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**SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR
DEVELOPMENT PROGRAM
TWENTY-EIGHTH QUARTERLY REPORT
AUGUST-OCTOBER 1968**

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

ADVANCED PRODUCTS OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

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SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

Twenty-Eighth Quarterly Report
August-October 1968

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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 August 1, 1968 to October 31, 1968 on the currently active tasks of this program is described in subsequent sections.

This is the twenty-eighth 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 under this contract include the following:

Monthly Progress Letters, Number 1-91, from July 1959 through July 1968.

- GEAP-3888 FCR Development Program - First Quarterly Report, October-December 1961.
- GEAP-3957 FCR Development Program - Second Quarterly Report, January-March 1962.
- GEAP-3981 FCR Development Program - Third Quarterly Report, April-June 1962.
- GEAP-4080 FCR Development Program - Fourth Quarterly Report, July-September 1962.

- GEAP-4158 FCR Development Program - Fifth Quarterly Report, October-December 1962.
- GEAP-4214 FCR Development Program - Sixth Quarterly Report, January-March 1963.
- GEAP-4300 FCR Development Program - Seventh Quarterly Report, April-June 1963.
- GEAP-4382 FCR Development Program - Eighth Quarterly Report, July-September 1963.
- GEAP-4480 FCR Development Program - Ninth Quarterly Report, October-December 1963.
- GEAP-4601 FCR Development Program - Tenth Quarterly Report, January-March 1964.
- GEAP-4640 FCR Development Program - Eleventh Quarterly Report, April-June 1964.
- GEAP-4723 FCR Development Program - Twelfth Quarterly Report, July-September 1964.
- GEAP-4795 FCR Development Program - Thirteenth Quarterly Report, October 1964-January 1965.
- GEAP-4860 FCR Development Program - Fourteenth Quarterly Report, February-April 1965.
- GEAP-4931 FCR Development Program - Fifteenth Quarterly Report, May-July 1965.
- GEAP-4982 FCR Development Program - Sixteenth Quarterly Report, August-October 1965.
- GEAP-5098 FCR Development Program - Seventeenth Quarterly Report, November 1965-January 1966.
- GEAP-5158 FCR Development Program - Eighteenth Quarterly Report, February-April 1966.
- GEAP-5198 FCR Development Program - Nineteenth Quarterly Report, May-July 1966.
- 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.

Supplementary Program Letters, No. 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 UO_2 and Mixed $\text{PuO}_2\text{-UO}_2$ 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).
- GEAP-3880 Comparative Study of PuC-UC and $\text{PuO}_2\text{-UO}_2$ as Fast Reactor Fuel
Part I - Technical Considerations; K. M. Horst and B. A. Hutchins (February 15, 1962).
Part II - Economic Considerations; G. D. Collins (November 15, 1962).
- GEAP-3885 Experimental Fast Ceramic Reactor Design, Status Report as of October 31, 1961; Edited by K. M. Horst (April 24, 1962).
- GEAP-3923 Resonance Integral Calculations for Evaluation of Doppler Coefficients - The RAPTURE Code; J. H. Ferziger, P. Greebler, M. D. Kelley, and J. Walton (June 12, 1962).
- GEAP-4028 A Fuel Reprocessing Plant for Fast Ceramic Reactors; H. W. Alter (February 1, 1962).
- GEAP-4058 Analytical Studies of Transient Effects in Fast Reactor Fuels; R. B. Osborn and D. B. Sherer (August 1962).
- GEAP-4090 FORE - A Computational Program for the Analysis of Fast Reactor Excursions; P. Greebler and D. B. Sherer (October 1962).
- GEAP-4092 Doppler Calculations for Large Fast Ceramic Reactors, Effects of Improved Methods and Recent Cross Section Information; P. Greebler and E. Goldman (December 1962).
- GEAP-4130 Experimental Studies of Transient Effects in Fast Reactor Fuels. Series I, UO_2 Irradiations; J. H. Field (November 15, 1962).
- GEAP-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).
- GEAP-4414 Reactor Safety and Fuel Cycle Economics, K. P. Cohen (November 1963).
- GEAP-4418 Liquid Metal Fast Breeder Reactor Design Study, 1000 MWe PuO_2 - UO_2 Fueled Plant, Vols. I² and II; M. J. McNelly, et al. (January 1964).
- GEAP-4420 Influence of Doppler Effect on Meltdown Accident; N. F. Friedman (December 1963).
- GEAP-4466 Experimental Studies of Plutonium and Fission Product Redistribution in Fast Reactor Fuels; T. Lauritzen, et al., (July 1966).
- GEAP-4471 Calculated Nuclear Parameters and Their Uncertainties in a 1000 MWe FCR; P. Greebler, et al., (October 1966).
- GEAP-4472 User Requirements for Cross Sections in the Energy Range from 100 eV to 100 keV; P. Greebler and B. A. Hutchins (1966).
- GEAP-4580 FLUZ - A Computer Code for Analyzing the Reactor Response to Small Reactivity Oscillations; P. Greebler, B. A. Hutchins, and M. D. Kelley (March 22, 1965).
- GEAP-4630 The Effects of Resonance Overlap on the Doppler Coefficient in a Fast Ceramic Reactor, Computer Codes RAVE and DOPIE; B. A. Hutchins (August 1964).
- GEAP-4675 Some Electrical Properties of the Plutonium-Uranium Mixed Oxide System; W. E. Bailly (July 1966).
- GEAP-4770 A Technical and Economic Evaluation of Vented Fuel for Sodium-Cooled Fast Ceramic Reactors; G. L. O'Neill, J. Duffy, J. C. Gilbertson, F. W. Knight, and D. B. Sherer (May 1965).
- GEAP-4787 An Analysis of Fast Reactor Transient Response and Safety in Selected Accidents; W. Meinhardt, Jr. (August 1966).
- GEAP-4804 Experimental Studies of Transient Effects in Fast Reactor Fuels, Series II; J. E. Hanson, J. H. Field, and S. A. Rabin (June 1965).
- GEAP-4819 Experimental Studies of Transient Effects in Fast Reactor Fuels, Series III, Pre-Irradiated Mixed Oxide (PuO_2 - UO_2) Irradiations, Interim Report, Steady State Irradiations; J. E. Hanson and S. A. Rabin (September 1965).
- GEAP-4822 Studies of Alternate Core Design Concepts for Fast Sodium Cooled Reactors; C. L. Cowan, (November 1966).
- GEAP-4870 Some Design and Irradiation Experience with Capsules for Long Organic Fuel Elements; F. W. Knight (July 1966).
- GEAP-4877 Diffusion of Plutonium and Uranium in Oxide Fuel at 2400°C; J. Davies and P. Novak (August 1966).
- GEAP-4878 The Solid-Liquid Phase Diagram for UO_2 - PuO_2 System; W. L. Lyon and W. E. Bailly (May 1966).
- GEAP-4897 Irradiation Behavior of Hyper- and Hypo-Stoichiometric Plutonia-Urania Fuel at 70,000 MWd/T; E. L. Zebroski, W. E. Bailly, and D. P. Hines (March 1965).
- GEAP-4922 BISYN - A Two Dimensional Synthesis Program; M. D. Kelley (December 1966).
- GEAP-4939 Safety and Economic Characteristics of a 1000 MWe Fast Sodium Cooled Reactor Design; K. P. Cohen and G. L. O'Neill, (April 1967).
- GEAP-5272 Evaluation and Computation of Pu-239 Cross Section Data for the ENDF/B Files; P. Greebler, P. Aline, D. Hutchins (January 1967).
- GEAP-5273 FORE II, A Computational Program for the Analysis of Steady-State and Transient Reactor Performance; B. E. Lawler, et al., (September 1967).
- GEAP-5340 Neutron Spectrum Measurements in a Fast Plutonium Assembly ZPR-III Assembly 48; P. S. Brown and A. Weitzberg (April 1967).
- GEAP-5494 RAPFU—A Computer Code; P. Greebler, C. Cowan (November 1967).
- GEAP-5411 Turbulent Velocity Distribution in a Rod Bundle; D. J. Bender, D. M. Switick, J. H. Field (January 1968).

- GEAP-4879 Thermal Conductivity of Uranium and Uranium Plutonium Oxides; T. E. Lannin (January 1968).
- GEAP-5570 Short Term Fast Flux (EBR-II) Irradiation of $\text{PuO}_2\text{-UO}_2$ Fuel Pins; S. A. Rabin, W. W. Kendall, W. E. Baily (February 1968).
- GEAP-4469 Experimental Studies of Transient Effects in Fast Reactor Fuels, Series III, Pre-Irradiated Mixed Oxide ($\text{PuO}_2\text{-UO}_2$), Irradiations Final Report, Transient Irradiations; J. E. Hanson, J. H. Field (February 1968).
- GEAP-5344 Electron Microanalysis of Radioactive Materials I, Preliminary Experiments and Plans for a Shielded Facility; H. Rosenbaum, et al., (February 1968).
- GEAP-5271 Implications of Recent Fast Critical Experiments on Basis Fast Reactor Design; P. Grebler, (March 1968).
- GEAP-5493 The Effect of Density on the Thermal Thermal Conductivity of Uranium Dioxide; R. Asamoto, (February 1968).
- GEAP-5556 In-Pile Comparison of Effective Thermal Conductivity (Pu-UO_2) Mixed Oxides; C. N. Craig, W. E. Baily (March 1968).
- GEAP-5635 Significance of Neutron Data Uncertainties to Fast Reactor Economics and Power Plant Design; B. A. Hutchins, et al., (June 1968).

SECTION II

SUMMARY

2.1 TASK B—DEFECTED FUEL BEHAVIOR

AEC approval was received for the fabrication of the first three-pin forced convection capsule (FCC) for test B9A. Thermal analysis of the capsule and fuel pin was completed and fabrication of parts was initiated. Further out-of-pile notched cladding tests were completed using UO_2 pellets and steel mandrels to establish reliability of the defecting mechanism at power. The steel mandrel tests, conducted at room temperature and 1100°F , confirmed earlier work which indicated that the differential fuel-cladding thermal expansion defect mechanism will produce clad failure after a diameter change of less than 0.001 inch. Fabrication of an in-pile proof test of this defect mechanism was initiated.

The first calibration test of the sodium-water reaction experiment ("Soda-Pop") was successfully conducted with 100 grams of NaK.

Interim examination of the sodium-bonded oxide fuel pins in capsule B8C by neutron radiography indicated that the cladding is intact and the fuel column is continuing to expand axially into the plenum region of both pins. This expansion appears to be linear with burnup with a $\Delta V/V$ of approximately 1%/10,000 MWd/Te. Peak burnups in these experiments are 96,000 and 48,000 MWd/Te on the top and bottom pins, respectively. Interim neutron radiography of the second sodium bonded oxide capsule (B8D) also indicates that the cladding is intact. The three pins in this capsule have reached burnups of approximately 25,000 MWd/Te, and as in

capsule B8C, they indicate an apparent sodium-fuel reaction product at the top of the fuel column.

2.2 TASK C—SAFETY EXPERIMENTATION

Post-irradiation examination of the axially-restrained powder fuel specimens is nearly complete. The failed specimen (C4J) shows that 50 to 60% of the fuel melted during the transient, with fuel relocation from both the upper and lower ends of the pin. Cladding examination indicates possible vapor blanketing above and below the primary failure zone.

Transient irradiation of two 24-inch long, pre-irradiated specimens (C5A, B) was completed. Equivalent peak powers reached in the transients were 160 and 155 kW/ft, respectively. In one pin, neutron radiographs show extensive cladding failure and ejection of molten fuel into the surrounding annulus. The second pin, which was identical except for a 0.070-inch central hole extending through the blanket, was exposed to a similar transient without failure. Radiographs reveal that molten fuel moved upward approximately 10 inches into the annular blanket.

Fabrication was initiated on two 3-pin capsules (C6A, B) designed to compare defected or sodium-bonded pellet and powder fuel under transient conditions.

The mobile data acquisition system for the Forced Convection Capsule has been assembled and is being utilized on the "Soda-Pop" tests.

2.3 TASK E—FUEL PERFORMANCE EVALUATION

Examination of the annular and solid pellet fuel pins from capsule E2J after a burnup of greater than 100,000 MWd/Te indicates no significant difference in performance. Fission gas release was in excess of 80% from both fuel types and is comparable to that obtained from earlier high-burnup thermal flux irradiations.

Capsule E1AA (In-Pile Measurement of Mixed Oxide Thermal Conductivity) was removed from GETR following failure of the tungsten-rhenium center fuel thermocouple. Before failure, the center fuel thermocouple indicated a slight increase in center temperature with exposure.

Capsule E5F (Linear Fuel Movement) successfully achieved a burnup of 50,000 MWd/Te and is continuing its irradiation to a target burnup of 100,000 MWd/Te. Data obtained from axial fuel expansion indicate that fuel-cladding interaction and axial fuel movement may be affected by thermal cycling.

Plasticity measurements on UO_2 were completed with results showing that under conditions where viscous creep occurs, the creep rate is (1) linear with stress, (2) increases with decreasing grain size, and (3) increases with decreasing density.

Capsules E5C, E5D and E5E (High Burnup Comparison of Solid Pellet, Powder, and Annular Pellet Fuels) have successfully achieved a burnup of 100,000 MWd/Te at linear powers of 19.5 kW/ft.

2.4 TASK F—INTEGRAL FAST FLUX IRRADIATIONS

The first unencapsulated fuel pins to be irradiated in EBR-II were included in S/A XO40, which was inserted into EBR-II during August. This subassembly contained the 16 Group 3A fuel pins of 0.290-inch diameter that have been at the EBR-II site since June 1967.

Post-irradiation examination of fuel pins F2C, H, R, T (~70,000 MWd/Te; S/A XG05) continued. Metallographic examination of the capsule containing fuel pin F2H showed the structure to be normal, with no evidence of deleterious microstructural characteristics. Overall length increases in the four pins ranged from 0.44% to 0.53%.

The Group-0 capsule in S/A XG02 reached the target burnup of 40,000 MWd/Te and was removed from the reactor for post-irradiation examination.

Reinsertion (in S/A XO39) of seven Group 4 capsules previously irradiated to 35,000 MWd/Te in S/A XO11 was accomplished. The 19 Group 8B (low power fuel swelling) encapsulated specimens were incorporated into S/A XO36 and inserted in EBR-II. The accumulated burnups of Groups 0, 3A, 3B, 4, 6, 7, 8A and 8B are calculated to be 43,200; 7,500; 26,800; 41,600; 43,700; 44,500; 26,000; and 5,700 MWd/Te, respectively.

As a consequence of the increase in EBR-II power level from 45 to 50 MW, the Group 5 (elevated cladding temperature) target conditions of 1185°F peak-cladding surface temperature can now be obtained in a standard Mark A, 19-capsule subassembly. Efforts to optimize the mixing plenum length requirements for the cold sodium by-pass subassembly will continue in anticipation of its use for the Group F10 experiments.

The hazards analysis for the Group 9A unencapsulated fuel pins (0.250-inch diameter) was completed and submitted to ANL for review. Fabrication of the Group 9C fuel pins is completed and fabrication is continuing on Groups 9B and 9D.

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

Computations of cross sections in the GAM-1 group structure with the ENDRUN code show excellent agreement with results from the codes MC², MUFT-4, GAM-1, and GAF-GAR codes. An analysis of resonance parameters for Pu-239 using energy-dependent neutron and fission widths has resulted in good agreement between measured and computed fission and total cross sections for parameters which are consistent with the ORNL-RPI alpha values.

The two-dimensional diffusion code 2DB and the two-dimensional transport theory code DOT have been combined to provide a single versatile nuclear analysis tool. This code will be capable of performing directly comparable, two-dimensional diffusion and transport theory calculations with greater efficiency than previously possible. The new code, called DOT2DB, is being debugged.

A heat transfer model has been developed to investigate the consequences of fault conditions in typical fast reactor fuel bundles. The results of the first accident studied indicate that sodium boiling is not likely to occur from the release of fission gas from one fuel pin.

A critical review of the VELVET code has uncovered a programming error. The revised velocity profiles were found to be inconsistent with available experimental data. A new calculational procedure was developed based on recent Soviet

studies. The results are now in good agreement with measured velocity distributions for rod bundles with pitch-to-diameter ratios of 1.05 to 1.20.

2.6 TASK L—CLADDING DEVELOPMENT

Irradiation of L-2I, L-6, L-14, and L-15 material capsules continued in EBR-II. Ten L-10 material capsules have been discharged from EBR-II after reaching their target fluence. Total fluence (EBR-II physics), as of September 30, 1968, is 4.8, 3.8, 2.8, 2.2 and 0.03×10^{22} n/cm² for the L-2I, L-6, L-10, L-14 and L-15 specimens respectively.

No significant difference in circumferential strain to fracture was found between irradiated Types 304 and 316L in the annealed condition when measured with the

ring ductility test. These results agree with those previously obtained from tensile results. Biaxial burst testing at 1300°F was successfully completed on clad from mixed oxide fuel pins irradiated in EBR-II.

