ft

FIGURE 7-5. MAXIMUM CLADDING TEMPERATURE VARIATIONS FOR WIRE-WRAPPED EDGE RODS, 50 MILS WALL SPACING

FIGURE 7-6. MAXIMUM CLADDING TEMPERATURE VARIATIONS FOR WIRE-WRAPPED EDGE RODS, 25 mils WALL SPACING

FIGURE 7-7. MAXIMUM CLADDING TEMPERATURE VARIATIONS FOR WIRE-WRAPPED EDGE RODS, 16 mils WALL SPACING

variation results with a wall spacing of 10 to 20 mils. The current study suggests this spacing should be 15 to 25 mils.

7.5.2 Fuel Swelling

7.5.2.1 Model Development

7.5.2.1.1 Analogy Between Swelling and Thermal Expansion

In the preceeding quarterly report,⁽⁷⁾ equations were reported which correlated, for Types-304 and -316 stainless steel, the swelling induced by irradiation with temperature and fluence. There is concern about the behavior of core structures which are also subjected to external stresses or strains. During this quarter, model development was initiated which will assist in predicting the performance of the fuel cladding, coolant channels, core grid structure, etc., under swelling conditions.

The previously reported relation correlating cladding data was of the form:

$$-\frac{\Delta\rho}{\rho} = AF^n \exp \left[\frac{C_1}{T} - \left(\frac{C_2}{T} \right)^2 \right] \quad (7-1)$$

where:

- $\frac{\Delta\rho}{\rho}$ = change in density (%)
 F = fluence/ $(10^{22} \text{ n/cm}^2, E > 0.1 \text{ MeV})$
 T = irradiation temperature ($^{\circ}\text{K}$)
 A, n, C_1, C_2 are constants.

For present demonstrative purposes, the following values were assumed corresponding to Equation (7-5) in the previous quarterly report.⁽⁷⁾

$$A = 6.7 \times 10^{-5}$$

$$n = 2.00$$

$$C_1 = 15,670$$

$$C_2 = 2710$$

Irradiation induced swelling should be analogous to thermal expansion based on two apparently valid assumptions: swelling is (a) isotropic and (b) independent of stress. With these assumptions, the linear expansion due to swelling can be described approximately* by the relation

$$\left(\frac{\Delta\ell}{\ell} \right)_s = -\frac{1}{3} \frac{\Delta\rho}{\rho} \quad (7-2)$$

Thermal Expansion is predicted by the relation

$$\left(\frac{\Delta\ell}{\ell} \right)_{T_1, T_2} = \alpha (T_2 - T_1)$$

where:

- α = linear coefficient of thermal expansion
 T_2 = temperature above some reference temperature, T_1 .

*It should be realized that since deformation may be large, the error involved in this approximation may reach several percent.

In general, α is a function of temperature while T can be a function of both time and distance. Similar parameters have been obtained for swelling by substituting Equation (7-1) into (7-2) so that

$$\left(\frac{\Delta \ell}{\ell}\right)_S = \beta \gamma. \quad (7-3)$$

where:

$$\beta = \exp \left[\frac{C_1}{T} - \left(\frac{C_2}{T} \right)^2 \right]$$

$$\gamma = \frac{A}{3} (\Phi t)^n$$

using

$$F = \Phi t.$$

where:

$$\Phi = \text{flux}$$

$$t = \text{time}$$

Note that β corresponds to α since it is a function of temperature and that γ corresponds to T since it is a function of both time and distance (flux varies with distance).

The total free linear expansion is then

$$\frac{\Delta \ell}{\ell} = \alpha T + \beta \gamma. \quad (7-4)$$

Therefore, in any previously developed stress analysis relation involving thermal expansion, the term αT can be replaced with the combination of terms (Equation 7-4). This does not provide a wealth of analytic tools from the thermal stress field, however. The magnitude and time dependent nature of the strains encountered in swelling push rigorous analysis to the limits of stress analysis technology.

It was possible, through a simplified approach, to estimate pertinent values of stress and strain without using a rigorous approach.

7.5.2.1.2 Hex Channel Analysis Example

To illustrate the approach used, the strain and resulting stress was estimated in a fuel assembly wrapper (hexagonal channel) at the edge of the core for the proposed 1000 MWe plant (Figure 7-8).⁽¹⁷⁾ The core edge position was chosen because it appeared that at this position the fluence gradient across a channel would be the greatest of any region in the core. However, this may not be the "worst" condition as will be discussed later. It was assumed that external forces held the channel straight (not bowed). It was also assumed that overall axial or radial extension of the channel was not restricted by external loading. To allow an approximate solution, it was assumed that the swelling was linear across the channel.

FIGURE 7-8. STRESS AND STRAIN DISTRIBUTIONS PRODUCED BY CHANNEL SWELLING—6 MONTHS EXPOSURE 1000 MWe CORE EDGE MIDPLANE

These assumptions allow the use of the following equation for restrained bowing.

$$\epsilon_Z = \pm \frac{\alpha \Delta T}{2} \quad (7-5)$$

where:

ϵ_Z = maximum axial strain on either side of the member

$\alpha \Delta T$ = differential expansion across member

Substituting $\Delta(\beta\gamma)$ for $\alpha \Delta T$ (neglecting thermal expansion for the present) and using (7-1), Equation 7-5 becomes:

$$\epsilon_{Z_S} = \pm \frac{A}{6} \left\{ F_1^n \exp \left[\frac{C_1}{T_1} - \left(\frac{C_2}{T_1} \right)^2 \right] - F_2^n \exp \left[\frac{C_1}{T_2} - \left(\frac{C_2}{T_2} \right)^2 \right] \right\} \quad (7-6)$$

where subscripts on variables indicate sides 1 or 2 (Figure 7-8). The values of flux shown in Figure 7-8 were estimates obtained from analysis of the 1000 MWe Sodium Cooled Fast Reactor. ⁽¹⁷⁾ The strain level at 6 months was 0.6% as shown in Figure 7-8. Six months operating time was chosen because it corresponds to the region of experimental data for Equation 7-1.

For comparison, the strain due to thermal expansion across the channel was calculated to be $\pm 0.02\%$.

As mentioned previously, another point worthy of note is the fact that the greatest tendency to bow may not be at the position of greatest fluence gradient. In Equation 7-6, the differential axial strain can easily be shown to be a function of both the differential and average values of fluence. Therefore, locating the limiting condition of maximum differential axial strain will require careful analysis.

Differentiating Equation 7-6 with respect to time with $F = \Phi t$, the following relation for strain rate was obtained:

$$\dot{\epsilon} = \pm \frac{nA}{6} t^{n-1} \left\{ \Phi_1^n \exp \left[\frac{C_1}{T_1} - \left(\frac{C_2}{T_1} \right)^2 \right] - \Phi_2^n \exp \left[\frac{C_1}{T_2} - \left(\frac{C_2}{T_2} \right)^2 \right] \right\} \quad 7-7$$

At six months, the maximum strain rate was calculated to be 2.6×10^{-6} in. per in./h (Figure 7-8). Assuming that steady-state creep data was applicable, a stress level of 41,000 psi was found from the recommended irradiated creep correlation. ⁽¹⁸⁾ Making the further assumption that the stress distribution across the channel is linear, the midplane bending moment was estimated to be 61,400 in.-lb. Using the moment and a suitable model for lateral loading on the channel, an estimate could be made of lateral pressure on the channel required to keep it straight.