Density measurements on samples irradiated in EBR-II (S/A XG05) at a total fluence of 5.7×10^{22} n/cm² (EBR-II physics) were performed with values ranging from 1.13 to 2.06% $\Delta\rho/\rho$. Observations of cavities by transmission electron microscopy were performed on Types 304, 347, and Incoloy 800. Cavity volumes in annealed 304 and Incoloy 800 were in close agreement with measured volume increases.

Helium analyses performed on irradiated tensile coupons show substantially higher helium concentration than could be accounted for by physics calculations.

TABLE 2-1

TASK B—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

Designation	Purpose and Major Parameters	Design Parameters							Operating Conditions					Status
		Fuel							Irradiation Facility	Cladding Surface Temperature (°F)	Peak Rating (kW/ft)	Burnup (MWd/Te)		
		Diameter (inch)	Length (inch)	Density (Smear) (% TD)	Composition			Cladding Material						
					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	83	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	15	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	96,000	Irradiation in progress.	
		0.204	2.25	84	0.017*	25	1.97					48,000		
B8D	Fuel-sodium compatibility versus density and stoichiometry. Three pins.	0.214	1.75	80-88	0.0065	40	1.97	Type 316 stainless steel	V-Raft GETR	950	17.5	23,400	Irradiation in progress.	
		0.214	1.75	80-88	0.0065	25	1.97					25,300		
			0.214	1.75	80-88	0.0065	25	1.94						23,700

GEAP-5700

*Large gap required to accommodate sampling device

TABLE 2-2

TASK C—CURRENT EXPERIMENTAL TEST SERIES

Fuel Specimen	Description*	Fabrication	Pre-Irradiated**		TREAT-Transient Irradiated		Nondestructive Examination†	Destructive Examination	Test Objective
			Yes	No	Calibration	Test			
C4A	UO ₂ -20% PuO ₂ fuel 90% smeared density pelleted fuel	C		X	C	C	C	C	Axial restraint at fuel melting (~ 5000°F).
C4B	Identical to C4A	C		X	C	C	C	C	Axial restraint similar to C4A. Test to identify vol % melting necessary to cause cladding failure.
C4C	UO ₂ -20% PuO ₂ fuel 80% smeared density pelleted fuel	C		X	C	C	C	C	Axial restraint similar to C4A. Test to observe effect of fuel density on incipient failure point identified in C4A.
C4D ⁺⁺	Identical to C4C	C		X	C	C	C	C	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).
C4E (57A)	UO ₂ -28% PuO ₂ fuel 80% smeared density pelleted fuel	C	P		P	P	<div><div>P+++</div><div>P</div></div>	P	Axial restraint designed to simulate the restraint of C4A after steady-state irradiation is complete. Specific transient test goals will be determined after nondestructive examination of the fuel pin, following its removal from EBR-II.
C4F (F7B)	Identical to C4E	C	P		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4G (F7C)	UO ₂ -28% PuO ₂ fuel 90% smeared density pelleted fuel	C	I		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4H (F7D)	Identical to C4G	C	I		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4I ⁺⁺	UO ₂ -20% PuO ₂ fuel 85% dense powder fuel	C		X	C	C	C	NC	Axial restraint similar to C4A. Test transient is intended to be similar to that of C4D.
C4J	Identical to C4I	C		X	C	C	C	C	Axial restraint similar to C4A. Test transient is intended to be similar to that of C4A.
C4K (F7E)	UO ₂ -28% PuO ₂ fuel 85% dense powder fuel	C	P		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C4L (F7F)	Identical to C4K	C	P		P	P	<div><div>P</div><div>P</div></div>	P	Identical to C4E.
C5A	10% enriched UO ₂ 90% smeared density pelleted fuel	C	C		C	P	<div><div>C</div><div>P</div></div>	P	Test to study mobility of transient melted fuel: investigate potential for fuel relocation under prototypical conditions of length, restraint, and fission gas inventory.
C5B	Identical to C5A	C	C		C	P	<div><div>C</div><div>P</div></div>	P	Same as C5A.

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

*—O/M Ratio 1.97 to 1.985 for all cases
**—~ 50,000 MWd/Te—EBR-II fast flux for Series IV specimens; ~ 30,000 MWd/Te—GETR thermal flux for Series V specimens
†—Consists of x rays, neutron radiographs, deencapsulation, cladding diametral measurements, and γ-transmission scan
++—Contains a small, lower fuel assembly during transient irradiation. This lower assembly contains a W-5% Re, W-26% Re in-fuel thermocouple

+++

Before Transient

After Transient

TABLE 2-3
TASK E—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

		Design Parameters						Operating Conditions						Status
		Fuel						Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)		
		Diameter (inch)	Fuel Length (inch)	Density, % TD		Composition								
				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	Examination in progress.	
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	stainless steel	GETR	974	20	500		
3		0.218	1.25	96.1	85.0	20	1.98	steel	GETR	956	20	500		
4		0.218	1.25	96.2	92.5	20	1.98		GETR	969	20	500		
E1C-1	Relative smeared conductivity. Two compartments in each capsule. Effect of stoichiometry.	0.218	0.8	95.7	93.8	0	2.00	Type 347 stainless steel	MTR	1170	39	< 100	Report issued.	
2		0.218	0.8	97.4	96.1	20	2.00		MTR	1170	39	< 100		
E1D-1	Relative smeared conductivity. Two compartments in each capsule. Effect of stoichiometry.	0.218	0.8	96.8	94.5	20	2.00	Type 304 stainless steel	MTR	1355	46	< 100	Report issued.	
2		0.218	0.6	96.9	94.8	20	1.98		MTR	1355	46	< 100		
E1E	Relative smeared conductivity at high burnup irradiation to high burnup and rabbit test E1F.	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		0.218	1.9	94.6	94.8	20	2.00		MTR	654/875	20/28	77,000		
E1G-1	Relative smeared conductivity. Two pins different density.	0.217	0.8	93.4	90.9	20	1.97	Type 304 stainless steel	MTR	1031	34	< 100	Report issued.	
2		0.218	0.8	87.2	85.4	20	1.97		MTR	1031	34	< 100		
E2C-1	High burnup. Reference fuel irradiated at near reference thermal conductivity.	0.218	5	84.5	82.2	25	2.00	Type 347 stainless steel	GETR	1060	20	100,000	Report in progress.	
2		0.218	5	94.3	91.4	25	2.00		GETR	1060	20	100,000		
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	120,000	Report in progress.	
2		0.150	5	91.6	88.3	25	1.96		GETR	1040	17	120,000		
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 progress.	
F		0.100	3	96.9	96.5	25	2.00		MTR	655	9.4	172,000		
G		0.100	3	97	95.6	25	2.00		MTR	560	7.9	255,000		
H		0.100	3	96.6	94	25	2.00		MTR	785	11.6			
E2J-1	High burnup. Annular versus solid pellet.	0.218	5	90.4	89.5	25	1.98	Type 316 stainless steel	GETR	1100	22	120,000	Examination in progress.	
2		0.218	5	96.0	88.5	25	1.98			1100	22			
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	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	1100	20	100,000	Report in review.	
2		0.218	4.9	84.1	84.1	25	2.00		GETR	1100	20	100,000		
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	1050	19.5	53,000	Irradiation in progress.	
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	1050	19.5	53,000	Irradiation in progress.	
E5E-1	Powder, pellet, and annular fuel comparison at high burnup.	0.215 (Annular)	16.0	90.5	83.3	25	1.96	Type 316 stainless steel	GETR	1050	19.5	53,000	Irradiation in progress.	
E5F	Linear fuel movement versus burnup.	0.214	10.0	92.0	87.1	25	1.97	Type 316 stainless steel	GETR	850	15	17,000	Irradiation in progress.	
E2P	Effect of radial restraint on fuel swelling.	0.219	2.0	92.0	91.0	25	1.98	Type 316 stainless steel	GETR	1160	15	500	Interim examination in progress.	
E2Q	Effect of radial restraint on fuel swelling.	0.229	2.0	92.0	91.0	25	1.98	Type 316 stainless steel	GETR	1160	15	500		
E1J-1	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	1.93	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-2	Effect of restructuring on smeared thermal conductivity	0.217	5.0	92.0	-	25	1.93	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-3	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	1.93	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-4	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	1.97	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-5	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	1.97	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-6	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	2.00	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-7	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	2.00	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-8	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	2.00	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	
E1J-9	Effect of restructuring on smeared thermal conductivity.	0.217	5.0	92.0	-	25	2.00	Type 316 stainless steel	GETR	1100	23	~ 2000	Fabrication in progress.	

TABLE 2-4
SUMMARY OF TASK F FAST FLUX FUEL TESTING PROGRAM IN EBR-II

October 25, 1968

GROUP	PURPOSE & MAJOR PARAMETERS	ROW	NUMBER		POWER kW/ft	TARGET MWd/Te	CALCULATED MWd/Te	IRRADIATION	
			S/A	PINS				FINISH*	STATUS
F-0	Low power swelling rate of high density fuel at various burnup levels.	7	XG02	1	5-6	40k	40. k	8/68	Discharged from EBR-II core.
			XG03	2		60k	43. 2k	8/70	In reactor.
			XG04	2		100k	43. 2k	5/73	In reactor.
F-2	Medium power scoping study. Major parameters include smear density, stoichiometry, cladding fuel interactions and burnup.	4	XG06	12	11-16	50k	51. 1k	4/67	Post exam completed.
			XG05	5		75k	74. k	6/68	Post exam in progress.
				4	11-16	100k	70. k		At EBR-II for incorporation in new S/A.
F-3A	Bare-pin irradiations to evaluate fuel cladding & cladding coolant interactions.	5	XO40	16	15-18	50k	7. 5k	10/69	In reactor.
F-3B (E1H)	Density and stoichiometry effects for encapsulated fuel pins.	4	XO27	18	13-16	40, 80k 120k	26. 8k	4/69, 4/70, 5/71	In reactor.
F-4	Investigate low density fuel behavior at high heat ratings.	2 3	XO11 XO39	9 7	15-18 15-18	50k 50k	35. k 41. 6k	 2/69	Post exam in progress. In reactor.
F-5	Fuel performance at FCR peak cladding temperatures.	4	1	16	15-16	100k	-----	3/71	Fuel fab. completed.
F-6	Fuel-cladding interactions under accelerated swelling.	7	XO10	4	7-9	50k	43. 7k	6/69	In reactor.
F-7	Pre-irradiated axially restrained fuel prior to transient test. Main variable is fuel density.	4	XO15	2	12-15	50k	44. 5k	3. 69	In reactor.
			1	4	12-15	50k	-----	12/69	Fuel at EBR-II site.
F-8A	Effect of density on swelling rate.	6	XO19	7	6-9	50k	26. 0k	5/70	In reactor.
			XO20	9		100k	26. 0k	12/72	
F-8B	Effect of void deployment method on fuel cladding interaction.	7	XO36	19	6-9	50k	5. 7	1/71	In reactor.
F-9A	Irradiation of unencapsulated pins to high burnup.	4	1	37	11-16	100k	-----	3/71	Fuel fab. completed.
F-9B	Irradiation of unencapsulated pins to high burnup.	4	1	37	10-13	100k	-----	3/71	Fuel fab. in process.

*Based on EBR-II operation at 45 MW as of 6/1/68 and at 50% load factor.

TABLE 2-5

SUMMARY OF TASK L MATERIALS IRRADIATIONS IN EBR-II

Series	Row-S/A	Capsules	Capsules Material	Number of Specimens		Target, 10 ²³ nvt	Total Fluence September 30, 1968		Projected Completion Date(2)
				Biaxial	Tensile		10 ²³ nvt ⁽³⁾	10 ²³ nvt ⁽⁴⁾	
<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 of Type 321 and 347 stainless steel; two of Type 304 stainless steel; three each of Type 316 stainless steel, and Incoloy 800.	70	60 ⁽¹⁾	0.20	0.28	0.22	October 1968
<u>Irradiation in Progress</u>									
L-2	4—XG05	5	One each of Types 304, 316, 321, and 347 stainless steels, and Incoloy 800.	15	80	0.76	0.46	0.34	October 1970
L-6	2—XO18	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 ⁽¹⁾	1.00	0.38	0.23	September 1971
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 steels.	133	114 ⁽¹⁾	0.20	0.22	0.17	December 1968
L-15	3—XO39	4	All five alloys.	None	288	0.20	0.03	0.2	May 1969
MT-3 ⁽⁵⁾ and 4		2	Thirty-two Incoloy-800, 32 Inconel-625, and 32 Hastelloy-X tens. samples.	None	96	1.00	0.48	0.36	December 1971

1. Includes specimens of experimental alloy of base compositions Fe, (9-14) Ni, (9-14) Cr, (0.0004 - 0.05) C.
2. Assuming EBR-II continues to operate at 45 MW with 50% load factor.
3. Fluence based on EBR-II core physics conversion factors.
4. Fluence based on flux wire analysis conversion factors.
5. Capsules from previous AEC program under contract AT(04-3)-189, P.A. 29.

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 fuels, 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 in future work on Task B.

3.2 DEFECTED FUEL TESTING (Series B8 and B9)

3.2.1 Forced Convection Capsule (Series B9)

Thermal analysis of the three pin forced convection capsule was completed.

A study is presently underway to ensure adequate de-entrainment of gas from the coolant sodium at the time of a fuel-pin failure. AEC approval was received for fabrication of the three pin forced convection capsule B9A. Drawings were released and fabrication of capsule parts initiated.

Testing of the reference defecting mechanism for capsule B9A continued with further out-of-pile tests and fabrication of an in-pile proof test. The out-of-pile tests are extensions of the "ballized" clad tests reported in the 26th Quarterly.⁽¹⁾ In the first series of tests, heated samples of notched cold-worked clad (15-mil wall thickness - 12-mil notch) were allowed to cool over UO₂ pellets. In the second series of tests, right cylindrical solid steel mandrels were slipped inside of notched cold-worked cladding specimens. The mandrels were then compressed axially and the resulting "barreling" caused clad strain and rupture (ring tests). These latter tests were conducted at both room temperature and 1100°F, both with and without a 100-hour, 1000°F anneal. In both series of tests, failure was observed after an o.d. change of less than 0.001 inch, confirming the earlier work and supporting the differential fuel/cladding thermal expansion defect mechanism planned for B9A.

Fabrication of the in-pile proof test capsule for the notched cladding, differential thermal expansion defect mechanism, was completed during this report period. Fuel specimen fabrication showed that notching the cold-worked cladding caused 0.001 to 0.002 inch ovality in the stainless

steel tubing. Tests with precision ground UO_2 pellets showed, however, that the required 0.0005 ± 0.0005 diametral fuel/clad gap could still be attained by force fitting the fuel pellets into the cladding in the notched region. The capsule is now waiting irradiation. Examination of the test capsule will include neutron radiography, gamma scanning, and destructive examination.

Fabrication of the sodium fill and clean-up system continued throughout the report period.

Development of techniques for post-irradiation oxygen analysis of the FCC sodium coolant has been initiated. Of the four methods of analysis considered (oxygen meter, gettering with uranium, amalgamation, and distillation), the distillation technique requires the least amount of development and the greatest chance for success in this application. Calculations indicate that an increase in oxygen concentration of up to 160 ppm could result from a 2.5% reduction in fuel O/M (assuming no oxygen gettering by other capsule components or the cold trap). Such concentrations are within the sensitivity of the distillation technique.

3.3 "SODA-POP" TESTS (Sodium Water Reactions)

Installation of instrumentation and equipment was completed. The explosive charge which will expose the sodium to the water was tested and modified to reduce the energy transmitted to the water. The first instrument calibration test using 100 grams of NaK was completed. Substantial pressure variations were noted on the pressure transducer readouts. From this data it was established that the frequency response of the pressure transducer was too low, and higher frequency transducers were required. On this basis the 100 gram NaK test was re-

run successfully, using sensors with a higher frequency response. The measured final system pressure was considerably lower than the calculated final system pressure, assuming stoichiometric reaction of the NaK and water to form hydrogen sodium hydroxide, and potassium hydroxide. The lower pressure appears to be caused by absorption of hydrogen gas into the water. All of the hydrogen gas generated from the 100 gm NaK test could theoretically be dissolved in the water. With the instrument checkout and calibration completed, the "Soda-Pop" test sequence is continuing.

3.4 CAPSULES B8C AND B8D (Fuel, Sodium, Cladding Compatibility Studies)

Capsule B8C was removed from the GETR for interim neutron radiography. The burnups on B8C-1b and B8C-2 sodium-bonded fuel pins at time of removal were 62,000 MWd/Te and 31,000 MWd/Te respectively. Each pin is operating at a linear power of 21 kW/ft. Both pins continue to show fuel expansion into the plenum region of the pins. Comparison of the presently observed fuel expansion with that of the previous interim examination indicates an approximately linear increase with burnup. A $\Delta V/V$ of approximately 1% per 10,000 MWd/Te is measured for the top fuel pin (B8C-1b). The bottom pin (B8C-2) shows a slightly smaller fuel expansion rate of about 0.7%/10,000 MWd/Te. Figure 3-1 shows the appearance of the fuel expansion of the top pin of capsule B8C at zero, 30,200 and 62,000 MWd/Te burnup. Figure 3-2 compares the presently observed fuel expansion rate to a gas bonded pin run at a comparable power to 125,000 MWd/Te burnup. As can be observed, the sodium-bonded fuel expansion rate of the sodium-bonded fuel is assumed to be caused by the formation of a low density sodium-fuel reaction product.

0 BURN UP. MWd/Te

30,200

60,200

FIGURE 3-1. OBSERVED FUEL SWELLING OF SODIUM BONDED PIN B8C-1B

FIGURE 3-2. MEASURED VOLUME CHANGE OF FUEL OF SODIUM BONDED PIN B8C-1B AND GAS BONDED PIN E2C VERSUS BURNUP

The top pin incorporates a bellows as part of the plenum area to measure fission gas release. The prior interim examination indicated 96% fission gas release for the fuel while operating at 21 kW/ft. The present interim examination indicates that the bellows has developed a leak and is no longer functioning properly.

After the interim neutron radiography, capsule B8C was reinserted into GETR and the current accumulated burnups are 96,000 MWd/Te and 48,000 MWd/Te on the top and bottom pins, respectively.

Capsule B8D was also removed from the GETR for interim neutron radiography. The burnup on the three sodium-bonded pins (peak linear power of 17.5 kW/ft) at time of removal was:

		<u>MWd/Te</u>
Top Pin	B8D-3	10,500
Middle Pin	B8D-2	11,500
Bottom Pin	B8D-1b	10,500

Neutron radiographs were taken of the top two fuel pins of this capsule and show the cladding of these pins to be intact. The center pin (O/M = 1.97) shows a center void 0.020- to 0.025-inch in diameter running most of the length of the fuel. The fuel pellet interfaces can still be distinguished in both pins. The top edge of the top fuel pellet in each pin appears, however, to be forming a fuel/sodium reaction product zone similar to that observed for fuel pins B8C-1b and B8C-2. No significant increase in fuel lengths was observed.

After neutron radiography, capsule B8D was reinserted into GETR and has accumulated the following burnup on the three pins:

		<u>MWd/Te</u>
Top Pin	B8D-3	23,400
Middle Pin	B8D-2	25,300
Bottom Pin	B8D-1b	23,700

3.5 FISSION PRODUCT DISTRIBUTION

3.5.1 Miniloop (Out-of-Pile Radio-Active Fission Product/Sodium Loop)

Fabrication of the loop components is complete except for the fin-tube cooler. Completion of the miniloop and containment system has been delayed because of the late delivery of the containment box for the loop due to a strike in the vendor's factory. The containment box was received on site in mid-October. Preparation of the containment box for installation of the loop is presently underway.

3.6 CAPSULE B3C [Intermediate Burnup, Typical FCR Length (20 inches), Sealed Fuel, Natural Convection Cooled]

Microprobe analysis was completed on the globule of metallic material found attached to one edge of the cladding in the failed region of the fuel pin and a metallic ingot in the center of the same fuel section. The appearance of the metallic globules analyzed is illustrated in Figures 3-8, 3-10, and 3-11 of the 26th Quarterly.⁽¹⁾ Analysis showed the two metallic globules to be composed of Mo, Tc, Ru, Rh, and Pd. No evidence of the major cladding constituents was found in either particle.

Microprobe analysis was also performed on one of the metallic like "stringers" (see Figure 3-3) found extending from the i.d. surface of the cladding into the fuel in regions of severe attack. Figure 3-4 shows the concentration of Fe, Cr, and Ni in the "stringer." As can be seen, the portion of the stringer closest to the cladding surface contained Fe, Cr, and Ni in similar proportions to the clad. However, in the region of the stringer further removed from the cladding surface the concentration of Cr and Ni is reduced.

250X

FIGURE 3-3. APPEARANCE OF METALLIC "STRINGER" USED FOR MICROPROBE ANALYSIS (B3C)

Fe

Cr

Ni

NOTE: THE STRINGER IS SHOWN REVERSED
LEFT TO RIGHT IN THESE PHOTO-
GRAPHS COMPARED TO THE PHOTO-
GRAPH OF FIGURE 2-3

FIGURE 3-4. RELATIVE CONCENTRATIONS OF Fe, Cr, AND Ni MEASURED IN
METALLIC "STRINGER" SHOWN IN FIGURE 3-3

This portion of the stringer also showed small amounts of fission products.

A scan was also made of the pitted and attacked region of the cladding. In addition

to normal alloy constituents of the cladding, the pitted region contained U, Pu, Ru, and Zr and slightly more Mo than the unpitted clad. No evidence of Cs could be found.

SECTION IV

TASK C—SAFETY EXPERIMENTATION

4.1 GENERAL

The purpose of Task C is to study fuel and cladding behavior under various casualty and accident situations, using both analytical techniques and an experimental program of testing the response of fuel to severe transients. These investigations include a study of the transient performance of axially restrained fuel as a function of density, performance of fuel in the presence of a defect, and comparison of the behavior of powder and pelleted fuel. Additional tests are conducted to determine the potential for molten fuel movement and to observe the effect of the prototypical range of fuel and blanket lengths on movement. Testing includes experiments which involve multiple-pin assemblies to study flow blockage conditions and experiments to evaluate the work-energy relationships and energy partition data needed in the analysis of large, terminal accidents.

During the current contract period, it is expected that work will progress through issuance of a final report on transient irradiation of zero-burnup, axially-restrained fuel as a function of density and fuel form (pellet and powder). Experiments will be conducted on pre-irradiated long pins to study relocation of molten fuel during tran-

sients and tests of defected or sodium-bonded fuel pins will be initiated. Out-of-pile scoping studies on flow blockage experiments will be performed, and conceptual design of work-energy experiments will continue.

4.2 AXIALLY RESTRAINED FUEL EXPERIMENTS

4.2.1 Post-Irradiation Examination, Zero-Burnup Fuel Specimens

Metallographic examination is complete for all zero-burnup fuel specimens in Series IV (C4A to D - pelleted fuel, and C4I and J - powder fuel).

Preliminary metallographic analysis of specimen C4I indicates that approximately 20% of the fuel volume (approximately 100 mil diameter central core of the fuel column) underwent some sintering and porosity movement, accompanied by slight equiaxed grain growth. It is believed that this section of the fuel reached its melting point, but did not pass completely through the heat of fusion. The peak instantaneous power experienced by this specimen in the transient was equivalent to 71 kW/ft. This result is similar (but on a smaller scale) to the previously tested specimen C4D. ^(2, 3)

Powder specimen C4J experienced a peak transient power equivalent to 132 kW/ft. Metallographic examination of this specimen (which failed during transient irradiation), (2, 4) indicated the following:

1. The fuel cladding underwent extensive failure, with fuel expulsion extending over approximately 63% of the fuel column length, centered about 1-1/2 inches below the midplane. Figure 4-1 of Reference 1 presents a full length neutron radiograph (90-degree view) of this specimen, showing the extent of the failure.
2. Approximately 65% of the molten fuel volume (approximately 31.5% of the total local fuel volume) was ejected upwards from the portion of the pin below the failure zone to the primary failure zone. Because of the low burnup, no fission product gases were present to cause this upward movement.
3. The primary mode of failure is difficult to ascertain, because of the large extent of clad melting and fuel relocation resulting from the failure. The extent of this post-failure melting and relocation can be seen in Figures 4-2 and 4-3.

Figure 4-1 presents the β - γ autoradiograph and a fine-ground macrograph of a longitudinal section at the bottom end of the specimen. The large central void is evidence of this fuel expulsion.* The molten-fuel boundary is delineated in the β - γ autoradiograph as the boundary between the visible fuel particles in the outer region of the fuel and the single-phase region adjacent to the central void. In addition to being expelled upward, the molten fuel core also penetrated the powder UO_2 insulator at the base of the fuel column, to a depth of approximately 0.15 inch.

*Fuel expulsion is defined as rapid movement of molten fuel to a new location (usually external to the cladding) under driving forces other than gravity.

Figure 4-2 presents a transverse section of the fuel pin in its heat sink near the lower end of the expelled fuel region. About two-thirds of the cladding circumference had at one time exceeded its melting point, but the fuel (Vipak powder fuel with no prior in situ sintering history) retained almost complete coherency. The main loss of fuel is again caused by axial expulsion from the central portion of this section, leaving behind a large central void similar to that noted in Figure 4-1.

The molten fuel boundary is again visible in the β - γ autoradiograph as well as the etched fuel montage. In the latter case, the melt structure of the fuel is also visible.

Figure 4-3 presents a transverse section located near the midplane of the expelled fuel region. As can be seen, only small remnants of the cladding remain at this elevation, indicating significant axial movement of the molten cladding. The fuel in this section has expanded radially about the central void, almost to the i. d. of the aluminum heat sink. There still appears to be some circumferential coherency to the fuel, particularly at its inner boundary. This final geometry of the failure would have resulted in at least local displacement of the sodium coolant (if such displacement or voiding had not already occurred).

4. Apparent gas or vapor blanketing caused axial extension of the cladding failure region upwards for approximately 1-1/2 inches above the main failure zone (defined as the region of major fuel relocation outside of the cladding). This may be an example of a form of burnout phenomenon, where clad melting is due to the insulating qualities of a gas or vapor blanket.

Figure 4-4a illustrates the external appearance of the fuel pin in this extended failure region (the portion of the fuel pin below this region has already been removed as part of the examination). There is no evidence of fuel outside of the cladding in this region.

Figures 4-4b and 4-4c show a transverse section through the region of the molten clad seen in Figure 4-4a. Again, as in Figure 4-2, one of the striking features of this view (shown in greater detail in Figure 4-5a) is the apparent coherency of the powdered fuel. The fuel actually bridges the 100-130 mil wide cladding breach with a span which has a minimum thickness of only 12-14 mils. There appears to be some washout from the outer portion of the fuel (in the region that did not sinter during the transient irradiation) on both sides of the cladding breach. This washout is probably a result of the sodium-removal operation associated with the post-irradiation destructive examination.

The molten boundaries of the cladding, at the edges of the breach, are shown in Figure 4-5b. The angle of the melt boundaries, indicating more melting at the inner surface, supports the contention of a burn-through from the inside,

as a result of insulation of the outer surface (e.g., sodium voiding or gas blanketing).

The effective cladding failure actually extended approximately 1/2 inch above the highest point of cladding melt-through (as seen in Figures 4-4a and 4-6a), in the form of a blister in the cladding. This blister (shown in detail in Figures 4-6 and 4-7) is formed from cladding which had become molten during the apparent insulation of the cladding (Figure 4-7).

Here again, the molten boundaries in the cladding sharply converge from the inner surface to the outer surface of the cladding. The shape of the molten region indicates that the melting began at the inside surface, and was the result of a smaller void or bubble than in the region of the cladding melt-through.

5. Temporary gas-blanketing was probably responsible for initiating clad melting along the lower end of the fuel column, below the main failure zone. Near the bottom of the fuel zone (Figure 4-1), cladding damage was not readily apparent; but when etched, it appeared that the inner cladding wall had become molten during the transient. This section of cladding is presented in the etched conditions in Figure 4-8. This partial cladding melt indicates that the gas or vapor blanketing probably extended both upward and downward from the main failure zone.
6. The region above the failure zone (see Figure 4-1 of Reference 1) seems comparatively undisturbed, except for the large central void left when approximately 40% of the local fuel volume moved to the failure region. Figure 4-9 presents a transverse section from this unfailed

FIGURE 4-1a. β - γ AUTORADIOGRAPH (POLISHED CONDITION)

FIGURE 4-1b. FINE-GROUND MACROGRAPH

FIGURE 4-1. TASK C SERIES IV POWDER FUEL SPECIMEN C4J: LONGITUDINAL SECTION SHOWING BOTTOM END OF FUEL COLUMN

FIGURE 4-2b. β - γ AUTORADIOGRAPH

FIGURE 4-2a. FINE-GROUND MACROGRAPH

FIGURE 4-2c. FUEL-ETCHED MONTAGE

FIGURE 4-2. TASK C POWDER FUEL SPECIMEN C4J: TRANSVERSE SECTION OF FUEL PIN AND HEAT SINK NEAR THE LOWER END OF THE MAIN FAILURE ZONE.

FIGURE 4-3. TASK C SERIES IV POWDER FUEL SPECIMEN C4J: FINE GROUND MACROGRAPH OF TRANSVERSE SECTION LOCATED NEAR MID-POINT OF EXPELLED FUEL REGION.

FIGURE 4-4a. EXTERIOR APPEARANCE OF CLADDING AND FUEL IN REGION OF CLADDING MELT-THROUGH

FIGURE 4-4b. TRANSVERSE SECTION IN REGION OF CLADDING MELT-THROUGH, β - γ AUTORADIOGRAPH (POLISHED CONDITION)

FIGURE 4-4c. TRANSVERSE SECTION IN REGION OF CLADDING MELT-THROUGH, MONTAGE-POLISHED CONDITION

FIGURE 4-4. TASK C POWDER FUEL SPECIMEN C4J: REGION OF CLADDING MELT-THROUGH, LOCATED IN THE REGION OF CLADDING FAILURE ABOVE THE MAIN FAILURE ZONE.

**FIGURE 4-5b. DETAILS OF MOLTEN PORTION
OF THE CLADDING (ETCHED-
CLADDING CONDITION)**

CLADDING BREACH

APPROXIMATE LOCATION OF
CLADDING MELT BOUNDARIES

MOLTEN FUEL STRUCTURE

**FIGURE 4-5a. DETAIL OF FUEL BRIDGE
(ETCHED-FUEL CONDITION)**

**FIGURE 4-5. TASK C POWDER FUEL SPECIMEN C4J: DETAILS OF THE
CLADDING MELT THROUGH REGION, LOCATED ABOVE THE
MAIN FAILURE ZONE**

FIGURE 4-6a. EXTERIOR APPEARANCE OF CLADDING IN REGION OF CLADDING BLISTER AND MELT-THROUGH

FIGURE 4-6b. TRANSVERSE SECTION IN REGION OF CLADDING BLISTER β - γ AUTORADIOGRAPH (POLISHED CONDITION)

FIGURE 4-6c. TRANSVERSE SECTION IN REGION OF CLADDING BLISTER MONTAGE-POLISHED CONDITION

FIGURE 4-6. TASK C POWDER FUEL SPECIMEN C4J: REGION OF CLADDING BLISTER, LOCATED IN THE REGION OF CLADDING FAILURE ABOVE THE MAIN FAILURE ZONE

FIGURE 4—7. TASK C POWDER FUEL SPECIMEN C4J: DETAIL OF CLADDING BLISTER,
LOCATED IN THE REGION OF CLADDING FAILURE ABOVE THE MAIN FAILURE ZONE.

FIGURE 4-8. TASK C POWDER FUEL SPECIMEN C4J: DETAILS SHOWING THE PARTIALLY-MOLTEN CLADDING FROM THE LONGITUDINAL SECTION LOCATED AT THE BOTTOM END OF THE FUEL COLUMN (SEE FIGURE 4-1).

FIGURE 4-9a. β - γ AUTORADIOGRAPH

FIGURE 4-9b. MONTAGE-POLISHED CONDITION

FIGURE 4-9. TASK C POWDER FUEL SPECIMEN C4J: TRANSVERSE SECTION IN UNFAILED PORTION OF FUEL PIN, LOCATED IN REGION ABOVE MAIN FAILURE ZONE.

region. Again the molten-fuel boundary can be identified readily, especially in the β - γ autoradiograph.

7. The central void left by the molten fuel movement extended virtually to the top of the fuel column (see Figures 4-10, 4-11, and 4-12). Again, about 40% of the fuel volume is missing in this longitudinal section, with the upper limit of the void within 10-20 mils of the fuel-blanket interface.

The interesting features visible in Figures 4-10, 4-11, and 4-12 follow:

- a) The apparent remnant of a molten fuel bubble
 - b) A predominance of fine fuel particles in the outer region of the fuel (as compared to the coarse fuel particles in the same region at the base of the fuel column - Figure 4-1)
 - c) Intermixing of fuel particles with the coarse UO_2 particles of the blanket
 - d) Fusion of the fuel and UO_2 particles at the fuel-blanket boundary, resulting from fuel particles becoming molten in the region of the first layer of UO_2 particles in the blanket. The axial gradation in UO_2 grain size is a result of the axial temperature gradient at this fuel-blanket interface.
8. The local fuel-melt volume varies from approximately 49% to 60% (a wider variation than previously seen in pelleted fuel specimens).^(1, 3, 5) This may be due to local variations in the distribution of fuel particle size, such as that noted above in the comparison of the top and bottom ends of the fuel column. This

variation in fuel particle size distribution could lead to axial differences in the local fuel-clad heat flux, producing variations in the local fuel-melt volumes. Alternatively, these variations could be a post-failure result of fuel, or coolant movement, or both.