The assumption made above, that the swelling distribution is linear across the channel, is an over-simplification since even if the flux were linear, it enters the swelling relation Equation 7-1 to the exponential power, n . All that can be stated without further detailed analysis is that the peak strain, strain rates, and stresses predicted above, in general, would be underestimated. Whether the bending moment is over- or underestimated is not readily apparent.

7.5.2.1.3 Fuel Cladding Analysis Example

The fuel cladding presents a different problem but can be handled with the same technique as the channel above.

Since the diameter is small, the differential swelling across the diameter is small and there is much less tendency to bow. However, since swelling was found to be a function of temperature coupled with the fact that a large temperature gradient can exist through the wall (58°F), differential swelling occurs through the cladding thickness. This problem is analogous to the thermal stress created by the radial temperature gradient through the cladding wall. The strain in the axial and hoop direction will be equal and determined by Equation 7-6. The fluence was assumed to be constant across the diameter of the cladding. Using the same technique and equations as for the channel, the calculated

surface strain, strain rates, and peak stresses are shown in Figure 7-9. Three months exposure was chosen to keep fluence level within the range of validity for the swelling correlation. For comparison, the strain due to the differential thermal expansion was found to be $\pm 0.03\%$. As before, the swelling gradient was assumed linear through the cladding thickness.

Analysis very similar to that for the channels and cladding, can be applied to the core support plate. Given the flux gradient through the plate and the operating temperature, the radial surface strain levels and rates could be predicted assuming the plate to be restrained flat. Then similar to the above analysis, an idea of the restraining forces required to hold the plate flat could be obtained.

7.5.2.2 Physical Properties Studies

There was no activity in this area during the quarter.

7.5.2.3 Correlation with Experimental Data

The effort is continuing to improve the correlation of fuel pin swelling data. An attempt was made to correlate the data with an equation of the form

$$\Delta D = A + BQ^n \exp\left(\frac{C}{T}\right)$$

FIGURE 7-9. STRESS AND STRAIN DISTRIBUTIONS PRODUCED BY CLADDING SWELLING-3 MONTHS EXPOSURE 1000 MWe CORE CENTER

where:

ΔD = change in pin diameter

Q = linear power

T = cladding temperature

A, B, n, C = empirical constants

met with only limited success.

An attempt was also made to correlate the effect of local changes in fuel density to local changes in fuel pin swelling. Although one pin showed an apparent correlation, the results must be termed inconclusive.

The computerized storage of profilometry data is currently being expanded.

7.5.2.4 Development of a Fuel Swelling Computer Program

A topical report covering the fuel swelling model and the computer program is being prepared.

7.5.3 VELVET Temperature Calculations

A topical report covering the VELVET computer code and parameter study is being published.

7.5.4 Pressure Drop in Wire Wrap Bundles

A study of pressure drop in wire wrapped rod bundles, spaced in a triangular array, was performed with the objective of improving existing correlations or proposing

a new correlation if necessary.

To study the increase in pressure drop due to the wire wraps, it was decided to plot the ratio $\frac{\Delta P}{\Delta P_0}$ as a function of wire-wrap pitch, wire diameter, rod diameter, and number of rods per bundle.

where:

ΔP is the pressure drop through a bundle with wire wraps,

ΔP_0 is the pressure drop through the bundle with the same mass flow rate without wire wraps.

The wire wrap pressure drops used in the study were those available from reported test results.

Since, not all the tests made measurements of the spacerless pressure drop (ΔP_0), it was first necessary to obtain a reliable calculation technique. Such a method was developed from the works of M. Kh. Ibragimov, et al.⁽¹⁹⁾ The Ibragimov method consists of calculating a pressure drop by employing the usual hydraulic diameter friction factor with a multiplicative factor which accounts for the effect on pressure drop of the skewing of the velocity profiles.

The accuracy of this method for calculating ΔP_0 was checked against experimental

data (20,21) which were not used in its derivation and was found to agree with good accuracy. Figure 7-10 shows a comparison between test and pressure drop calculation.

Only three sets of data were found to be strictly applicable to rods in a triangular array with helically coiled wires. These are the CVTR(22), FFTF(20), and Karlsruhe(23) data. The CVTR data consisted of pressure drop measurements for pitch-to-diameter ratios ranging from 1.12 to 1.262 at a wire-wrap pitch of 12 inches, and wire wrap pitches ranging from 15 to 4 inches for a pitch-to-diameter ratio 1.205.

The FFTF (Fast Flux Test Facility) data was most extensive. Geometrical parameters for which test data was obtained are shown in Table 7-6.

The Karlsruhe data is for a pitch-to-diameter ratio of 1.417 for bundle sizes of 7, 19, and 37 rods and a pitch-to-diameter ratio of 1.215 for a 61 rod bundle. Rod diameter was 12 mm. The wire pitches were 200 mm and 300 mm, respectively.

Other data that might have been included are those of Waters(21) for a seven rod bundle in a circular channel. This arrangement of rods contains triangular subchannels and edge channels that are similar to the edge channels of a

seven rod bundle in a hexagonal can. These results are useful for comparison purposes, however, the differences in flow distribution due to different shaped edge channels could lead to different and possibly erroneous interpretations of the other pressure drop data.

Most of the above test data are plotted in Figure 7-11. The pressure drop ratio is expressed as a function of wire-pitch-to-rod-diameter ratio (ℓ/D) and rod-spacing-pitch-to-diameter ratio (P/D). Most of the data are for 37 rod bundles, therefore, in order to cross-plot the data on a consistent basis, it was assumed that the pressure drop ratio ($\Delta P/\Delta P_0$), for some finite wire pitch, increases or decreases with respect to numbers of rod bundles in the same way as the straight-wire-pressure-drop ratio. The data for rod bundle sizes other than 37 pins was made equivalent to the 37 pin rod bundle size by this method. The data was then cross-plotted and interpolated. The results of the interpolations are shown in Figures 7-12 and 7-13. During the process of cross-plotting the data, an additional factor was found to influence the pressure drop: namely, the additional flow area resulting when the size of the hexagonal channel is larger than theoretically necessary to accommodate the rods. Differences in pressure drop caused by the excess area were small when the wire-pitch-to-diameter ratio (ℓ/D) is greater