4.2.2 EBR-II Irradiations

EBR-II burnup has continued for specimens C4G and H (EBR-II capsules F7C and D). The accumulated burnup for these two specimens stands at approximately 44,500 MWd/Te (out of a target 50,000 MWd/Te burnup) as of October 22, 1968. The four remaining specimens designated for EBR-II irradiation (specimens C4E, F, K, and L) are still awaiting assignment of core position.

Design effort has been initiated for a TREAT capsule to be used in the transient irradiation of the pre-irradiated fuel specimens once they have completed their EBR-II irradiations.

4.3 AXIAL FUEL RELOCATION EXPERIMENTS

TREAT transient irradiations of capsules C5A and C5B were completed and the capsules returned to the Vallecitos Nuclear Center for destructive examination. TREAT test parameters are shown below:

	C5A	C5B
Period (sec)	0.131	0.136
Reactivity (%)	1.37	1.35
Integrated Power (MWsec)	309	310
Peak Power (MW)	520	505

It can be seen from the above that the two specimens were subjected to nearly identical conditions, and it is expected that the volume of molten fuel was the same in both cases. Peak specimen power in the tran-

FIGURE 4-10. TASK C POWDER FUEL SPECIMEN C4J: LONGITUDINAL SECTION SHOWING THE TOP END OF THE FUEL COLUMN.

FIGURE 4-11a. POLISHED MONTAGE

FIGURE 4-11. TASK C POWDER FUEL SPECIMEN C4J: POLISHED AND FUEL-ETCHED MONTAGES OF LONGITUDINAL SECTION LOCATED AT THE TOP END OF THE FUEL COLUMN.

FIGURE 4-11b. FUEL-ETCHED MONTAGE

sients was equivalent to 160 kW/ft (C5A) and 155 kW/ft (C5B).

Neutron radiographs of C5B (Figure 4-13) indicate satisfactory performance with no evidence of cladding failure. There was upward movement (≈ 10 inches, measured from the fuel/blanket interface) of molten fuel into the annular blanket. A central void (≈ 50 to 100 mils) has been formed, continuous with several fuel "plugs" in the upper portion of the fuel column and intermittent in the lower portion, indicating possible fuel movement in both directions.

A check of C5B fuel column lengths (pre- and post-transient) reveals apparent "growth" (≈ 0.20 inch) of the uppermost pellet in the fuel column. A situation analogous to that observed in capsule C4A may have occurred; i. e., molten fuel pushing up against the blanket and filling the resulting fuel/blanket separation.⁽¹⁾ Radiographic detail is not adequate to precisely determine the volume of molten fuel or detect clad distortion (if any).

Radiographs of C5A (Figure 4-14) reveal fuel pin failure with extensive fuel movement into the coolant annulus, primarily in the upward direction. This capsule was identical in all respects to C5B, except that the blanket was solid rather than annular pellets (the hole through the C5B blanket was 0.070-inch diameter).

4.4 DEFECTED OR SODIUM-BONDED FUEL EXPERIMENTS

The high pressure sodium-fill system has been completed and is undergoing functional checkouts prior to installation. A mockup powder fuel pin (natural UO_2) simulating the C6 powder pins will then be sodium filled as a final system checkout.

The seal test rig to proof-test the lower end-closure design of the TREAT pressure

vessel has been received. Tests were conducted with soft carbon steel and aluminum seal rings. No leakage was observed with any of the seals when pressurized with water to 15,000 psi.

Although final TREAT capsule assembly drawings have been awaiting final results on the test of the end closure seal, procurement or fabrication, or both, of inner capsule assembly components has been initiated. The fuel pins are being fabricated and will be ready for sodium filling by late October.

4.5 DESIGN BASIS ACCIDENT STUDIES

Feasibility studies continued in an effort to find facilities suitable for conducting useful work-energy experiments. Discussions were held with personnel at the Lawrence Radiation Laboratory (Livermore) concerning use of SUPER KUKLA. This facility appears to provide ample space for experiments (compared to SPERT), but the available power density may be limiting.

4.6 FLOW BLOCKAGE EXPERIMENTS

Nondestructive tests on the fuel simulators (electrical resistance heaters) have revealed potential hot spots in five of the six "primary design" heaters. These flaws were indicated by the presence of temperature discontinuities on the heater surface at low power and are presumed to indicate partial breakdown in the electrical resistance of the insulating material. Although successful operation of these heaters is not definitely precluded, the probability of success is probably reduced. As a result, it has been decided to use the backup design in the first out-of-pile tests.

The mobile data acquisition system has been received and checkout has been completed. The system was used to record

FIGURE 4-12. TASK C POWDER FUEL SPECIMEN C4J: DETAIL OF FUEL-BLANKET INTERFACE AT TOP END OF FUEL COLUMN.

FIGURE 4-13. NEUTRON RADIOGRAPH OF CAPSULE C5B FOLLOWING TREAT TEST TRANSIENT, TT 1233

FIGURE 4-14. NEUTRON RADIOGRAPH OF CAPSULE C5A FOLLOWING TREAT TEST TRANSIENT, TT1239

transient instrumentation in the "Soda Pop" tests (see Task B for information on "Soda Pop" tests). The transient response of the data system exceeds the capabilities of the "Soda-Pop" transducers for pressure pulses with rise times of less than 0.001 sec.

Mockup 1A drawings have been completed. The design of Mockup II is continuing, and the test specification for Mockup II operation is being generated. This specification will reflect, as nearly as possible, the requirements of the first in-pile flow coast down test, C10A.

SECTION V

TASK E - FUEL PERFORMANCE EVALUATION

5.1 GENERAL

The purpose of Task E is to develop technology required to specify the fuel materials, fuel performance, and quantitative design limits for fast reactor oxide fuels. A typical target performance is to operate at linear powers up to 18.5 kW/ft (600 W/cm) and to peak burnups greater than 130,000 MWd/Te (35×10^{20} fission/cc). The program includes determination of physical and mechanical properties of fuel using both in- and out-of-pile testing. Emphasis is on determining the properties of the fuel, and its in-pile behavior at power levels typical of near-term FCR cores. This task has emphasized careful characterization and control of test parameters which have been possible so far primarily in thermal test reactors.

5.2 CENTER TEMPERATURE LIMITS

5.2.1 Capsule EIAA (Direct Measurement of In-Pile Thermal Conductivity)

Capsule EIAA was removed from the GETR following failure of the tungsten-rhenium center fuel thermocouple. During the irradiation, fuel thermal conductivity data

were obtained over the temperature range of 1200 to 2100°C. Peak burnup on the fuel was calculated to be approximately 18,000 MWd/Te. Analysis of the thermocouple data is in progress. Shown in Figure 5-1 is a plot of fuel center temperature versus linear power. A linear relation was obtained until a temperature of 2000°C was reached, corresponding to a linear power of 15 kW/ft. During steady-state operation at 15 kW/ft, a slight increase in fuel center temperature was observed followed by a corresponding decrease. The temperature increased from 2000°C to approximately 2100°C over a period of about 3-1/2 days. A plot of fuel center temperature versus time during this period is shown in Figure 5-2. The reason for this increase is not understood but it is hypothesized that it may result from a change in thermal conductivity of the fuel because of changes in oxygen-to-metal ratio across the fuel radius or a decalibration of the in-fuel thermocouple. Out-of-pile tests reported previously ⁽¹⁾ showed that a significant oxygen-to-metal ratio gradient can be produced in mixed oxide fuel subjected to a temperature gradient. The decrease in the center temperature is hypothesized to result from fuel restructuring.

FIGURE 5-1. FUEL CENTER TEMPERATURE AS A FUNCTION OF LINEAR POWER - CAPSULE E1AA

FIGURE 5-2. CHANGE IN FUEL CENTER TEMPERATURE OF CAPSULE EIAA WITH TIME AT CONSTANT POWER

A computer program was developed to calculate capsule power based on the temperatures indicated by the thermal barrier thermocouples. The program accounts for gamma heating effects and calculates cladding and thermal barrier temperatures. Preliminary calculations indicate that circumferential temperature variations of 50° to 80° F existed around the fuel pin cladding during operation at constant power.

5.2.2 Capsule Series E1J (Effect of Restructuring on Smeared Thermal Conductivity)

Fabrication of the E1J series of capsules was delayed because of modifications required on the thermal dams as a result of the analysis of the E2P and E2Q capsules.* Fabrication of all capsule bodies and six fuel pins is complete.

5.3 HIGH BURNUP IRRADIATIONS

5.3.1 Capsule E2J (High Burnup Comparison of Pellet and Annular Fuel)

Post-irradiation examination of the solid pellet and annular pellet fuel pins from capsule E2J was completed. Examination of the fuel indicated that after a burnup in excess of 100,000 MWd/Te in a thermal flux, no significant difference is discernible in the performance of annular and solid pellet fuel (smeared density of both pins was 89%).

Fission gas analyses indicate gas releases in excess of 80% in both the solid annular fuel. This result is in agreement with previous high burnup, high power, thermal flux irradiations.

* See Section 5.3.2

5.3.2 Capsules E2P and E2Q (Variable Radial Restraint)

Fabrication of the replacement fuel pin for capsule E2P was completed. Calculations of thermal convection occurring in the E2P and E2Q capsules showed that a thermal convection loop could be set up within the capsule which would cause errors in the capsule power calculated from the thermocouples. The capsule thermal dams were redesigned to eliminate the convection currents and new thermal dams fabricated. Re-encapsulation of the fuel pins is in progress.

5.4 PHYSICAL PROPERTY STUDIES

5.4.1 Mechanical Properties of Ceramic Fuels

The plasticity of UO₂ samples of various grain sizes and densities was measured in a comparison creep apparatus. All tests were made using stoichiometric UO₂ with the following grain sizes and densities:

<u>Percent Theoretical Density</u>	<u>Grain Size (Microns)</u>
93	10-12
95	1-3
95	12-18
95	19-28
97	~3
97	7-9
98	~18

The data obtained indicate that three distinct regions of creep response occur under constant load. The creep rate in each region is described by one of the following equations:

1. Viscous Creep Region ($\sigma \leq 2500$ psi)

$$\dot{\epsilon} = A' \sigma \exp \left(- \frac{90,000}{RT} \right)$$

2. Power Law Stress-Strain Region
($\sigma \geq 6000$ psi)

$$\dot{\epsilon} = A'' \sigma^{4.5} \exp \left(- \frac{109,000}{RT} \right)$$

3. Transition Region ($2500 \leq \sigma \leq 6000$ psi)

$$\dot{\epsilon} \propto A'' \sigma^n \exp \left(- \frac{Q}{RT} \right)$$

where $1 < n < 4.5$

A plot which is typical of all the data is shown in Figure 5-3. The continuous change of the stress exponent "n" in the transition region indicates that a gradual change in the dominant deformation mechanism occurs between the viscous creep region and the power law stress-strain region.

In the viscous stress region, the creep rate was found to vary inversely with the square of the grain size.

$$\dot{\epsilon} \propto \left(\frac{1}{G.S.} \right)^2$$

This relationship is shown in Figure 5-4, which is a plot of data obtained for a stress of 2000 psi at two temperatures, 1650°C and 1750°C. This type of behavior would be expected where a Nabbero-Herring diffusion mechanism is the rate controlling process in the creep deformation. Under conditions of high stress (> 5000 psi) creep rate was found to be independent of grain size.

Differences in the creep rate of 94.8 and 97% dense samples were found not to be appreciable. However, 92.4% dense material deformed at about twice the rate determined for 97% dense material. This response may be attributed to the large increase of grain boundary porosity (associated with the density decrease) and the subsequent contribution of grain boundary sliding to the total deformation.

Initiation of work on measurements of plasticity of mixed plutonium-uranium oxides is planned during the next quarter.

5.4.2 Thermal Conductivity of Oxide Fuels (Out-of-Pile)

Uranium oxide specimens with densities between 92 and 97% of theoretical were fabricated for checkout of the conductivity furnace, as well as to provide additional information on thermal conductivity of UO_2 as a function of density and thermal gradient. Initiation of tests to check out the conductivity equipment was delayed due to delayed shipment of a transformer for the power supply.

5.4.3 High Temperature Phase Studies

Lattice parameter measurements were made on UO_2 and $(Pu_{0.25}U_{0.75})O_{2-x}$ with oxygen-to-metal ratios of 1.999 and 1.923. Preliminary analysis of data obtained by diffractometry of the mixed oxides indicate the material to be multi-phase.

Lattice parameter measurements of UO_2 were made in order to test the experimental apparatus and procedures. The results obtained are shown in Figure 5-5 compared with results obtained in earlier work.

The experimentally determined lattice parameter as a function of temperature for the 1.999 O/M mixed oxide is shown in Figures 5-5 and 5-6. Figure 5-5 indicates the thermal expansion of mixed oxide to be slightly greater than that of UO_2 in the temperature range from 800°C to 1650°C. Figure 5-6 shows the absolute expansion of the mixed oxide. The results were obtained from four individual temperature cycles, three from data obtained while heating the specimens and one while cooling the specimen. The excellent agreement between individual runs indicates the good degree of reproducibility in the experimental data.

Both the 1.999 and 1.923 O/M samples were found to be multi-phase. Preliminary analysis of diffraction data on the 1.923 O/M samples (Table 5-1) indicate the material is

FIGURE 5-3. STRESS VERSUS STEADY-STATE CREEP RATE FOR UO_2

FIGURE 5-4. EFFECT OF GRAIN SIZE ON STEADY-STATE CREEP RATE OF UO_2

FIGURE 5-5. THERMAL EXPANSION OF UO_2 AND $(\text{Pu}_{0.25}\text{U}_{0.75})_{0.999}$ TEMPERATURE ($^{\circ}\text{F}$)

FIGURE 5-6. LATTICE PARAMETER VERSUS TEMPERATURE FOR 25 wt. % $\text{PuO}_2\text{-UO}_2$ IN A He GAS ATMOSPHERE (O/M ratio is equal to 1.999)

TABLE 5-1

TENTATIVE PHASE IDENTIFICATION FOR (Pu, U)O_{1.9226} MIXED OXIDE SPECIMEN

Observed Data			(Pu, U)O _{2-x}			(Pu, U)O _{1.6}			(Pu, U)O _{1.5}		
$\sin^2 \theta$	d Å	Relative Intensity	d Å	I/I ₁ (1)	hkl	d Å	I/I ₁ (2)	hkl	d Å	I/I ₁ (3)	hkl
0.04878	3.490	VVW				3.47575	17.8	310			
0.05984	3.151	VS	3.14766	100 ⁺	111						
0.07972	2.730	VS	2.72595	100 ⁺	200						
0.09148	2.549	VVW				2.59067	16.2	411, 330			
0.09807	2.462	VVW									
0.10664	2.361	VW							2.36059	52.7	332
0.11280	2.295	W									
0.11838	2.2405	VS				2.24358	100 ⁺	422			
0.12379	2.191	M									
0.13007	2.1375	M									
0.13659	2.086	M									
0.14307	2.038	W							2.02149	100	521
0.15029	1.989	VW									
0.16002	1.889	VS				1.88499	100 ⁺	530, 433			
0.18149	1.810	VVW									
0.18820	1.777	VVW				1.78302	20.2	611, 532			
0.19424	1.749	VVW							1.75066	16.4	620
0.20660	1.696	VVW				1.69599	18.8	541			
0.21968	1.645	VS	1.64381	100 ⁺	311						
0.23935	1.576	VS	1.57383	100 ⁺	222						
0.24420	1.560	VW							1.56584	55.8	710
0.30906	1.387	VVW									
0.31919	1.365	S	1.36298	100 ⁻	400						
0.32555	1.351	VW				1.36293	64.7	554			
0.33129	1.339	VW							1.34269	42.3	644
0.34383	1.315	VW				1.3371	35.6	653			
0.34873	1.305	VW							1.30486	34.6	660
0.36386	1.278	W				1.27771	87.4	750			
0.37946	1.250	VS	1.25075	100 ⁺	331						
0.39946	1.220	VS	1.21908	100 ⁺	420						
0.43993	1.162	VVW				1.15858	26.2	930			
0.47950	1.113	VS	1.11286	100 ⁺	422						
0.48997	1.101	VW				1.09913	40.1	1000			
0.50873	1.081	VVW				1.07778	22.7	1020			
0.51501	1.074	VVW									
0.53845	1.0529	VS	1.04922	100 ⁺	511						
0.63790	0.965	VS	0.963769	100 ⁺	440						
0.65102	0.955	VVW							0.956487	28.6	1132
0.69841	0.922	VS	0.921539	100 ⁺	531						
0.71801	0.909	VS	0.90865	100 ⁺	600						
0.79804	0.862	VS	0.862021	100 ⁺	620						
0.83320	0.845	W				0.842992	88.0	1251			
0.83715	0.843	W									
0.84036	0.841	W							0.839377	55.4	1075
0.84418	0.839	VW				0.838077	100.0	1066			
0.85785	0.832	VS	0.831407	100 ⁺	533						

S = Strong M = Medium
W = Weak V = Very

*Denotes lines which have not been identified, of these lines the ones marked VW or VVW may not actually exist and may merely be high background intensities.