FIGURE 7-10. COMPARISON OF SPACERLESS CORRELATION WITH TESTS

TABLE 7-6

FFTF PRESSURE DROP TEST PARAMETERS

<u>No. of Rods Per Bundle</u>	<u>Rod Diameter (in.)</u>	<u>Wire Diameter (in.)</u>	<u>P/D</u>	<u>Wire Pitch (in.)</u>
37	0.25	0	1.12	0
37	0.25	0.03	1.12	3
37	0.25	0.03	1.12	6
37	0.25	0.03	1.12	12
37	0.25	0.03	1.12	24
37	0.236	0.0425	1.18	12
37	0.266	0.016	1.06	6
37	0.266	0.016	1.06	12
37	0.196	0.0235	1.12	12
37	0.240	0.040	1.167	12
37	0.300	0.0360	1.12	12
37	0.250	0.0792	1.317	6
37	0.250	0.0792	1.317	12
19	0.250	0.030	1.12	12
217	0.250	0.030	1.12	12

than 25, however, these differences became significant with ℓ/D less than 25. To correlate this difference, it was assumed that the excess area was in the outer region of the bundle between the last row of rods and the channel wall. The parameter chosen was:

$$\frac{H/D}{P/D - 0.5}$$

where:

H = the perpendicular distance from the last row of rods to the bundle wall

D = rod diameter

P = rod spacing

The curves of Figures 7-12 and 7-13 were fitted resulting in the expression.

$$\frac{\Delta P}{\Delta P_0} = \frac{\ell/D}{B(\ell/D) - A}$$

FIGURE 7-11. PRESSURE DROP RATIO VERSUS WIRE PITCH TO DIAMETER RATIO

FIGURE 7-12. PRESSURE DROP RATIO VERSUS WIRE PITCH TO DIAMETER RATIO

FIGURE 7-13. PRESSURE DROP RATIO VERSUS WIRE PITCH TO DIAMETER RATIO

where A and B are constant for a particular pitch to diameter ratio (P/D) and edge spacing parameter $\frac{H/D}{P/D-0.5}$. Figure 7-14 shows the variation of A and B with pitch-to-diameter ratio and the edge spacing parameter. Only two sets of values were obtained from the data.

To extrapolate the data to rod bundles with a greater number of fuel rods, it is assumed that the increase with number of rods per bundle for a given pitch-to-diameter

ratio and wire-pitch-to-diameter ratio is the same as the increase for the pressure drop ratio ($\Delta P/\Delta P_0$) for straight wires. This extrapolation method should yield reasonably accurate results in the range ℓ/D less than 25.

Overall accuracy of this method of calculating the pressure drop is expected to be best in the range of ℓ/D greater than 25.

FIGURE 7-14. EMPIRICAL CONSTANTS A AND B

7.6 REFERENCES

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SECTION VIII

TASK L—CLADDING DEVELOPMENT

8.1 GENERAL

The technical and economic feasibility of ceramic-fueled fast reactors depends on the attainment of high levels of performance of the fuel system, especially the survival of the cladding at temperatures and specific powers of practical interest. The present status of testing and analysis of potential cladding material with regard to alloys and environmental variables indicates good prospects that adequate initial performance of cladding for fast reactor fuel elements can be obtained by using carefully specified materials within the range of compositions of commercial processing methods. No material, however, has yet been tested under fully typical conditions. Data are not yet available to establish optimum compositions or structures, although some tentative trends have been suggested. This lack of test experience to support the rational selection of preferred cladding specifications is regarded as a major remaining obstacle to the attainment of the reliable operation of an LMFBR fuel system.

The general objectives of the cladding development program for the next several years are to contribute to the attainment of

the desired levels of fast reactor fuel element performance which, in turn, are required by the desired fuel-cycle economics. The objectives of specific activities follow:

- Determine the performance characteristics—including mechanical properties, fracture characteristics, radiation damage, and structural stability—of the most likely cladding material candidates under test conditions simulating the LMFBR environment.
- Obtain experimental data and perform systematic data analysis of the response of candidate alloys to test environments which simulate reactor applications, on which to base selection of a preferred alloy for near-term LMFBR applications.
- Determine the in-depth material behavior required to establish performance limitations (such as allowable stress levels, creep rates, maximum strains, local temperatures), on selected alloys, based on initial fast-flux test results, and recommend preferred material

specifications and design limits for LMFBR use.

8.2 MATERIALS IRRADIATION PERFORMANCE

8.2.1 Series L-2, L-6, and L-15--EBR-II Materials Capsules

The series L-2 (four capsules), L-6, and L-15 materials capsules are at EBR-II in various stages of their scheduled irradiations. The L-2 (four capsules) are in Row 4 subassembly (S/A) designated XO50. Irradiation of this S/A began in January and is scheduled to continue until late 1969 when the target fluences for these capsules will be reached. The L-2 capsules have accumulated $6. \times 10^{22}$ n/cm² total fluence. *

The L-6 materials capsules are in a row 4 S/A designated XO18A and have accumulated a total fluence of 5.2×10^{22} n/cm². These capsules have been discharged and will be incorporated into a new subassembly. Further irradiation in Row 2 is planned for the L-6 capsules to acquire the target total fluence of 1.3×10^{23} n/cm² in a reasonable time period.

The L-15 materials capsules previously irradiated in S/A XO39 to 1.4×10^{22} n/cm² total fluence are still waiting reinsertion into another Mark A-19 S/A for further irradiation. Target total fluence for the L-15 is 3×10^{22} n/cm².

8.2.2 Series L-10 EBR-II Materials Capsules

A description of ten L-10 capsules and the initial post-irradiation examination have been presented earlier.^{(1)*} These capsules were irradiated in row 4 (S/A XO16) of EBR-II for ~175 EFPD** to an estimated peak total fluence of 2.8×10^{22} n/cm² and peak temperature during irradiation >1185°F, probably closer to 1400°F.

Examination of 30 pairs of tensile specimens removed from the L-10 capsules revealed that 11 pairs were stuck together. Included in these specimens were five commercial alloys (Types 304, 316, 321, and 347 and Incoloy-800) and the high purity 14Ni-14Cr-Fe alloy. These specimens were in pairs sandwiched between tungsten heater strips. The sandwich assembly was then inserted into a tubular specimen and filled with high purity static sodium under pressure.

*References are in subsection 8.4.

**EFPD = Effective full power days at 45 MW.

*Unless otherwise noted, all reported fluences are based on EBR-II physics predictions.

Various attempts were made to separate the commercial alloy tensile specimens. Attempts to forceably separate the specimens caused specimen bending. Another attempt was exposure for 1 hour in a boiling solution of 5% potassium permanganate, 20% potassium hydroxide, and the balance H_2O . No reaction was noted and the specimens remained bonded. A second attempt using a solution of 5% nitric acid at 130°F for 1 to 2 hours was used. A slight reaction was noted and one pair of specimens separated. Two other pairs of specimens which were bonded with larger surface area did not separate. No macro attack was noted. The final solution used was 3% bromine in dry methanol at room temperature. The specimens were left in this solution for time periods of 1 to 24 hours. A general dissolution of the specimens occurred with increased times with no indication of specimen separation.

One pair of specimens (commercial Incoloy-800 and Type-316 stainless steel) were examined by optical metallography. A discontinuous metallurgical bond was clearly evident over a large portion of the gage section of these specimens (Figure 8-1). The gross intergranular corrosion which occurred in the Incoloy-800 specimen during the chemical treatment given these specimens in attempting to separate them is also readily apparent. The large second phase particles in the Type-316 specimen are thought to be sigma phase which is commonly

observed in this steel after prolonged exposures above 1300°F. The large size of these sigma particles is typical only for temperatures well above 1300°F and adds further support to the present prediction that the irradiation temperature of the L-10 core specimens was in the neighborhood of 1400°F.* The discontinuous nature of the bond zone is typical of a growth process and one can speculate that the initial bonding was probably a point contact that grew with time at the high temperatures. The presence of sodium at the interface results in very clean, highly-active surfaces which are a prerequisite for strong metallurgical bonding. No apparent reaction product was present at the bond interface as would be expected from the high mutual solid solubilities of the major constituent (Fe, Ni and Cr) in the composition ranges of these alloys.

Based on the above observations, it is reasonable to assume that the adhesion experienced by the L-10 tensile specimens renders them unacceptable as specimens for tensile test. Any mechanical testing of these specimens should be performed only in pairs of specimens and the test results would be interpreted with difficulty and serious uncertainty. However, careful metallographic examination of failed pairs of specimens after tensile testing could prove fruitful in determining the nature of the fracture and ductility (fracture strain

*Errors in Gamma heating data provided by EBR-II in 1966 probably caused error in temperature.

FIGURE 8-1. TYPICAL BOND ZONE OF STUCK L-10 SERIES TENSILE SPECIMENS.
MODIFIED GLYCEREGIA ETCH.

or reduction in area) experienced by each of the specimens during the tensile test. This information can be compared with the non-stuck specimens from L-10. Performing density measurements with the non-stuck specimens from L-10 appears to be less promising since these specimens could contain void space at the interface, as shown in Figure 8-1, which can act as internal porosity resulting in low density readings.

8.2.3 Series L-14 EBR-II Materials Capsules

The purpose of the L-14 capsule irradiation experiment was to expose specimens of LMFBR cladding candidate alloys to a fast neutron flux and, subsequently, to evaluate the effect of the irradiation on material properties. All five austenitic alloys—Types 304, 316, 321, 347 and Incoloy-800—currently being considered for LMFBR cladding were included as both tubular and flat sheet tensile specimens. Both types of specimens have been thermally treated to effect a structural modification which is intended to improve high temperature dynamic properties after irradiation.

For convenience, the post-irradiation program for the series L-14 materials capsules will consist of two phases.

- An investigation of the irradiation performance of the capsules.
- Mechanical and physical property evaluation of materials specimens.

8.2.3.1 Capsule Description

The L-14 capsules were designed for insertion in an EBR-II Mark B-19 core subassembly (ANL drawing EB-1-38928D).

The contents of each capsule included seven test sections and a sodium reservoir attached coaxially by umbilical tubing, similar to the L-10 capsules⁽¹⁾. Each capsule was identified by a code designation (Table 8-1), and only one tubular test material was used per capsule. The test sections consisted of a 0.250-inch o.d. x 0.220-inch i.d. x 3.4-inch long tube fitted on both ends with open end plugs. Four test sections, the two on each end of the capsule insert, contain only tungsten rod for gamma heating. The remaining three test sections (located in the core region) contain two tensile specimens, tungsten holders for gamma heating, and a fusible metal thermometer capsule, called a sentinel. Flux wires, one iron and one nickel, were strung axially along the external surface of each test section to monitor the neutron flux.

TABLE 8-1

TUBULAR TEST MATERIALS IN CAPSULE L-10

<u>Capsule Designation</u>	<u>Material</u>	<u>Heat Treatment</u>
A & B	304 SS	1650°F, 24 h
C & D	304 SS	As received
E & F	316 SS	1650°F, 24 h
G & H	316 SS	As received
I	316 SS	2200°F, 24 h 2000°F, 1/2 h
J & K	321 SS	1650°F, 24 h
L	321 SS	As received
M	347 SS	1650°F, 24 h
N	347 SS	As received
O & P	Incoloy-800	1650°F, 24 h
Q & R	Incoloy-800	As received
S	Incoloy-800	2200°F, 24 h 2000°F, 1/2 h

8.2.3.2 Irradiation History

The L-14 capsules were irradiated in the EBR-II subassembly designated XO25. This subassembly was irradiated in Row 4 from October 1967 to December 1968. The XO25 subassembly has received 8,399 MWd of EBR-II exposure which is equivalent to $\sim 3.1 \times 10^{22}$ n/cm² total fluence in the peak core position. The total exposure time for XO25 was ~ 187 EFPD ($\sim 4,500$ hours).

Since thermocouples were not available during the irradiation, heat transfer

calculations were used to define the peak temperature during irradiation for each test section. The temperatures for individual test sections were similar to those calculated for L-10⁽¹⁾.

8.2.3.3 Post-Irradiation Examination

The first phase of the L-14 examination consisted primarily of non-destructive examination of the capsules and test sections, and disassembly of the

capsule. The steps taken and the observations made are presented in the following paragraphs.

The capsules were identified and visually examined. All of the capsules were straight and clean. The total package of 19 capsules were photographed as shown in Figure 8-2. The outside spiral wires were removed, and no residual sodium was observed on the capsule surfaces.

Gross gamma scans of capsules A, B, and O were made prior to sectioning to confirm the position of the different capsule materials. High resolution gamma scans were repeated over the locations of the temperature sentinels. In all cases the results indicated that the temperature of the highest melting alloy (1185°F, 640°C) had been attained or exceeded in all three capsules. Similar observations were made on the L-10 capsules.

Following sectioning and removal of the test inserts, the capsule tubes A, B, G, J, and N were gross gamma scanned. The gamma activity distributed along the length of the capsule body was used to determine the relative fluence along the axis of the core. All gamma energies above 0.5 MeV were counted, and the results of each capsule were normalized to obtain the relative axial distribution of the gamma

activity. The normalized results are plotted in Figure 8-3. The measured gamma activity was not symmetrical about the core midplane. The peak gamma activity occurred 1/4 inch below the core midplane (about 6-1/2 inches above the bottom of the core). Capsule A was situated closest to the core center during irradiation, while capsule N was farthest from the core center. The peak gross gamma activity was greater for capsule A than for capsule N, indicating about a 14% flux drop across the subassembly.

A spectral gamma scan of capsule tube A indicated that ^{60}Co (1.33 MeV), ^{58}Co (0.81 MeV) and ^{59}Mn (0.835 MeV) were the primary activation products present. ^{54}Mn and ^{58}Co result from the reactions $^{54}\text{Fe} (n, p) ^{54}\text{Mn}$ and $^{58}\text{Ni} (n, p) ^{58}\text{Co}$ which have threshold energies of approximately 3 MeV, and therefore are a good detector for the fast neutron flux.