1. Intensities were recorded as either 100⁺ or 100⁻ because all but two of the peaks were of sufficient strength to exceed the scale used in recording the data.
2. Two peaks exceeded the scale used, a lesser peak (10, 6, 6) was used for the relative intensity calculations. The two peaks which exceeded the scale were recorded as 100⁺.
3. A value of 9 on the recorder chart was used as the maximum intensity (since that was the largest value which could be recorded above background accurately, for the scale used). All peaks above this were recorded as 100⁺.

composed of three phases. The primary or main phase is $(\text{Pu}_{0.25}\text{U}_{0.75})\text{O}_{1.923}$ with the fluorite structure and a room temperature lattice parameter of $5.4519\text{\AA}$. The two additional phases, which are present in small amounts appear to be $\text{PuO}_{1.6}$ [or $(\text{Pu}, \text{U})\text{O}_{1.6}$] and $\text{PuO}_{1.5}$ [or $(\text{Pu}, \text{U})\text{O}_{1.5}$] with room temperature lattice parameters of $10.9913\text{\AA}$ and $11.0721\text{\AA}$, respectively. Both phases have a cubic structure and are thought to be body-centered cubic. The lattice parameters and structures for the latter two phases agree well with those reported by Sari, et al. ⁽⁶⁾ and Mulford, et al. ⁽⁷⁾

Analysis of the data for the 1.999 and 1.923 O/M material is continuing for temperatures up to 1650°C . Work will be initiated on mixed oxides with O/M of 1.94 and 1.97 in the near future.

5.5 INTEGRAL THERMAL FLUX IRRADIATIONS

5.5.1 Capsules E5C, E5D and E5E (Comparison of Solid Pellet, Annular Pellet and Powder Fuels)

Capsules E5C, D, and E continue to operate successfully at linear powers of 19.5 kW/ft . Calculated burnup for these capsules through October 31, 1968 is $100,000\text{ MWd/Te}$.

5.5.2 Capsule E5F (Linear Fuel Movement)

The purpose of capsule E5F is to measure the axial change in length of mixed oxide in-pile as a function of power and burnup. Changes in the fuel length in-pile are monitored via a transducer (capsule and fuel pin drawings were presented in the Twenty-Fifth Quarterly Report). ⁽²⁾

Figure 5-7 and 5-8 show the power output and corresponding axial change in fuel length as a function of time and temperature for power traverses made at $19,500\text{ MWd/Te}$.

and $33,000\text{ MWd/Te}$. During the power traverse at $19,400\text{ MWd/Te}$, no relative axial expansion was observed while the power was lowered and then raised to about 13 kW/ft . This lack of response is attributed to fuel/clad interaction and agrees well with results reported by Bailly. ⁽⁸⁾ Above 13 kW/ft , the fuel appeared to behave quite erratically for a period of an hour and a half. This behavior may be caused by the release of fission gases as a result of the temperatures encountered at these powers. At lower powers fuel temperatures are too low to allow the release of appreciable amounts of fission gases. At 13 kW/ft , 10% of the fuel volume operates at temperatures high enough to permit release of 100% of the fission gases which were accumulated at lower powers in this volume. This theory is in agreement with results reported by Nelson. ⁽⁹⁾

The power traverse shown in Figure 5-8 shows very good response of the fuel to power changes. The absence of the lack of response as compared to Figure 5-7 is believed to be caused by the differences in the physical conditions of the fuel. The power traverse of Figure 5-7 was performed after the fuel had run for a long period of time at 10 kW/ft and allowed the fuel to sinter and bind against the clad. The power traverse shown in Figure 5-8 was performed immediately after a reactor down-period and the fuel is believed to have been well broken up as a result of thermal shocking caused by the reactor shutdown procedure. Also, it may be noted that no erratic behavior was observed during the power traverse of Figure 5-8. This may be due to the fact that during this traverse the power was never raised high enough to encounter temperatures which would allow the release of appreciable amounts of fission gases.

Irradiation of the capsule is continuing. The transducer instrumentation has been checked out and found to be operating successfully. The capsule has attained a burnup of $46,000\text{ MWd/Te}$ as of October 15.

GEAP-5700

RELATIVE AXIAL EXPANSION CONDITIONS AT 19,400 MWd/Te
5/9/68

FIGURE 5-7. CAPSULE E5F POWER TRAVERSE AT 19,400 MWd/Te BURNUP

FIGURE 5-8. CAPSULE E5F POWER TRAVERSE AT 30,000 MWd/Te BURNUP

SECTION VI

TASK F—INTEGRAL FAST FLUX
IRRADIATIONS

6.1 GENERAL

This task factors the more promising results of the basic fuel technology developments into 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 although this has not as yet been possible in EBR-II.

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 idealized, differential experiments (such as those in Task E and Task L). In addition, the results of prior integral 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. It is planned to increase the propor-

tion of unencapsulated specimens in EBR-II for low risk tests and to continue some encapsulated tests for higher risk configurations.

6.2 MEDIUM POWER, LONG-TERM IRRADIATIONS

6.2.1 Group 2

Four of the nine capsules from Group 2 are being retained at the EBR-II site for additional irradiation to their target burnup of 100,000 MWd/Te. These capsules were originally irradiated in subassembly XG05 to peak burnups ranging from 59,000 to 73,000 MWd/Te.

Metallographic examinations of the F-2H capsule tubing in the peak flux region and the blanket region has been completed (see Task L). The structure was normal considering its time-temperature history, and there was no evidence of deleterious microstructural changes. This examination was a prerequisite for the reinsertion of the four remaining Group-2 capsules.

Post-irradiation examination of capsules F2C, H, R and T continued. The lengths of the Group-2B fuel pins have been measured and are compared to the pre-irradiation lengths in Table 6-1. The changes are relatively consistent and all positive. Residual heat in the clad due to gamma heating would only account for 3 to 4 mils (assuming the clad is 100°F over a 14-inch length). The remaining increase can be related to clad growth. Assuming the growth took place over 14.25 inches of the clad (corresponding to the fuel length), the percent increase can be calculated, Table 6-2.

TABLE 6-1
FUEL PIN LENGTH

Fuel Pin	Pre-Irradiated* (in.)	Post-Irradiated (in.)	ΔL (in.)	Change** (%)
F2C	35.517	35.587	+0.070	0.463
F2H	35.535	35.602	+0.067	0.442
F2O	35.525	---	---	---
F2R	35.481	35.559	+0.078	0.519
F2T	35.558	35.638	+0.080	0.533

* Nominal Length $35 \frac{17}{32} = 35.531$.

** $\% = \frac{\Delta L - 0.004}{14.25} \times 100$.

TABLE 6-2
PERCENT DIAMETRAL VERSUS LENGTH CHANGES

Fuel Pin	Clad Type	Average Diameter Change (%)	Average Length* Change (%)	$\frac{\% \Delta \bar{D}}{\% \Delta \bar{L}}$	$\frac{\% \Delta \bar{D} - \% \Delta \bar{L}}{\% \Delta \bar{D}} \times 100$
F2C	Inc-800	0.51	0.463	1.10	9.2
F2H	316 stainless steel (SS)	0.92	0.442	2.08	52.0
F2R	347 stainless steel (SS)	1.12	0.519	2.16	53.6
F2T	347 stainless steel (SS)	1.10	0.533	2.06	51.5

* Assuming growth occurred over a 14.25 inch length of the clad.

Fuel Pin	Max ΔD (mil)	Max ΔD^*
		$\frac{D}{\%}$
F2C	2.0	0.794
F2H	3.6	1.45
F2R	4.4	1.76
F2T	4.3	1.72

* Percentage based on pre-irradiation micrometer readings.

To compare directly the percent length increase to the percent diametral increase observed, an average diametral increase must be calculated over the 14.25 inches of fuel pin clad. If one assumes the diametral increase followed a cosine distribution along the fuel region of the clad, the integrated average increase can be shown to be 0.637 times the maximum diameter increase observed. The maximum diameter increase varied from 2.0 to 4.4 mil corresponding to an average percent change between 0.57 and 1.27%: i.e., $\frac{2.0 \times 0.637}{250} \times 100 = 0.51\%$. This calculation neglects any temperature dependence on swelling. These values are tabulated in Table 6-2 and compared with the percent length changes measured.

If the growth in the clad were strictly caused by clad swelling, independent of stress, one could assume isotropic conditions: i.e., percent growth in any lineal direction would be the same, and a percent diameter change to percent length change ratio would be one. A ratio of diameter change length change greater than one could be attributed to stress from fuel swelling or fission gas pressure, or both. On this basis, the data in Table 6-2 suggests the 50 - 60% of the diametral increase in F2H, F2R and F2T may be related to fuel swelling. The low value for F2C may be attributed to its relatively flat fuel pin profilometer trace.⁽⁴⁾

6.2.2 Failed Fuel Pin F-2F Examination (58,000 MWd/Te)

Electron probe microanalysis of sample F2F was completed. This specimen of UO_2 -20 without PuO_2 operated to 58,000 MWd/Te. The microstructure of Figure 6-1 showed considerable sodium attack and a sodium-uranate-plutonate phase at the fuel-cladding interface. Results are listed below:

1. The Pu/U+Pu ratio (expressed as percent) showed no significant variation across the fuel radius, the average of 13 readings in 1 radial scan being 11% and the average of a second scan being 19.0%. These ratios had standard deviations of 0.88% and 1.16%, respectively indicating that there were no gross inhomogeneities present.

Similar intensity ratios taken on unfailed sample F2S at a similar burnup showed an equivalent average composition of 19.5% and a standard deviation of 0.66% indicating a slightly narrower range of values. The uniformity of composition and absence of selective leaching indicate composition stability despite the prolonged irradiation in sodium. A radial plot of plutonium concentration for all three scans is shown in Figure 6-2.

2. Figure 6-2 shows that the Pu/U+Pu concentration is virtually the same in the fuel matrix and in the reaction phase. Sodium was found only in the reaction phase and its presence there was heterogeneous.
3. The large solid fission product at the center of Figure 6-1 was analyzed to be 49% Ru, 24% Mo, 15% Tc, 8% Rh and 3% Pd. A second agglomerate consisted of 35% Mo, 34% Ru, 18% Tc, 10% Rh and 3% Pd.
4. No significant transfer of cladding alloying elements to fuel or transfer of fuel or Na to cladding was detected. While Cs has been observed at the fuel-cladding interface of some test pins, none was detected for F2F. This may be due to leaching away of the cesium by the sodium in which it is soluble.

FIGURE 6-1. MICROPROBED SAMPLE OF FAILED F2F

FIGURE 6-2. RELATIVE RADIAL CONCENTRATIONS OF U AND Pu IN FAILED AND UNFAILED OXIDE FUEL AT 58,000 MWd/Te

6.3 LOW POWER FUEL SWELLING IRRADIATIONS

The Group 0 capsule in S/A XG02 was removed from the EBR-II after having achieved its target burnup of 40,000 MWd/Te.

The irradiation of the remaining Group 0 and Groups 6, 8A and 8B capsules was temporarily discontinued because of fission gas activity in the EBR-II cover gas. Their accumulated exposures are present in Table 2-4.

The nineteen Group 8B capsules, incorporated into S/A X036, were inserted into EBR-II and have accumulated 5,700 MWd/Te burnup.

6.4 HIGH-POWER INTERMEDIATE BURNUP IRRADIATION

Reinsertion (into S/A X039) of seven Group 4 capsules previously irradiated to 35,000 MWd/Te in subassembly X011 was accomplished. These are the first irradiated capsules to receive interim non-destructive examination and then be reinserted into the EBR-II for additional exposure. Their cumulative burnup is 41,600 MWd/Te and the target burnup is 50,000 MWd/Te.

6.5 ELEVATED CLADDING TEMPERATURE TESTS

As a consequence of the EBR-II power increase from 45 to 50 MW, peak cladding temperatures of 1121°F on the cladding o.d. can be obtained in a standard Mark A, 19-capsule subassembly. This is adequate to meet the Group-5 target conditions. The design and fabrication of three more capsules was initiated to supplement the 16 capsules originally scheduled for the modified Group-F5 subassembly. (Approval in principle was given for 19 capsules.)

Fuel pin design study was initiated for Group 10. A detailed biaxial stress analysis was completed which resulted in a 12-inch-long gas plenum for the Group 10 pins. This will allow the six to seven-inch mixing plenum required for the cold sodium by-pass flow in a modified 19-capsule subassembly.

6.6 UNENCAPSULATED PIN IRRADIATIONS

6.6.1 Group 3A (High Power)

The 16 unencapsulated Group-3A pins (0.290 in. o.d.) were incorporated into subassembly X040 and inserted into the EBR-II core. This is a milestone as these are the first unencapsulated mixed oxide fuel pins to be tested in EBR-II. These pins have accumulated 7,500 MWd/Te of their target exposure of 50,000 MWd/Te.

6.6.2 Group 9A (High Power) and Group 9B (Medium Power)

The hazards analysis for the Group 9A pins was completed and submitted to ANL for review. Twenty-nine pins of Group 9A have been shipped to the EBR-II site for pre-irradiation inspection. One pin is being returned because of porosity in an end plug weld detected by x-ray inspection. Fabrication of the remaining eight pins was completed but four pins were rejected because of chipped fuel pellets and are being rebuilt.

Fabrication is continuing on the encapsulated fuel pins for Group 9B.

6.6.3 Group 9C (High Power) and Group 9D (Medium Power)

The fabrication of the 37 fuel pins of Group 9C has been completed. Final inspection and preparation of the hazards analysis are in progress.

Construction of the hardware for the 9D Group of fuel pins has begun. The fuel fabrication has been completed and is ready for insertion into the 316 stainless-steel cladding. A procedure has been established

for tagging these fuel elements with a xenon tracer gas as proposed by ANL. A check out of this procedure is being made in an attempt to qualify the technique before the final end-plug welds are made.

SECTION VII

TASK G—CORE NEUTRONICS, THERMAL HYDRAULICS AND SAFETY ANALYSIS

7.1 GENERAL

The purpose of Task G is 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 required 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.

During 1968, methods development and parametric analyses are 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 CONSTANTS

7.2.1 ENDF/B Evaluation and Testing

The ENDRUN code has been used to generate average cross sections for oxygen and Pu-239 in the GAM-1 group structure for comparison with the equivalent results from the MC², GAM-1, MUFT-4, and GAF-GAR codes. This activity is part of the Phase-IIIB data testing of ENDF/B being conducted by the Cross Section Evaluation Working Group and provides a good test of averaging procedures in ENDRUN. Tables 7-1 and 7-2, listing some of the results, show that there is generally excellent agreement between ENDRUN and the other codes, thereby verifying the averaging procedures used in ENDRUN for obtaining infinitely dilute cross sections.

The ground rules for performing these computations were the following:

1. The source of pointwise nuclear data was the latest ENDF/B file tape distributed by BNL.

TABLE 7-1
INFINITELY DILUTE CROSS SECTIONS FOR OXYGEN

	Group	ENDRUN	MC ²	GAM-1	MUFT-4	GAF-GAF
Elastic Scattering	1	0.7226	0.552	0.7224	0.7224	1.178
	10	4.848	3.872	4.849	4.849	4.914
	20	3.510	3.443	3.510	3.510	3.510
	30	3.635	3.490	3.635	3.635	3.635
	40	3.672	3.519	3.672	3.672	3.672
	50	3.709	3.554	3.709	3.709	3.709
	60	3.746	3.590	3.746	3.746	3.746
	68	3.765	3.608	3.765	3.765	3.765
Elastic Removal i-i +1,	1	0.2841	0.2854	0.2990		
	10	2.281	2.272	2.172		
	20	1.730	1.745	1.747		
	30	1.747	1.762	1.762		
	40	1.762	1.777	1.777		
	50	1.779	1.795	1.795		
	60	1.797	1.813	1.813		
	68	1.806	1.822	1.821		

TABLE 7-2
INFINITELY DILUTE CROSS SECTIONS FOR PU-239

	Group	ENDRUN	MC ²	GAM-1	MUFT-4	GAF-GAR
Elastic Scattering	1	3.176	0.5067	3.176	3.176	3.433
	10	4.654	2.644	4.654	4.654	5.435
	20	10.74	8.363	10.76	10.76	9.161
	30	12.23	11.52	12.22	12.22	11.88
	40	13.60	13.17	13.50	13.50	13.51
	50	17.28	6.904	19.33	19.68	18.75
	60	9.940	11.07	9.940	9.940	11.48
	68	16.09	15.80	16.09	16.09	16.07
Capture	1	0	0		0	0
	10	0.05685	0.05685		0.05685	0.0588
	20	0.2192	0.2185		0.2222	0.2211
	30	1.287	1.282		1.316	1.306
	40	9.217	9.388		9.517	8.918
	50	63.46	64.51			63.79
	60	1.443	1.436			1.384
	68	143.9	283.4		143.9	143.1
Fission	1	2.410	2.41		2.48	2.634
	10	1.719	1.719		1.719	1.720
	20	1.473	1.458		1.477	1.473
	30	2.476	2.529		2.531	2.499
	40	13.07	13.52		13.64	12.59
	50	18.02	19.23			18.75
	60	10.16	10.14			9.662
	68	288.0	518.5		288.0	286.8
Inelastic	1	0.8191	0.8191		0.8893	
	10	0.7863	0.7864		0.7863	
	20	0.2710	0.2710		0.2710	
	30	0	0		0	
	68	0	0		0	

2. Averaging began at 10 MeV and $1/E$ neutron flux spectrum was assumed at all energies.
3. Only infinitely dilute cross sections were to be calculated.