^{60}Co originates from the reaction $^{59}\text{Co} (n, \gamma) ^{60}\text{Co}$, but its gamma activity corresponds to relative flux for softer epithermal neutrons. A comparison of the relative amounts of ^{60}Co , ^{58}Co , and ^{54}Mn activity along the capsule tube indicates that in the core region there is only a slight thermal flux contribution to the total flux. The fast flux (~3 MeV) makes the major contribution but rapidly falls off outside

FIGURE 8-2. L-14 CAPSULES PHOTOGRAPHED THROUGH HOT CELL WINDOW

FIGURE 8-3. NORMALIZED GROSS GAMMA SCANS OF L-14 CAPSULES

the core region. This relationship is shown in Figure 8-4. The spectral gamma scan data and the gross gamma scan for capsule A both show a similar asymmetrical gamma activity along the capsule length.

The tubular test sections were cut open, and the tungsten inserts, the temperature sentinels, and the tensile coupons were removed for further examination.

8.2.4 Cladding Swelling

The objective of these investigations is to monitor cladding swelling in available PA-10 irradiated specimens to establish the relationship between swelling and irradiation, alloy microstructure, and stress history. Techniques used in measuring cladding swelling are liquid immersion density, dimensional analysis, and transmission electron microscopy.

8.2.4.1 Water Immersion Density Technique

The density check performed on irradiated stainless steels has been modified to (a) make the sample specimen as small as practical and (b) to obtain a sufficient amount of data to place reasonable confidence limits on the measured density. A new "Torbal" analytical balance is being used to make the dry and immersed weighings required by the technique. The dry sample weights are reproducible within ± 0.1 mg, while wet immersed weights are reproducible within ± 0.2 mg. As the sample weight becomes smaller, the variations in these weighings

become increasingly important and have a larger and larger effect on the measured density. The smaller the sample, the greater will be the spread of density values that are obtained for that sample. This is shown in Table 8-2 for three different size samples. The large spread in the density data for the tensile coupons can be compensated for by taking a larger number of density checks. Approximately 10 or 12 complete density checks are necessary to place the density of a tensile coupon in a 95% confidence interval of $\pm 0.25\%$, while 6 sets of data are required to place the density of a 1 inch section of tubing into a 95% confidence interval of $\pm 0.1\%$. Both confidence intervals could be made smaller by making additional density measurements.

8.2.4.2 Density of As-Received Tubing

Density measurements were made on 14 pieces (2 inches long) of unirradiated Type-316 stainless steel tubing, randomly selected from the same heat of as-received material. The objective of these measurements was to determine the density variation to be expected in as-received stainless steel tubing presently used on PA-10 programs.

The samples were initially cleaned in acetone and methyl alcohol. Two complete sets of density data were obtained on each sample (Table 8-3). The maximum difference between the two density determinations for any one sample was 0.008 g/cc.

FIGURE 8-4. SPECTRAL GAMMA SCAN FOR CAPSULE A OF L-14

TABLE 8-2

SAMPLE SIZE VERSUS SPREAD IN DENSITY RESULTS

<u>Size</u>	<u>Approximate Weight (gm)</u>	<u>Difference Between Wet and Dry Weights (gm)</u>	<u>Variability In Measured Densities (%)</u>
2-1/2-inch length of 0.250-inch cladding	3.6	0.45	±0.04
1-inch length of 0.250-inch cladding	1.5	0.20	±0.1
tensile coupon	0.5	0.067	±0.3

The largest measured density was 8.012 gm/cc (Sample No. 14) and the lowest 7.987 gm/cc (Sample No. 4). This represents a maximum difference in density of 0.3%. Analysis of all the experimental data by conventional statistical procedures produced the following results:

Sample Mean ($\bar{X}$) = 8.000

Sample Standard Deviation (S_x) = 0.0058

Based on the distribution of the 28 experimental data points, the density of a randomly picked piece of tubing would be expected to fall within the range

$$\bar{X} \pm t_{m-1} S_x \quad (95\% \text{ confidence})$$

or

$$8.000 \pm (2.052) (0.0058)$$

or

$$8.000 \pm 0.0119 \text{ (95\% Confidence)}$$

A confidence interval estimate of the population mean density (μ_x) was made using the following formula:

$$P(\bar{X} - t_{m-1} S_x \leq \bar{X} + t_{m-1} S_x) = 0.95$$

where:

$$S_{\bar{x}} = S_x / \sqrt{m} = 0.0011$$

$$m = \text{sample size (28 data points)}$$

$$t_{m-1} = 2.052$$

$$P(8.000 \pm 0.002) = 0.95$$

The above statement predicts with 95% confidence that the interval 8.002 to

TABLE 8-3

DENSITY DATA ON AS RECEIVED 316 STAINLESS STEEL TUBING

<u>Sample No.</u>	<u>Density gm/cc</u>	<u>ΔP gm/cc</u>
1	8.006 8.004	0.002
2	8.001 7.999	0.002
3	8.005 7.999	0.006
4	7.987 7.995	0.008
5	8.012 8.005	0.007
6	8.005 7.999	0.006
7	7.996 7.998	0.002
8	7.995 7.996	0.001
9	7.995 7.998	0.003
10	7.993 8.001	0.008
11	8.005 7.999	0.006
12	8.000 7.994	0.006
13	8.005 8.001	0.004
14	8.007 8.012	0.005

7.998 gm/cc contains the actual mean density of the 316 stainless steel tubing.

8.2.4.3 Density of Irradiated L-10 Tubular Sections

The densities of six sections of 0.250-inch o.d. mill annealed tubing irradiated in the L-10 series have been measured. Each sample was 1-inch long and came from the same axial position in the core. The densities of the irradiated materials were compared against the densities of unirradiated control materials of the same heat number and heat treatment (Table 8-4). The estimated fluence and temperature of these sections during irradiation were 2.05×10^{22} n/cm², >0.1 MeV and ~1400°F, respectively.

Samples JA4 (Incoloy-800), JE4 (Type 316) and JH4 (Type 304) exhibited small changes [both positive (+0.16%) and negative (-0.03%)] in density. The exact unirradiated densities of these samples are not known but can be expected to vary within $\pm 0.15\%$. It appears then that little, if any, cladding swelling had occurred in these tubular sections. Samples JD4 (Type 316) and JJ4 (Type 321), exhibited density decreases slightly larger (0.30% to 0.45%) than those which can be attributed to variations in starting material. Profilometer

traces of the tubular test section also indicated that a large diameter increase (5.2% to 9.2%) had occurred during irradiation. Further examination of the microstructure of these materials is required to define the cause of the slight density decrease in these deformed test sections.

8.2.4.4 Alpha Decontamination of Irradiation Fuel Cladding

The purpose of the alpha-decontamination effort is to obtain cladding samples that (a) can be transferred from the alpha-cell to a beta-gamma cell and (b) are suitable for both density measurements and destructive mechanical properties tests. The majority of the fuel in a fuel-cladding section is removed by a drilling and reaming operation. However, a small quantity of alpha-emitting material remains as abrasive fuel particles embedded in the cladding surface. The contamination levels at this point are low enough to permit nondestructive measurements, but are not low enough to permit destructive mechanical property tests where the fuel particles could be released.

An electropolishing technique has been developed for removing the residual alpha-contamination from fuel cladding. The electropolishing is intended to dislodge

TABLE 8-4

DENSITY RESULTS OF L-10 CLADDING MATERIAL

Sample Size -- 1 in. length, 0.250 in. diameter
 Core Location of Sample -- 7.9 in. from bottom of core
 Flux Peak Location -- 6.9 in. from bottom of core
 Sample Fluence (>0.1 MeV) -- $\sim 2.8 \times 10^{22}$ n/cm²
 Operating Temperature -- $\sim 1400^{\circ}\text{F}$

Capsule	Material	Heat Number	Heat Treatment	Unirradiated Control Material	
				Mean Density (gm/cc)	95% Confidence Limits (gm/cc)
J <u>A</u> 4	I-800	4167	A. R. *	7.9357	± 0.0009
J <u>D</u> 4	316	850145	A. R.	7.9523	± 0.0005
J <u>E</u> 4	316	850145	A. R.	7.9747	± 0.0018
J <u>H</u> 4	304	125653	A. R.	7.8935	± 0.0009
J <u>I</u> 4	347	99973	A. R.	7.9025	± 0.0011
J <u>J</u> 4	321	31053	A. R.	7.8836	± 0.0010

Capsule	Irradiated Material			Max Change in Diameter (% $\Delta D/D_o$)	Designed In-Core Stress Level (psi)
	Mean Density (gm/cc)	95% Confidence Limit (gm/cc)	% Change in Density		
J <u>A</u> 4	7.9450	± 0.0056	+0.12**	0	12,000
J <u>D</u> 4	7.9286	± 0.0044	-0.30	9.2	13,000
J <u>E</u> 4	7.9720	± 0.0053	-0.03**	0	13,000
J <u>H</u> 4	7.9065	± 0.0053	+0.16**	0	8,000
J <u>I</u> 4	7.8920	± 0.0011	-0.13	3.2	12,000
J <u>J</u> 4	7.8478	± 0.0016	-0.45	5.2	10,000

*As-received material

**Profilometer traces indicated that tubes did not swell.

the embedded fuel by removing 2 or 3 mils of metal from the cladding inner diameter. Lengths of tubing from 1 to 2 inches can be electropolished, and metal is only removed from the inside surface. The electrolyte solution is allowed to flow by gravity between the central electrode and the cladding. By establishing the proper voltage and current between the cladding and electrode, metal is removed from the cladding and deposited on the electrode. Dislodged fuel particles are washed away by the electrolyte.

The apparatus (Figure 8-5) has been checked out using unirradiated tubing samples, and the results are encouraging. The equipment is presently being prepared for insertion into a beta-gamma cell and should be ready for operation in the third quarter of 1969.

8.3 MATERIAL MANAGEMENT AND STANDARDS

The materials management program is charged with the responsibility of assuring that the highest quality material (consistent with need) is provided for cladding of LMFBR fuel. In fulfilling this charter, the program has concentrated heavily on detailed characterization of tubing. This phase of the program serves two functions:

- To establish a basis for assessing the capability of industry to provide

a quality of tubing that exceeds standard ASTM specifications

- To provide experimenters with samples whose pre-irradiated physical properties are known.

8.3.1 Quality Control and Vendor Assessment

Although the latter function is a continuous one, the former is limited to the development of a tubing specification that reflects the needs of the nuclear industry and the capabilities of commercial tube fabricators. The assessment of tube fabricator capability was discussed in some detail in an earlier document⁽¹⁾. It will be summarized only briefly here.

Mill run quantities of seamless Types -304H and -316H tubing, 0.250 inch in diameter with a 0.015 inch wall, were procured from four commercial vendors and produced to virtually identical specifications. Characterization of all material with regard to chemistry, structure, dimensions, surface condition, and mechanical properties was performed on representative quantities of each heat. All experimental and analytical work involved in this assessment, except the mechanical properties studies, have been completed and are being incorporated in a topical report.

FIGURE 8-5. SKETCH OF α - DECONTAMINATION APPARATUS

Detailed procedures covering the general characterization of a tubing heat and the complete inspection of material prior to its use in irradiation experiments have been prepared. These procedures, which specify that complete records of all inspection steps and their results be documented, will be implemented on all future tubing requests. An outline of the general characterization inspection, which is intended to establish the overall quality of any given heat, was presented previously⁽²⁾. This inspection includes chemical analyses and metallography on a sample representative of each lot of material, dimensional analysis and surface condition on a statistical sampling from each lot, and mechanical properties evaluation. Tubing slated for subsequent experimentation, particularly that material used in fuel irradiation programs, will undergo 100% inspection. Steps in this inspection are itemized, along with criteria established for rejection, in Table 8-5.

8.3.2 Evaluation of Tubing Defects

Closely associated with the tubing characterization phase of the program is the study of the effect of wall thinning on the mechanical properties of cladding tubes. Documentation of the nature of naturally-

occurring defects was presented previously,⁽²⁾ as was an evaluation of the influence of artificially-produced defects on short-term biaxial properties⁽³⁾. A similar evaluation was performed on tubing containing natural surface defects. Tubes containing defects as deep as 10% of the wall were burst tested at 1100°F and 1300°F. The results of both test series showed no significant reduction in tangential rupture strength from that of undefected tubing of the same heat tested under identical conditions.

An investigation designed to assess the effect of defects on the long-term, low-strain rate mechanical properties of cladding tubes (a test more representative of in-service conditions) has been initiated. This experiment will test both artificially, and naturally, defected tubing selected from tube sections examined and fully characterized in the quality control phase of the program. Control and preliminary artificial defect testing are currently underway.

8.3.3 Test Technique Development

8.3.3.1 Biaxial Creep Facility

The biaxial creep facility was designed and fabricated to provide RML

TABLE 8-5

SEQUENCE ESTABLISHED FOR INSPECTION OF TUBING FOR
FUEL PIN FABRICATION

<u>Inspection Step</u>	<u>Basis for Rejection</u>
First Visual Examination	Excessive Bowing, Crimping, Flattening, etc., Obvious Scars, Scratches, etc.
Tube Section Identification	(None)
First Cleaning	(None)
Second Visual Examination	Any section containing o.d. surface defect equal to or greater than 0.015-inch deep.
I.D. Measurements	Minimum i.d. less than <u>0.2190</u> inch Maximum i.d. greater than <u>0.2210</u> inch
O.D. Measurements	Minimum o.d. less than <u>0.2490</u> inch Maximum o.d. greater than <u>0.2501</u> inch Ovality greater than <u>0.0020</u> inch
Wall Thickness Measurements	Minimum wall less than <u>0.0140</u> inch Maximum wall greater than <u>0.0160</u> inch Eccentricity greater than <u>0.0010</u> inch
Ultrasonic Examination	Any defect equal to or greater than the established standard.
Cutting and Facing	Length more than 1/32-inch shorter than nominal.
Straightness Examination	(None) At this point in the inspection, tubes will be straightened if bow exceeds 0.010 inch per linear foot.
Archive Storage	(None)
Cleaning and Passivation	(None)
Tube Disposition	(None)

with facilities to creep test irradiated tubing materials at elevated temperatures to provide creep rate data on highly irradiated tubular specimens. The equipment is being rewired for installation at GE-VNC where the final checkout of the equipment will be made starting in July.