The choice of oxygen and Pu-239 provides comparisons for (1) a material which has all cross sections given as smooth data on ENDF/B and (2) a material which contains both unresolved and resolved resonance parameters. For Pu-239 the values in groups 19 through 41 were computed from unresolved resonances plus smooth data, groups 42 through 65 from resolved resonances plus smooth data, and the remainder of the groups from purely smooth data.

For oxygen elastic scattering, which is totally smooth data, there is almost identical agreement between ENDRUN, GAM-1, and MUFT-4. The ENDRUN elastic removal cross sections are slightly smaller, possibly due to a small difference in the way they are computed. For Pu-239 there is again excellent agreement between ENDRUN, GAM-1, and MUFT-4 in those groups containing only smooth data. In the unresolved resonance groups ENDRUN agrees well with the others, the difference probably being caused by variations in the ways the parameter distributions are treated and the intermediate pointwise results are averaged. ENDRUN agrees more closely with GAF-GAR than with MC² at the higher end of the resolved resonance energy range and more closely with MC² at the lower end.

In addition to the generation of GAM-1 group constants for oxygen and Pu-239, ENDRUN is currently being used to compute data from ENDF/B for computations of ZPR-III, Assemblies 11 and 48. Comparison of these results with similar results using MC² will provide a more stringent test of ENDRUN since resonance self-shielding is

important in these assemblies. If the comparison is favorable, the ENDRUN-FCCIV approach may be a relatively inexpensive technique for testing the ENDF/B data. Twenty-nine group cross sections (26 groups of 0.5 lethargy width, and 3 of 1.5) have been generated for H, D, Li-7, Be, B-10, C, Ni, O, Mg, Al, Ti, V, Fe, Cr, Ni, Xe, Pu-238 and Pu-241. Cross sections for U-238, Pu-239, Pu-240, Pu-242, U-236, U-235, Na, Mo, and Mn are now being generated. Self-shielding factors are also being computed where they apply. All material pointwise cross sections have been taken from the most recent ENDF/B file tapes distributed by Brookhaven National Laboratory, except Al, Be, Ni, Cr, and Pu-241. For the 4 latter materials a modified file was used which is based on the previous ENDF/B category 1 tapes and changes by Argonne National Laboratory to conform to MC² requirements. The same requirements prohibited the use of the BNL tapes with the ENDRUN code.

7.2.2 Cross Section Files

An analysis of s wave statistical resonance parameters for Pu-239 has shown good agreement between computed cross sections and measured values of the total and fission cross sections and the RPI-ORNL⁽¹⁰⁾ alpha values. Since both the precise measurements of Uttley⁽¹¹⁾ and James⁽¹²⁾ indicate variations in the total and fission cross sections below 2 keV (where s-wave predominates) which are over and above the normal resonance structures, variable neutron and fission widths (for $J = 1$) were assumed in this analysis. The average reduced neutron width was chosen as a function of energy to give the total cross section as measured by Uttley. Using these values of $\langle \Gamma_n^0 \rangle$, the average $J = 1$ fission width was determined as a function of energy such that the computed alpha value agreed with the preliminary ORNL-RPI measurements. The resulting fission cross section

was in substantial agreement with measured values. Table 7-3 lists measured and computed data as well as the average reduced neutron and fission widths at a number of energy points.

7.2.3 Multigroup Cross Section Adjusting

Checkout of the ENDRUN code is continuing, with the computations performed in support of ENDF/B data testing also providing a good test of the averaging techniques in ENDRUN. (See Section 7.2.1 ENDF/B Evaluation and Testing.) Comparison of results for oxygen and Pu-239 in the GAM-1 group structure indicate that the ENDRUN averaging routines for infinitely dilute cross sections are operating correctly. The computations of ZPR-III, Assemblies 11 and 48, will provide further check out of ENDRUN.

The ENDRUN code provides the first step in the general system of multigroup cross section adjusting. A generalized multigroup cross section file, GMUG, has been specified and the capability for generating this file put into ENDRUN. This file is more comprehensive than the FCC files, also generated by ENDRUN, in that elastic scattering, inelastic scattering and $(n, 2n)$ reactions have separate matrices and there are fewer restrictions on accuracy. A code for servicing the GMUG file, PASSOP, has been written. The TDOWN code will use the GMUG file and compute regionwise, effective microscopic cross sections in a specified two-dimensional reactor description. The output cross sections from TDOWN can be used directly in reactor design physics codes, such as BISYN. The first version of TDOWN, which does not perform flux solutions, is about one-half completed.

7.3 CORE ANALYSIS

7.3.1 Multigroup Cross Section Condensation Methods

Earlier studies⁽⁴⁾ indicated that, by using a special recipe for choosing a condensed group structure, 9 and 16 group axial calculations of a large LMFBR give essentially the same results as 60-group problems. To obtain an idea of the accuracy of these condensed cross sections in two-dimensional problems, the same 9 and 16 group cross-section sets were used in calculations of the one-dimensional radial model of the same reactor. The results, given in Table 7-4, are compared to the results of the corresponding problem in 60 groups. The agreement obtained is as good as that obtained in the axial problems. Therefore, use of the special recipe for choosing a condensed group structure with a one-dimensional flux solution for condensation should give excellent accuracy when the condensed cross sections are used in two-dimensional design calculations.

7.3.2 Two-Dimensional Methods

A new code, called DOT2DB, combines the two-dimensional transport theory code DOT⁽¹³⁾ with the two-dimensional diffusion theory code 2DB^(14, 15), thereby providing a versatile nuclear analysis tool, capable of performing directly comparable, two-dimensional diffusion and transport theory calculations with greater efficiency than now available.

The code 2DB, written at PNL, has been demonstrated to be at least four times faster than the similar code, EXTERMINATOR-2.

TABLE 7-3

RESULTS OF ANALYSIS OF PU-239 S-WAVE RESONANCE PARAMETERS

Energy (keV)	$\sigma_t(b)$		$\sigma_f(b)$		α		$\langle \Gamma_n^0 \rangle_{J=0}$	$\langle \Gamma_n^0 \rangle_{J=1}$	$\langle \Gamma_f \rangle_{J=1}$
	Measured (Uttley)	Calculated	Measured (James)	Calculated	Measured (ORNL-RPI)	Calculated	(MeV)	(MeV)	(MeV)
0.300	41.39	41.39	14.3	9.8	1.50	1.50	1.16	0.411	11.8
0.350	32.79	32.79	9.5	8.8	1.13	1.13	0.904	0.321	25.9
0.400	28.99	28.99	9.6	9.7	0.68	0.68	0.803	0.285	78.7
0.450	25.19	25.19	9.7	8.5	0.56	0.56	0.679	0.241	119.4
0.500	32.69	32.69	13.6	11.9	0.57	0.57	1.08	0.382	104.3
0.550	40.19	40.19	17.5	13.1	0.73	0.73	1.51	0.535	56.1
0.600	31.79	31.79	10.8	6.51	1.45	1.45	1.13	0.402	11.4
0.650	23.39	23.39	4.22	4.00	1.62	1.62	0.717	0.255	9.7
0.700	23.19	23.19	4.39	4.02	1.55	1.55	0.733	0.260	10.8
0.750	22.99	22.99	4.55	4.75	1.15	1.15	0.747	0.265	23.6
0.800	22.69	22.69	4.44	5.07	0.98	0.98	0.753	0.267	33.9
0.850	22.39	22.40	4.32	5.23	0.88	0.88	0.758	0.269	42.6
0.900	25.04	25.04	5.34	6.23	0.84	0.84	0.950	0.337	44.1
0.950	27.69	27.67	6.36	7.31	0.78	0.78	1.15	0.409	48.6
1.000	23.99	23.99	5.09	6.08	0.76	0.76	0.930	0.330	54.3
1.200	21.19	21.19	3.81	4.95	0.74	0.74	0.811	0.288	58.1
1.400	21.19	21.19	3.81	4.51	0.83	0.83	0.876	0.311	43.7
1.700	21.19	21.19	3.81	3.85	1.0	1.0	0.965	0.343	26.7
2.000	20.29	20.29	3.62	3.25	1.1	1.1	0.960	0.341	20.3
2.500	19.39	19.39	3.43	3.01	1.01	1.01	0.977	0.347	23.8
3.000	18.74	18.74	3.03	2.87	0.92	0.92	0.994	0.353	28.6
3.500	18.09	18.09	2.63	2.69	0.87	0.87	0.991	0.352	31.7
4.000	17.64	17.64	2.47	2.56	0.82	0.82	0.998	0.355	35.3
4.500	17.19	17.19	2.32	2.42	0.79	0.79	0.994	0.353	37.7

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TABLE 7-4

GROUP CONDENSATION RESULTS IN THE RADIAL DIRECTION

	Base Case 60 Groups	Condensed Group Structures	
		9 Groups	16 Groups
<u>k_{eff}</u>			
Sodium In	0.995890	0.995585 (-0.03%)	0.996114 (+0.02%)
Sodium Out	1.00376	1.00326 (-0.05%)	1.00378 (+0.00%)
<u>Doppler $\left(T \frac{dk}{dT}\right)$</u>			
Sodium In	0.005433	0.005362 (-1.30%)	0.005312 (-2.22%)
Sodium Out	0.003477	0.003454 (-0.66%)	0.003428 (-1.41%)
<u>(Δk_{Na}) Total Loss</u>	0.00790	0.00771 (-2.45%)	0.00769 (-2.61%)
<u>Peak-to-Average Power</u>			
Sodium In	1.2121	1.2160 (+0.32%)	1.2103 (-0.15%)
Sodium Out	1.2672	1.2715 (+0.35%)	1.2662 (-0.07%)
<u>Breeding Ratio</u>			
Sodium In	0.95129	0.95164 (+0.037%)	0.95086 (-0.046%)

The convergency rate of the two-dimensional discrete-ordinates transport code DOT, written at ORNL, can be increased by using the acceleration techniques of 2DB. Furthermore, by using diffusion theory to obtain approximate convergence before switching to transport theory, the running time of DOT should be considerably reduced. Since 2DB and DOT have essentially identical input sub-routines, incorporation of the diffusion sub-routines of 2DB into DOT can be accomplished relatively easily. This new code is in the debug stage.

7.3.3 Fuel Cycle Methods

Checkout problems with the BICYCL code have been run correctly and the code is essentially checked-out. The large number of user options in this type of code makes it

virtually impossible to pre-test all possible cases. Some options will be rarely used, so these will not be checked-out until they are used for production runs.

The fact that the BICYCL code does not automatically conserve the power distribution makes it cumbersome to use in many cases. An approximate means for including such a conserved power distribution within the code is being investigated.

Specifications have been written for the versatile fuel management code FUMBLE. The aspect of fuel management dealt with in this code is that related to fuel burnup in reactor operation. This includes evaluation of reactivity effects, breeding (or depletion) and inventories of fuel, and fuel costs for changing economic conditions and supplied

fuel compositions throughout the reactor lifetime. Fuel cost components associated with fabrication and reprocessing are computed by treating these operations in a "black box" manner, using input data parameters such as reprocessing time, fissile material losses, unit fabrication costs, etc. Considerable flexibility is allowed in the specification of reactor fueling. Flexibility is provided in (1) fuel compositions for the startup reactor, (2) recycle schemes and fissile makeup compositions for subsequent core loadings, (3) amounts of fuel to be replaced in different positions of the reactor at any refueling, and (4) shuffling of fuel either from one part of the reactor to another or by temporarily storing fuel from the reactor for later additional burnup. The refueling specifications may be changed from one refueling interval to another, as well as operating interval time, power rating, and load factor. Fuel costs may be evaluated for several different sets of cost input data and may be based on net costs accrued and energy produced by spent fuel batches or on net costs incurred for all fuel held and energy produced by the entire reactor for each operating interval.

Computational features of the code include a synthesis approach to reactivity calculations which will considerably reduce problem running time and thereby allow economical use of the code as an analytical tool for optimization of fuel management strategy; a routine for calculation of combinations of recycle and makeup materials for reload fuel that satisfies both required reactivity and a specified power distribution and a comprehensive input data checking routine.

7.4 TRANSIENT AND SAFETY ANALYSIS

7.4.1 Accident Analysis

General specifications for the transient calculation portion of the FREADM code

have been completed. Coding of radial heat transfer routines has been completed for both steady state and transient conditions and is being checked out. Coding of the heat transfer calculations for a complete fuel bundle has begun, using these radial heat transfer subroutines. Point kinetics coding is completed except for some time step control options and coding of the sodium void and flow model is in progress.

Preliminary calculations using the point kinetics calculation with one axial fuel pin node indicate that the time step control used in the FREADM code will be quite effective in minimizing computation costs. A transient was initiated from operating conditions with a \$2 ramp insertion of reactivity in 0.1 sec. The allowable estimated fuel temperature increase between heat transfer calculations was 50°F. With these conditions, thirteen kinetics time steps were taken before returning to the heat transfer calculation, with the kinetics time steps determined by reactivity changes of 10¢ per step.

7.4.2 Fuel Failure and Propagation Study

A heat transfer model of a portion of a fuel bundle was developed to investigate the consequences of fault conditions in typical fast reactor fuel assemblies. The model was set up for solution by the Transient Heat Transfer (THT) computer program. Several accident conditions were considered in which local blockages were investigated.

The geometry and boundary conditions of the model are shown in Figure 7-1. Heat flow through the sections was assumed to be two-dimensional, which means that axial effects are neglected. This assumption tends to result in conservative estimates of temperatures because disturbances are assumed to occur over infinite lengths. Care must therefore be taken in interpreting results. The geometry chosen was based on a fuel pin of 0.250 inch diameter. The cladding is

FIGURE 7-1. MODEL GEOMETRY AND BOUNDARY CONDITIONS

0.015 inches stainless steel. Pins were spaced in a triangular array with a pitch-to-diameter ratio of 1.15. Such a tight spacing was chosen because of the inflexibility of the model to geometric changes. This value corresponds to a rod geometry in which conservative features and tolerance effects are included. In this way, the results should tend to be pessimistic for geometries where the pitch-to-diameter ratio is greater than 1.15.

It is hoped that this model will provide data indicating the relative probability of possible modes of fault propagation within a subassembly for given initial conditions. Such information can only be obtained with a very flexible model in which conditions can be examined at a particular instant and the analysis restarted with slightly different parameters. Velocity profiles, flux profiles, mixing, and spacers can all be changed at any time step, but unfortunately the geometry cannot be changed easily.

The model contains the following characteristics:

1. There are five coolant nodes, five clad nodes, and one fuel node in each column.
2. A heat transfer coefficient can be input between the fuel and clad.
3. The coolant velocity profile can be simulated.
4. Heat transfer between fuel nodes is not considered.

The first accident condition analyzed results when a highly pressurized fuel pin bursts so that all of the fission gas in the plenum is released in about one-half second. This gas release will probably occur at the hottest region, i.e., the narrowest coolant

gap. Escaping gas will spread into the two triangular channels, the shaded area in Figure 7-1. The thermal gradients in the opposite pin will be identical to those calculated by this model. The failed pin will have a low pressure, but the adjacent pin may have up to 1000 psi internal pressure.

The temperature profiles shown in Figure 7-2 are representative of those which would exist in undisturbed bundles. The worst possible case exists when all of the flow in the shaded area is reduced to zero and the blocked area has stagnant sodium. Heat transfer can only occur by conduction in this region to the cooler flowing sodium. The results indicate that the temperature approaches a quasi-steady state distribution within 0.3 seconds. The temperatures at 0.3 seconds after the flow blockage are shown in Figure 7-3. A more realistic case which was postulated was a flow reduction in the shaded area to about one-third of the normal flow-rate, lasting for about one-half second. The temperatures at 0.3 seconds after the flow reduction are shown in Figure 7-4.

The objective of this analysis was to assess the probability of fission gas release causing failure propagation. It appears that sodium boiling will not occur; however, the cladding temperature rise may cause severe thermal bowing and thermal stresses. Additional analysis will be required before any conclusions can be drawn relative to the probability of failure propagation.