8.3.3.2 Uniaxial Creep Facility

The uniaxial creep facility was designed and fabricated at GE's expense to provide RML with facilities to creep test irradiated flat sheet type specimens to provide creep rate data on highly irradiated tensile specimens.

All equipment is on order and delivery is expected in July. This facility will be used initially to creep test selected L-10 and L-14 tensile specimens.

8.3.3.3 Stress-Rupture Rig

A stress-rupture facility was put into operation whereby 54 concurrent tests on tubular sections are being performed under long term creep conditions. This facility will be used in the materials management programs for evaluation of creep rupture properties of unirradiated tubing.

The stress-rupture unit, is capable of testing the stress-rupture properties of

54 tubular specimens (1/4-inch diameter) simultaneously at one temperature, under one of six pressures. Nine specimens are placed in one of six pressure zones, with each zone having its own pressure regulator and pressure gage. Failure of each specimen is automatically recorded. The remaining eight specimens in the pressure zone are unaffected. The failed specimen is relatively easily removed by disconnecting Swagelok fittings. A new specimen may then be inserted (if the existing pressure is satisfactory), or the low pressure tubing (specimen housing) can be capped off.

8.4 REFERENCES

1. "Sodium-Cooled Reactor, Fast Reactor Development Program, Thirtieth Quarterly Report, February - April 1969," (GEAP 10028).
2. "Sodium-Cooled Reactor, Fast Reactor Development Program, Twenty-eighth Quarterly Report, August-October 1968," (GEAP-5700).
3. Sodium-Cooled Reactor, Fast Reactor Development Program, Twenty-Seventh Quarterly Report, May - July 1968, " (GEAP-5677).

APPENDIX

PRIOR REPORTS

Prior progress reports to the Commission under this contract include:

Monthly Progress Letters, Number 1 — 94, from July 1959 through October 1968.

GEAP-3888	FCR Development Program - First Quarterly Report, October — December 1961.	GEAP-4640	FCR Development Program - Eleventh Quarterly Report, April — June 1964.
GEAP-3957	FCR Development Program - Second Quarterly Report, January — March 1962.	GEAP-4723	FCR Development Program - Twelfth Quarterly Report, July — September 1964.
GEAP-3981	FCR Development Program - Third Quarterly Report, April — June 1962.	GEAP-4795	FCR Development Program - Thirteenth Quarterly Report, October 1964 — January 1965.
GEAP-4080	FCR Development Program - Fourth Quarterly Report, July — September 1962.	GEAP-4860	FCR Development Program - Fourteenth Quarterly Report, February — April 1965.
GEAP-4158	FCR Development Program - Fifth Quarterly Report, October — December 1962.	GEAP-4931	FCR Development Program - Fifteenth Quarterly Report, May — July 1965.
GEAP-4214	FCR Development Program - Sixth Quarterly Report, January — March 1963.	GEAP-4982	FCR Development Program - Sixteenth Quarterly Report, August — October 1965.
GEAP-4300	FCR Development Program - Seventh Quarterly Report, April — June 1963.	GEAP-5098	FCR Development Program - Seventeenth Quarterly Report, November 1965 — January 1966.
GEAP-4382	FCR Development Program - Eighth Quarterly Report, July — September 1963.	GEAP-5158	FCR Development Program - Eighteenth Quarterly Report, February — April 1966.
GEAP-4480	FCR Development Program - Ninth Quarterly Report, October — December 1963.	GEAP-5198	FCR Development Program - Nineteenth Quarterly Report, May — July 1966.
GEAP-4601	FCR Development Program - Tenth Quarterly Report, January — March 1964.	GEAP-5292	FCR Development Program - Twentieth Quarterly Report, August — October 1966.
		GEAP-5433	FCR Development Program - Twenty-First Quarterly Report, November 1966 — January 1967.

- GEAP-5491 FCR Development Program - Twenty-Second Quarterly Report, February—April 1967.
- GEAP-5522 FCR Development Program - Twenty-Third Quarterly Report, May—July 1967.
- GEAP-5541 FCR Development Program - Twenty-Fourth Quarterly Report, August—October 1967.
- GEAP-5584 FCR Development Program - Twenty-Fifth Quarterly Report, November 1967—January 1968.
- GEAP-5631 FCR Development Program - Twenty-Sixth Quarterly Report February—April 1968.
- GEAP-5677 FCR Development Program - Twenty-Seventh Quarterly Report May—July 1968.
- GEAP-5700 FCR Development Program - Twenty-Eighth Quarterly Report August—October 1968
- GEAP-5753 FCR Development Program - Twenty-Ninth Quarterly Report, November 1968—January 1969.
- GEAP-10028 FCR Development Program - Thirtieth Quarterly Report, February—April 1969.

Supplementary Program Letters,
Nos. 1—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 300 MWe Reactor Core; P. Greebler, P. Aline, and J. Sueoka (November 10, 1959).
- GEAP-3347 Fast Oxide Breeder - Stress Considerations in Fuel Rod Design; K. M. Horst (March 28, 1960).
- GEAP-3486 Fast Oxide Breeder Project - Fuel Fabrication

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

Part II - Fabrication of Plutonium - Uranium Dioxide Specimens by Swaging; M. E. Snyder and W. C. Cowden (August 15, 1960).
- GEAP-3487 Fast Oxide Breeder - Preliminary Sintering Studies of Plutonium - Uranium Dioxide Pellets; J. M. Cleveland and W. C. Cavanaugh (August 15, 1960).
- GEAP-3646 Calculation of Doppler Coefficient and Other Safety Parameters for a Large Fast Oxide Reactor; P. Greebler, B. A. Hutchins, and J. R. Sueoka (March 9, 1961).
- GEAP-3721 Core Design Study for a 500 MWe Fast Oxide Reactor; K. M. Horst, B. A. Hutchins, F. J. Leitz, and B. Wolfe (December 28, 1961).
- GEAP-3824 Fabrication Cost Estimate for UO₂ and Mixed PuO₂-UO₂ Fuel; G. D. Collins January 24, 1962).

- GEAP-3833 The Post-Irradiation Examination of a $\text{PuO}_2\text{-UO}_2$ Fast Reactor Fuel; J. M. Gerhart (November 1961).
- GEAP-3856 Experimental Fast Oxide Reactor; K. P. Cohen, M. J. McNelly, and B. Wolfe (November 27, 1961).
- GEAP-3876 Plutonium Fuel Processing and Fabrication for Fast Ceramic Reactors; H. W. Alter, G. D. Collins, and E. L. Zebroski (February 1, 1962).
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- GEAP-4058 Analytical Studies of Transient Effects in Fast Reactor Fuels; R. B. Osborn and D. B. Sherer (August 1962).
- GEAP-4090 FORE - A Computational Program for the Analysis of Fast Reactor Excursions; P. Greebler and D. B. Sherer (October 1962).
- GEAP-4092 Doppler Calculations for Large Fast Ceramic Reactors, Effects of Improved Methods and Recent Cross Section Information; P. Greebler and E. Goldman (December 1962).
- GEAP-4130 Experimental Studies of Transient Effects in Fast Reactor Fuels. Series I, UO_2 Irradiations; J. H. Field (November 15, 1962).
- GEAP-4226 Conceptual Mechanical Design of 565 MWe Fast Ceramic Reactor; A. G. Silvester (April 1963).
- GEAP-4271 Measurement of Oxygen-to-Metal Ratio in Solid Solutions of UO_2 and PuO_2 ; W. L. Lyon (May 31, 1963).
- GEAP-4283 Experimental Studies of Sodium Logging in FCR Fuels; G. L. O'Neill, P. E. Novak, M. L. Johnson, and W. E. Baily (September 1963).
- GEAP-4350 Fast Reactor Meltdown Excursions - Analysis and Computer Program FARM I; B. C. Luci, D. R. Riley, and N. F. Friedman (December 5, 1963).
- GEAP-4410 Calculations of Doppler and Sodium Reactivity Effects; P. Greebler (November 1963).
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- GEAP-4418 Liquid Metal Fast Breeder Reactor Design Study, 1000 MWe PuO₂-UO₂ Fueled Plant, Vols. I and II; M. J. McNelly, et al. (January 1964).
- GEAP-4420 Influence of Doppler Effect on Meltdown Accident; N. F. Friedman (December 1963).
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CONTRIBUTORS

TASK B—DEFECTED FUEL BEHAVIOR

<u>BRDO</u>	<u>IPO</u>
K.E. Gregoire	D.L. Brown
R.E. Murata	R.H. Eymann
P.E. Novak	F.J. Lamphere
D.E. Plumlee	E.D. Ormiston
	J.P. Peterson
	V.D. Springer
	G.L. Stimmel
	G.W. Tunnell
	L.W. Waters
	E.E. Walker
	D.L. Zimmerman

TASK E—FUEL PROPERTIES AND PERFORMANCE EVALUATION

P.E. Bohaboy	K.J. Perry
A.L. Barzin	H.S. Rosenbaum
G.F. Melde	B.F. Rubin
D.A. Cantley	P.E. Ray
E.A. Aitken	F.H. Scott
W. Conant	M.L. Thompson
R.N. Duncan	
R.H. Eymann	
J. Lewis	

TASK F—INTEGRAL FAST FLUX IRRADIATIONS

TASK C—SAFETY EXPERIMENTATION

<u>BRDO</u>	<u>IPO</u>
J.H. Field	G.W. Tunnell
H.C. Pfefferlen	D.L. Brown
T. Hikido	D.E. Hafer
G.R. Thomas	R.H. Eymann
R.C. Thompson	F.J. Lamphere
D.D. Freeman	G.C. Linder
	G.L. Stimmell

<u>NFD</u>	<u>RML</u>
M.L. Thompson	R.E. Smith
G.C. Berg	D.L. Orton
S.F. Cefalu	N.T. Corege
	D.G. Karris
	S.M. Lukezic
	D.L. Zimmerman

<u>BRDO</u>	<u>PFD</u>
W.E. Baily	G.C. Berg
A.L. Barzin	T.A. Koppel
P.E. Bohaboy	E.F. Kurtz
D.A. Cantley	E.J. McAlpine
C.N. Craig	C.M. Ryer
R.N. Duncan	M.L. Thompson
G.R. Hull	
R.C. Nelson	
K.J. Perry	
P.E. Ray	
F.H. Scott	

<u>IPO</u>	<u>MS&P</u>
R.H. Eymann	E.A. Aitken
R.M. Irvine	S.K. Evans
F. Lamphere	W.H. Kimball
J.H. Morgan	H.S. Rosenbaum
D.L. Zimmerman	

TASK G—CORE NEUTRONICS, THERMAL
HYDRAULICS AND SAFETY
ANALYSIS

F. G. Aquino	S. Crick
C. F. Barrett	C. Cowan
D. J. Bender	N. Evatt
J. R. Cox	D. Freeman
J. H. Field	P. Greebler
D. P. Hines	T. Hitchcock
L. Lee	B. Hutchins
P. M. Magee	M. Kelley
H. Matsuo	P. Linder
L. M. McWethy	L. Price
W. G. Meinhardt	R. Protsik
H. Moddison	S. Seymour
S. Oldberg	S. Stewart
S. Sangster	J. Turner

J. D. Stephen	P. Vaughan
R. G. Stuart	J. Douglass
G. R. Thomas	R. Thiel
R. C. Thompson	G. Leff
S. Weber	A. Miller

TASK L—CLADDING DEVELOPMENT

J. J. Albanese	R. H. Eymann
R. R. Asamoto	J. P. Ferguson
J. S. Armijo	D. H. Frisby
C. F. Barrett	T. Lauritzen
L. Brown	A. Montes
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Mr. L. W. Fromm, Manager 1000 MWe LMFBR Follow-On Study Project Building 208 Argonne National Laboratory 9800 South Cass Avenue Argonne, Illinois 60439	2	RDT Site Office U.S. Atomic Energy Commission Atomic Power Development Associates, Inc. 1911 First Street Detroit, Michigan 48226	1
Mr. C.A. Anderson, Project Mgr 1000 MWe LMFBR Follow-On Study Westinghouse Electric Corporation Advanced Reactors Division Waltz Mill Site P.O. Box 158 Madison, Pennsylvania 15663	2	RDT Senior Site Representative Oak Ridge National Laboratory P.O. Box X Oak Ridge, Tennessee 37830	1
FFTF Project P.O. Box 220 Richland, Washington 99352 Attn: Configuration & Data Management	8	W.B. Cottrell, Director Nuclear Safety Program Oak Ridge National Laboratory P.O. Box Y Oak Ridge, Tennessee 37830	2
W. P. Staker, Project Manager 1000 MWe LMFBR Follow-on Study Combustion Engineering, Inc. P.O. Box 500 Windsor, Connecticut 06095	2	RDT Senior Site Representative U.S. Atomic Energy Commission P.O. Box 550 Richland, Washington 99352	1

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Atomics International P. O. Box 309 Canoga Park, California 91304 Attn: FBR Project Manager (R. J. Beeley)	1	Detroit Edison Company 1911 First Street Detroit, Michigan 48226 Attn: A. S. Griswold	1
Dow Chemical Company Rock Flats Division P. O. Box 888 Golden, Colorado 80401 Attn: R. D. Forest	1	Atomic Power Development Associates 1911 First Street Detroit, Michigan 48226 Attn: A. A. Shoudy	1
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Dr. John C. Woodhouse 1 Guest Lane Wilmington, Delaware 19809	1	Westinghouse Electric Corporation Advance Reactors Division P. O. Box 217 Cheswick, Pennsylvania 15656 Attn: W. E. Ray	1
Power Reactor Development Corp. 1911 First Street Detroit, Michigan 48226 Attn: W. J. McCarthy	1	Nuclear Materials & Equipment Corp. Plutonium Laboratory Leechburg, Pennsylvania 15656 Attn: William J. Ross	1