7.5 THERMAL HYDRAULICS AND FUEL ANALYSIS

7.5.1 Heat Transfer

Three finite difference methods of solving the transient heat transfer equations were examined: i.e., explicit, implicit, and Crank-Nicolson. It was found that all three methods yielded very accurate results

FIGURE 7-2. COOLANT TEMPERATURE PROFILE (normal operation)

FIGURE 7-3. COOLANT TEMPERATURE PROFILE (local flow stoppage)

FIGURE 7-4 COOLANT TEMPERATURE PROFILE (local flow reduced to 1/3 normal)

when run with the stability limited time step of the explicit technique. The implicit and Crank-Nicolson schemes were also sufficiently accurate when time stepped with an allowable temperature change of 25 - 50°F per time step. It was found that the implicit and Crank-Nicolson schemes did not require a significant increase in computation time over the explicit technique.

To correct for possible numerical inaccuracies in the models presently used in calculating convective heat transfer, a new approach to the problem has been formulated. The resulting solution is an explicit unlimited stability solution with an improved description of convective heat transfer. The model has been set up in a computer program and the results are being compared against results from other models.

7.5.2 Revision of VELVET Velocity Distribution Calculations

A critical review of the VELVET code⁽¹⁶⁾, used to calculate detailed velocity and temperature distributions in fuel rod bundles, has been completed during the past quarter. A major programming error was found and corrected. An incorrectly written FORTRAN statement in the calculation of the circumferential eddy diffusivity ($E_{m\phi}$) had resulted in this term being an order of magnitude greater than, instead of approximately equal to, the radial diffusivity (E_{mr}). Revised velocity profiles were found to be inconsistent with the available experimental data. It was concluded that the VELVET diffusivity model was incorrect, and a new calculational procedure was developed based on recent Soviet studies^(17,18). Results are now in excellent agreement with measured velocity distributions for rod bundles with pitch-to-diameter ratios of 1.05 to 1.20.

The only published study which recognizes the importance of flow channel shape on circumferential mixing and momentum transfer is that of Ibragimov, et al.^(17,18) They suggest that the total circumferential transfer of momentum is the sum of two processes: a gradient (or diffusion) transfer of momentum due to molecular friction and small-scale turbulent eddies, and a convective transfer of momentum due to the large-scale motion of eddies. Convective transfer becomes important for channels with sharply varying cross-sectional shapes (e.g., close-packed rod bundles) where exchange between eddies of considerably different velocities may occur.

Based on these physical ideas, Ibragimov formulates expressions for the channel friction factor and the variation of wall shear stress around the channel periphery. Agreement with experimental data, for a great variety of complex shaped channels, is truly remarkable.

Knowledge of the wall shear stress distribution permits the entire velocity field to be defined provided one can assume some sort of a universal turbulent profile. Fortunately, experimental velocity measurements in tube bundles are consistent with the Von Karman circular-tube profile. Levchenko, et al.⁽¹⁹⁾, carefully measured the velocity distribution in a close-packed ($P/D = 1$) tricuspid channel and found excellent agreement with the Von Karman profile along all normal checked ($\phi = 7.5^\circ$ to 30°). Eifler and Nijssing⁽²⁰⁾ obtained velocity measurements in channels with P/D from 1.05 to 1.15 and concluded that "the velocity distribution normal to the rod surface obeys - at each peripheral position - the same laws as those valid for round tubes" although they did not make a direct comparison to the Von Karman profile.

The flow field calculations of VELVET have been modified to incorporate this calculational scheme. As discussed below, the results are in excellent agreement with experimental data, i.e., better agreement than that of any other published analysis. The method has the additional computational advantage of being non-iterative and is simple enough for hand calculations.

In the original VELVET report (GEAP-5411), VELVET predictions are compared with the experimental data of Eifler and Nijssing for the case $P/D = 1.10$ and $Re = 28,000$. Attention was drawn to the test section configuration and to the fact that, compared to an infinite, spacerless array, the flow will redistribute towards the center of the test area with maximum peaking at $\phi = 30^\circ$.

Experimental data points are plotted with the original and revised (corrected for $E_{m\phi}$ only) VELVET predictions in Figure 7-5. With $E_{m\phi}$ correctly established as being of the same order of magnitude as E_{mr} , the velocity distribution spreads circumferentially and does not fit the data well. In view of this difficulty, the original VELVET velocity profile calculation was replaced by the Russian method discussed above. Apparently, the agreement of the original prediction with the data was fortuitous, the result of counterbalancing errors.

Velocity distributions for P/D ratios of 1.05, 1.10 and 1.15 (all at Re of approximately 30,000) were calculated by this method, and are compared with the corresponding measurements of Eifler and Nijssing in Figures 7-6, 7-7, and 7-8. In each case, the agreement is excellent. Even the experimental peaking at 30° is observable.

Future work will concentrate on the thermal effects of the flow distribution. In particular, it will be noted that the flow field

is now determined without an explicit definition of the radial and circumferential momentum diffusivities. It will be necessary to define these in order to calculate the turbulent thermal diffusivities.

7.5.3 Fuel Swelling

7.5.3.1 Model Development

During the quarter the case of the swelling of gas bubbles arranged in a spacing analogous to face centered cubic (FCC) spacing of atoms in a metal lattice was completed. It was assumed in this study that hydrostatic pressure as well as viscous and surface tension forces resist bubble growth, and the bubble starts growing from a bubble with a radius equal to the lattice spacing of the fuel. Previously⁽⁵⁾ it had been assumed that bubbles start at $r_0 = 0$ at time = 0 and the hydrostatic pressure was zero. These new assumptions lead to a more complex differential equation but it is believed that physical conditions during the start of bubble growth are described more accurately than previously. This analytical model is being used in the computer program discussed in Section 7.5.3.4.

7.5.3.2 Physical Property Studies

The creep properties of 304 and 316 stainless steel have been examined over the temperature range of 900° to $1400^\circ F$ in an attempt to arrive at an expression which would correlate the creep rate of the clad as a function of both temperature and stress level. This expression is particularly valuable for assessing local restraint on the fuel when the clad is under an axial temperature distribution and will provide an expression which could be modified to show the dependence of the fast neutron flux on the clad restraint. Using creep curves presented in reference (21) and techniques discussed by Garafalo⁽²²⁾, the following expression was

FIGURE 7-5. VELOCITY CORRELATION-VELVET: EIFLER AND NIJSING

FIGURE 7-6. VELOCITY CORRELATION AT $P/D = 1.05$

FIGURE 7-7. VELOCITY CORRELATION AT P/D = 1.10

FIGURE 7-8. VELOCITY CORRELATION AT $P/D = 1.15$

determined to describe the minimum creep rate of 304 and 316 stainless steel, where the values of the constants appear following the equation.

$$\dot{\epsilon} = A_2 \left[\sinh \frac{\alpha \sigma}{nT} \right]^n \exp \left(-\frac{Q}{RT} \right)$$

where

$\dot{\epsilon}$ = Strain rate (in./in.-h.)
 σ = Tensile stress (psi)
 T = Temperature (°K)
 R = Gas constant (cal/deg mole)
 n = 1.98

Material Constant	304	316
A_2	2.34×10^{11}	3.9×10^{12}
α	0.733	0.253
n	3.66	3.66
Q	78,000 cal/mole	78,000 cal/mole

The above constants when used in the previous expression were found to agree with curves to within 85%.

At stress levels less than 10,000 psi, the above expression approaches the form:

$$\dot{\epsilon} = A \sigma^n \exp \left(-\frac{Q}{RT} \right)$$

where

$$A = A_2 \left[\frac{\alpha}{nT} \right]^n$$

Experimental data and reports discussing theoretical studies have been collected to assess how the fast flux may affect the parameters of the creep equation at different temperatures.

7.5.3.3 Correlation with Experimental Data

A computer program is currently being written to store and analyze fuel pin dimensional data generated by the irradiation tests performed under Task F. Currently there is provision for seventeen stored variables including the maximum and minimum cold pellet length, the initial fuel density, and the local power level. Ridges on post-irradiation clad profiles are recorded and are associated with particular pellet interfaces. Statistical correlations between pre-irradiation clad strains can then be determined by means of different computer subroutines. The program to store the information and the subroutine to analyze the data, assuming linear relations between pre and post data, has been completed and is operational using data from one fuel pin. If this procedure continues to show promise it will be applied to other pins and the effect of variables such as burnup and local power assessed.

7.5.3.4 Development of Computer Program

A computer program has been prepared to predict local fuel swelling as a function of local fuel temperature and burnup using the model discussed in 7.5.3.1. The program determines the local fuel temperature and pertinent fuel properties from which local fission gas bubble sizes are predicted as a function of fission rate and burnup. If the bubbles exceed a critical size, which is dependent upon the local temperature and the temperature gradient, motion of the bubbles up the temperature gradient to the grain boundary is predicted. If deposition of bubbles at grain boundaries is predicted, the local swelling caused by grain boundary bubbles is determined.

The total fuel swelling is then determined by adding the local swelling due to

bubbles within the grains to the swelling due to bubbles which have accumulated at grain boundaries and swelling due to solid fission products. The local swelling is determined in this manner at 50 equally spaced radial positions across the fuel pellet.

The program is currently being debugged. When the program is operational it can be used as an input in a fuel-clad interaction program, such as that of K.R. Merx⁽²³⁾, to predict local clad strain as a function of power level and burnup.

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 prospect 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 four 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. These objectives of specific activities are:

- a. 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 LMFBR environmental parameters. The candidate materials are within the range of compositions and structures which can be produced by present commercial process.
- b. Obtain 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.
- c. 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 MATERIAL IRRADIATION PERFORMANCE

8.2.1 Series L-2, L-6, L-10, L-14, and L-15—EBR-II Materials Capsules

Irradiation of L-6, L-14, L-15 series of material capsules is continuing in EBR-II. The L-10 series capsules reached the target total fluence of 2.0×10^{22} n/cm² and were discharged. The L-2 material capsules were removed from S/A XG05. One capsule L-2I was transferred to S/A X039 and reinserted into row 4. The other four capsules will be transferred to a subsequent S/A for continued irradiation later this year. As of September 30, 1968, capsule L-2I, and the capsule series L-6, L-10, L-14, L-15 have accumulated the following total fluences: 4.8, 3.8, 2.8, 2.2 and 0.03×10^{22} n/cm², respectively, based on EBR-II core physics. Complete descriptions of these series and their experimental objectives were given previously. (24, 25)

8.2.2 Series L-2', L-4/L-4'—EBR-II Materials Capsules

A description of these capsules was presented earlier with results regarding the post-irradiation testing of tubular and flat tensile samples. (1, 2)

Ring ductility tests have been conducted on rings from materials capsules L-2Q (Type 304) and L-2M (Type 316L). The ring test technique was described in a previous quarterly report. (1)

Ring samples from L-2Q were irradiated to $3.6 - 7.0 \times 10^{21}$ n/cm² ($E_n > 1$ MeV). * In reactor, these tubular sections were subjected to an initial hoop stress of 8.8 ksi at 1000 to 1300°F for ~ 5,000 hours. A highly

* Established by flux wire determinations.

localized creep strain from 1% to 12% $\Delta D/D$ resulted, which varied along the test section length.

The L-2M series ring samples were exposed to $0.5 - 7.2 \times 10^{21}$ n/cm² ($E_n > 1$ MeV). * In reactor these tubular sections were subjected to an initial hoop stress of 9.0 ksi at 1000 to 1300°F for ~ 5,000 hours. A highly localized creep strain from 1% to 14% $\Delta D/D$ resulted, which varied along the test section length.

Circumferential strain, from ring testing, was obtained from calibrated macro-photographs (Figure 8-1). Since failure was not always detectable on the load-deflection charts, the circumference of the ring (C_R) was obtained from the following relation:

$$C_R = \left(2 - \frac{\theta}{180}\right) \pi R$$

The symbols are defined in Figure 8-1.

The experimental design consisted of testing a total of 54 rings (0.075 ± 0.005 inch wide by 0.350 inch diameter and 0.015 inch wall thickness) in triplicate at 1300°F and four cross head rates on rings of known in-reactor strain history. Results are tabulated in Table 8-1 (Type 304) and Table 8-2 (Type 316L) and plotted in Figures 8-2, 8-3, and 8-4.

Ring ductility was observed to decrease with decreasing strain rate for both Types 304 and 316L (Figure 8-2). The error band for Type 316L falls within the error band of Type 304; thus, there appears to be no significant difference in their response to strain rate effects as determined from the ring test. These results reinforce the conclusions from earlier data, also shown in Figure 8-2, which indicated no difference in tensile ductility between Types 304 and 316L.

FIGURE 8-1. MACROPHOTOGRAPH OF FAILED RING USED TO DETERMINE CIRCUMFERENTIAL STRAIN

An extrapolation of the curves in Figure 8-2 suggests that at very low strain rates (creep range), the strain obtained from ring tests would approach the total elongation obtained from tensile tests.

Ring ductility decreases with fast neutron exposure for both Types 304 and 316L with the ring ductility always greater than tensile ductility at comparable neutron exposures (Figure 8-3). The disparity between the ring ductility data and the tensile elongation decreases with fast-neutron fluence and may be coincidental at $\sim 10^{22}$ n/cm² ($E_n > 1$ MeV).^{*} An anomaly exists in the type 316L data at 3.8×10^{21} n/cm².^{*} These data show that the circumferential strain at 3.8×10^{21} n/cm² is a maximum ($\sim 40\%$), which is greater than the strain at lower exposures ($\sim 25\%$ at 0.7×10^{21} n/cm²).^{*} This anomaly is currently unexplained, but may be attributed to the spread in these data.

^{*} Established by flux wire determinations.

The effect of in-reactor creep strain on circumferential strain is presented in Figure 8-4. Both types 304 and 316L lie within each others error band with the test strain decreasing with increasing in-reactor strain.

Observations made from ring test data are similar to those previously made for tensile data; i. e., in general no significant difference exists in the post-irradiation ductility of mill-annealed types 304 and 316L.

As discussed earlier,⁽¹²⁾ in the ring test surface shear forces develop which support the entire gage length of the specimen and thus suppress localized necking. These shear forces help to extract considerably more ductility from a specimen than any test that does not support necking regions such as tensile and pressure tests.

The effect of specimen support is represented graphically in Figure 8-5. As long as instabilities resulting in necking do not

TABLE 8-1

1300°F RING DUCTILITY FOR MILL-ANNEALEDTYPE 304 IRRADIATED AT 1000-1300°F

Neutron Exposure $\times 10^{21}$ n/cm ² (En > 1 MeV)	Average Strain Rate, min ⁻¹	In-Reactor Strain %	Failure Strain %	Notes
3.6 - 4.7	0.39	1	17, 23, 34	
3.6 - 4.7	0.038	1	14, 26, 28	Specimen tested to 26% did not fail.
3.6 - 4.7	0.0039	1	11, 25, 28	
3.6 - 4.7	0.00041	1	11, 21, 23-----	Tested 8 hours without failure; retested 2 more hours without failure; tested additional 3 hours and failed at 21%. Specimen tested to 23%, did not fail.
6.9 - 7.0	0.038	12	0, 0, 0	
6.9 - 7.0	0.038	2	12, 14, 15	
6.5 - 6.6	0.038	7	12	
6.5 - 6.6	0.038	7	3	Ring misaligned during test.
6.5 - 6.6	0.038	8	2	

TABLE 8-2

1300°F RING DUCTILITY FOR MILL-ANNEALEDTYPE 316L IRRADIATED AT 1000-1300°F

Neutron Exposure $\times 10^{21}$ n/cm ² (En > 1 MeV)	Average Strain Rate, min ⁻¹	In-Reactor Strain %	Failure Strain %	Notes
0.5 - 0.9	0.39	1	27, 30, 34	
0.5 - 0.9	0.038	1	19, 24, 25	2 cracks on 19% specimen
0.5 - 0.9	0.0039	1	12, 19, 21	
0.5 - 0.9	0.00041	1	17, 21, 22	Multiple cracks on 17% and 22% specimens
5.6 - 5.9	0.038	14	0.1, 0, 0	
7.1 - 7.2	0.39	1	25, 31, 32	
6.9 - 7.2	0.038	1	15, 20, 27	Multiple cracking on specimen tested to 15%
6.9 - 7.2	0.0039	1	19, 24, 25	Multiple cracks on specimen tested to 19%
6.9 - 7.2	0.00041	1	18, 19, 31	Multiple cracking on specimen tested to 31%
5.5 - 5.8	0.038	2	9, 13, 16	
3.7 - 3.9	0.038	1	28, 39, 40

FIGURE 8-2. EFFECTS OF DEFORMATION RATE ON RING DUCTILITY AT 1300°F FOR IRRADIATED MILL-ANNEALED TYPE 304 AND 316L

FIGURE 8-3. EFFECTS OF NEUTRON FLUENCE ON RING DUCTILITY FOR MILL-ANNEALED AND IRRADIATED TYPE 304 AND 316 L (Tested at 1300°F)

FIGURE 8-4. EFFECTS OF IN-REACTOR CREEP STRAIN ON RING DUCTILITY AT 1300°F FOR MILL-ANNEALED AND IRRADIATED TYPE 304 AND 316 L

FIGURE 8-5. COMPARISON OF STRAINS UNDER SUPPORTED AND UNSUPPORTED LOADING CONDITIONS

occur (i. e., uniform elongation as observed), the local strain at any point is equal to the total elongation and deformation is along the 45° straight line. In a test with unsupported gage length, necking will initiate at some point A. The measure of strain at the point is termed "uniform elongation." The strain in the material away from the neck increases at a slower rate from this point while the strain in the neck region increases rapidly to failure. If the gage section is supported so that necking cannot occur, then theoretically strain in all parts of the gage length will reach the strain at failure at the same time, and the uniform elongation would be equal to the local strain at failure (point B). It can be seen that for the greatest possible utilization of potential ductility complete support of potential neck areas is required.

Realizing that careful investigation of the effects of local deformation is still necessary, observations suggest that the loading pattern associated with fuel swelling may support the clad in a manner similar to that imposed upon ring specimens in the ring ductility test.

It cannot be stated with confidence that the above support is perfect (no necking allowed). The friction factor may be high between mandrel and specimen and between fuel and clad, but it may not be high enough to prevent all slipping. There is some evidence to support the belief that fuel and clad are in such intimate contact that slipping will not occur. But under long term loading, the surface shear stresses, helping to prevent necking, may be relaxed due to creep in the fuel. This means that support may only be partial and the deformation may be represented by the dotted curves in Figure 8-5 between the two limits shown (A and B).

The significant point is that the ring test results indicate that current fuel design

limits on allowable strain, which were obtained by unsupported testing, may be overly conservative because of the possibly supported loading by swelling fuel.

8.2.3 Burst Testing of Irradiated Fuel Cladding

Biaxial burst tests were successfully performed on Task F mixed-oxide fuel elements irradiated in EBR-II. These fuel elements were from the F-2 series irradiated in subassembly XG06 to a fluence of $1.5 - 6.7 \times 10^{21}$ n/cm² (En > 1 MeV),* and at irradiation temperatures as high as 1045°F.

Two candidate cladding alloys were tested: types 316 and 347, both in the mill-annealed condition. Test sections were taken from the first two inches in the upper blanket and the core region. The data, tabulated in Table 8-3, show that the burst strength for Type 347 clad was similar to that of Type 316; both decreasing with increasing test temperature. These trends are consistent with tensile properties from irradiated materials specimens. From Table 8-3 it is also apparent that fuel cladding from the core region has greater burst strength than cladding from the blanket region. Similar results were obtained from biaxial burst testing of F-2 capsule tubing,⁽²⁾ and is associated with the higher neutron fluence in the EBR-II core.

Future biaxial-burst testing is planned on other sections from F-2 fuel pins. For these tests, a biaxial-strain measuring system is being built so that the entire biaxial stress-strain curve can be developed.

8.2.4 XG05 Materials Examination

The XG05 subassembly was irradiated in EBR-II for ~ 12,616 MWD exposure or

* Established by flux wire determinations

TABLE 8-3
BURST TEST RESULTS FROM XG06 F-2 FUEL ELEMENTS

Fuel Element Designation	Material	Approximate Irradiation* Temperature (°F)	Test Temperature (°F)	Burst Pressure, ksi	Hoop Stress, ksi	Notes**
F2A (blanket)	347	930	1100	7.0	54.8	
F2N	347	930	1100	5.0	58.0	10-mil wall
F2Q	347	930	1300	5.2	40.7	
F2S	347	950	1100	13.0	54.2	30-mil wall no clear burst but leaked
F2U	347	930	1300	5.4	42.3	
F2W	347	890	900	8.0	62.6	
F2B	316	930	1000	6.9	54.0	3-mil dent in clad wall
F2E	316	930	1000	8.0	62.6	
F2Y	316	890	1300	5.0	39.2	
F2Z (blanket)	316	930	1300	5.2	40.7	
F2Z (Core)	316	1000	1000	6.6	51.7	
F2Y	316	960	1000	8.7	68.1	
F2Y	316	960	1000	9.5	74.4	
F2Q	347	1000	1000	8.0	62.6	
F2Q (Core)	347	1000	1000	8.2	64.2	

* Fluence for blanket specimens was $2.0 - 3.6 \times 10^{21}$ n/cm² (En > 1 MeV)
Fluence for core specimens was $5.0 - 7.0 \times 10^{21}$ n/cm² (En > 1 MeV)

** All cladding sections had an inner diameter of 0.220-inch and generally a 15-mil wall thickness, except where noted.

~ 280 EFPD. Structural materials from this subassembly which are presently scheduled for post-irradiation examination include:

two Hex tube flats
two 1/2-inch diameter shroud tubes
four 3/8-inch diameter capsule tubes
four 0.032-inch diameter wires.

All these materials are Type 304. The post-irradiation examination is aimed at providing data in three major areas:

1. Relative clad swelling

2. Uniaxial tensile properties

3. Biaxial burst and creep properties for highly irradiated Type 304

Initial γ -scans were performed on all materials to locate the flux centerline for reference purposes. All materials were measured to determine if any radiation induced dimensional changes had occurred. Diameter measurements were also taken at the EBR-II site on four remaining L-2 capsule tubes which are awaiting reinsertion into another subassembly. These capsule tubes

were previously irradiated in XG05 and allow for direct comparison with the four F-2 fuel capsule tubes from XG05. The results, summarized in Table 8-4, show that the percentage increase in diameter for the XG05 capsule tube increases with increasing irradiation temperature.

The shroud tube diameters were difficult to measure since they are dimpled along the entire length. However, there was a measurable increase in diameter (~ 1.2 mils) in the core region above that measured in the blanket region. Since the location of the shroud tubes is not known the irradiation temperature could not be accurately determined. Thickness measurements were taken along the length of both hex tube flats, and no systematic variations in thickness were detected which could be attributed to clad swelling. Unfortunately the scatter in these measurements was greater than the accuracy required to measure thickness variation due to swelling.

Selected sections of the XG05 materials were transferred to the clad swelling program for density measurements (see subsection 8.3). Following the density work, tensile specimens will be cut from sections of the hex flat for tensile testing.

8.3 CLAD SWELLING PROGRAM

An experimental program intended to assess the effect of fast flux irradiation on the density of FCR candidate cladding alloys was initiated. This program is designed to evaluate (1) the change in clad density as a function of alloy irradiated at a given temperature and fluence and (2) the effect of radiation temperature on changes in density of Type 304 stainless steel irradiated to equal fluences.

8.3.1 Density Measurements

During the past quarter examination of Type 304 hardware recovered from EBR-II subassembly XG05 was initiated. This subassembly was irradiated in Row 4 and accumulated an estimated total fluence of 5.7×10^{22} n/cm² in the peak core position. Specimens for density measurement were selected from the hex, shroud, and capsule tubes which represent in ascending order a radial temperature profile of 700 to 800 °F. The hex and shroud tubes were sampled at positions adjacent to the radial centerline of the core; the capsule tube was sampled axially, at positions above and below the core centerline. All measurements were made by the technique of immersion density described previously.⁽⁴⁾

TABLE 8-4
COMPARISON OF DIMENSIONAL CHANGE
OBSERVED IN S/A XG05 MATERIALS

<u>Material</u>	<u>Estimated Irradiation Temperature (°F)</u>	<u>% Increase in Diameter at Core Centerline</u>
F-2T Capsule Tube (3/8 in. diam)	850	0.98% (3.65 mils)
L-2G Capsule Tube (3/8 in. diam)	700	0.05% (0.2 mils)
Shroud Tube (1/2 in. diam)	>750*	0.24% (1.2 mils)

The density change data are presented in Table 8-5 and plotted as a function of fluence in Figure 8-6. Also included in the figure are data from other sources. (The GE-APO values for fluence were converted to equivalent EBR-II values to permit this comparison.)* The XG05 data compare well with the results of previous work and show that the swelling is indeed increasing with increasing fluence. It is also apparent from Tables 8-4 and 8-5 that the measured density changes are indeed manifested in measurable dimensional changes, with a 3 to 1 relation (density to diameter) that one would expect if swelling is isotropic.

8.3.2 Electron Microscopy and Gas Analysis

In support of the density measurements, analytical programs utilizing transmission electron microscopy and helium analysis have been initiated to provide information regarding the mechanism of radiation-induced

swelling in austenitic stainless steel. Work on both programs has concentrated on specimens with radiation histories, identical to those previously examined,⁽⁴⁾ to permit direct correlation of observed swelling with microstructural and chemical changes.

Examination of Types 304, 347 and Incoloy 800 specimens by transmission electron microscopy (TEM) has been completed. The results, along with the previous density determinations, are given in Table 8-6.

The most significant observation these data provide is the degree of correlation of observed pore (or cavity) volume with measured changes in density. A basic assumption used in the bulk of the clad swelling work performed to date, both in the U.S. and abroad, is that volume increase (or density decrease) occurring as the result of irradiation in a fast flux is equivalent to the total volume of pores generated and maintained in the structure. The results shown on the annealed 304 and Incoloy 800 specimens

* GE values based on flux-wire data are about 60% of those reported by EBR-II.

TABLE 8-5

VOLUME SWELLING OF XG05 MATERIALS (TYPE 304)

<u>Material</u>	<u>Sample No.</u>	<u>Irradiation Temperature (°F)</u>	<u>Average Total Fluence</u>	<u>$\Delta\rho/\rho_0$ (%)</u>	<u>Average $\Delta D/D_0$ (%)</u>
Capsule Tube	F2R-a	760 - 810	3.2×10^{22}	1.35	0.51
Capsule Tube	F2R-b	810 - 855	4.5×10^{22}	1.95	0.73
Capsule Tube	F2R-c	855 - 890	4.5×10^{22}	2.06	0.77
Capsule Tube	F2R-d	890 - 910	3.2×10^{22}	1.41	0.51
Shroud Tube	B-b (SB-3)	750*	4.5×10^{22}	1.64	0.24
Hex Tube	HA-5	740*	4.3×10^{22}	1.13	

* These temperatures were estimated and could vary from 700 to 775°F for the shroud tube, depending on what kind of capsule it contained, and 737 to 753°F for the Hex tube, depending on what capsules were adjacent to these flats.

FIGURE 8-6. RADIATION INDUCED SWELLING OF AUSTENITIC STAINLESS STEELS (Data are on Annealed Type 304, Except Where Otherwise Noted)

TABLE 8-6
TRANSMISSION ELECTRON MICROSCOPY DATA ON
CAVITIES IN IRRADIATED AUSTENITIC ALLOYS

MAT'L	SAMPLE NUMBER	CONFIGURATION	AS-REC'D CONDITION	ELECTRON MICROSCOPY			DENSITY	
				AVG. CAVITY DIAMETER, Å	AVG. CAVITY DENSITY, cm ⁻³	AVG. CAVITY VOLUME, %	$\Delta\rho/\rho$ (%)	SAMPLE NO.
304*	212, 218	tensile coupon	Annealed	277	8.4×10^{14}	1.14	1.21	L2Q-2
304**	S7	tensile coupon	Carbide Agglomerated	--	0	0	0.14	L4F-4
347	L20-2	tube	Annealed	76	2.2×10^{15}	0.07	0.54	L20-2
I-800	L2K-3	tube	Annealed	177	7.0×10^{15}	2.1	2.47	L2K-3

* Data are average of numerical results on five regions from two specimens (irradiated in L2Q-2 and L2Q-3).

** Coupon S7 was irradiated in test section L4F-4.

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attests to the validity of this assumption. One should be cautioned, however, that TEM data of this type cannot in all cases be used to accurately monitor volume changes occurring in a material. This is due primarily to experimental problems or limitations inherent in all TEM. A case in point is the 347 data in which density measurements indicated a volume increase of approximately one-half percent while the TEM data showed a cavity volume an order of magnitude lower. Note, however, that the average cavity diameter observed in this specimen is one-half to one-third of that in the specimens which showed close correlation with density measurements. This emphasizes that the smaller the cavity size, the less reliable are TEM measurements of volume changes in irradiated steels.

The data on carbide agglomerated 304, although it cannot be judged conclusive at this stage, is significant from an applications standpoint. This material contained only one observable cavity and experienced the smallest density change of any alloy yet tested in this program. The cause of this apparent immunity to swelling is not fully understood; further studies on material with this structural condition are in progress.

An integral part of the swelling program is the attempt to assess the influence of gases (produced by neutronic reactions) on the swelling of austenitic stainless steels. Although substantial quantities of two gases (hydrogen and helium) are produced by neutronic reactions, the only real concern is the amount of entrapped gases which, at the temperatures of FCR interest, is expected to consist predominantly of helium. Because of its low mobility, moreover, virtually the total amount of helium produced remains in the material and can thus be measured.

One technique recently developed is that of isotopic dissolution analysis in which the

He⁴ recovered from a sample by vacuum fusion is diluted with a known quantity of He³, after which the ratio of the two isotopes is quantitatively measured by mass spectrometry. Analyses based on this technique were performed at Atomic International on samples of the five austenitic stainless steel alloys examined in the earlier reported density program. The samples were cut from tensile coupons irradiated in the L-2' series of materials capsules. The maximum radiation temperature of all coupons was established by fusible metal thermometry to be slightly in excess of 1200°F. Fast ($E > 1$ MeV) neutron exposure, determined by flux wire analysis, was $6.2 - 6.7 \times 10^2$ n/cm². All coupons were mechanically tested at 1300°F after irradiation. The results of the analyses are shown in Table 8-7, along with the as-received contents of the principal helium-producing elements.

Although the data do not show a meaningful correlation with density changes, the results do emphasize that substantially more helium is generated in austenitic alloys than physics calculations have indicated. Calculations using the most recent cross-section data predict that no more than 16 ppm of helium should be generated in any of the alloys tested. The data do suggest, however, that nitrogen is a prime source of helium in irradiated 18-8 alloys.

8.4 MATERIALS MANAGEMENT

8.4.1 Characterization of Incoming Material

One of the prime functions of the materials management program is the complete quality characterization of incoming FCR cladding material, the ultimate objective of this work being the establishment of a realistic, yet custom, specification for the procurement of nuclear grade austenitic stainless steel tubing. To provide input data for such an evaluation and at the same time provide cladding material for current irradiation

programs, identical orders for tubing were placed with four commercial vendors. The tubing orders called for the production of 1000 feet of Type 304H and 2000 feet of Type 316H, 0.250 inch outer diameter and 0.015 inch wall. At this time, three of the four tubing orders have been received and are undergoing inspection as outlined in Table 8-8.

Much of the work during the current period, however, was directed toward the complete characterization of the first large order of seamless thin wall tubing which included all five candidate cladding alloys. Table 8-9 lists the alloys examined in this primary program. The entire tube order was processed by a single vendor, Superior Tube Company. Shown in Table 8-9 are the vendor-supplied analyses of primary constituents and independent analytical results on the concentration of tramp or impurity elements.

One of the most critical control areas for tubing intended for use in experimental fuel pins irradiations is that of dimensional variations. Figures 8-7 and 8-8 illustrate

the type of data being generated in this area. Figure 8-7 shows the variations in outer and inner diameter occurring along 4-foot tubing lengths; Figure 8-8 illustrates the degree of eccentricity existing in randomly selected tubing sections. The results show that the bulk of all sections examined are within the limits specified for all three dimensions.

Still another type of dimensional variation is that caused by surface defects, inclusions, pits, and the like. Such local irregularities are not only the most difficult to detect but the most difficult to control or eliminate. One of the purposes of the current program is to characterize surface defects, to identify sources of such irregularities and assess their effect on mechanical properties of tubing on which they occur. Typical surface defects found on tubing sections by visual examination are shown in Figure 8-9 and 8-10. Based on these observations, it was concluded that the majority of surface defects are produced during tube forming. This work resulted in the modification of then-existing procurement specifications to assure close control over all forming and handling operations.

TABLE 8-7
HELIUM ANALYTICAL DATA

MATERIAL	SPECIMEN NUMBER	IRRADIATION TEST SECTION	ELEMENTAL COMPOSITION, % (AS-REC'D)				He ⁴ CONCENTRATION (ppm, atomic)	$\Delta\rho/\rho_0$ TEST SECTION (%)
			Fe	Ni	B*	N*		
304	217	L2Q-2	69.55	9.15	82.4	2160	33.7	1.21
316L	108	L2M-2	64.45	13.31	49.5	2000	14.7	1.25
321	258	L2P-3	70.29	9.53	95.2	438	7.0	0.14
347	167	L20-2	69.21	9.32	61.8	2240	78.5	0.54
I-800	37	L2K-3	46.09	32.00	61.8	3710	21.2	2.47

* ppm (Atomic)

TABLE 8-8
INCOMING TUBING QUALITY CONTROL

* Ambient and Elevated Temperature

TABLE 8-9
QUALITY CONTROL INSPECTION OF SEAMLESS TUBING
CHEMISTRY AND STRUCTURE

Material	Heat Number	CHEMISTRY: CERTIFIED ANALYSIS AND CHECK ANALYSIS														STRUCTURE		
		C	Mn	P	S	Si	Ni	Cr	Mo	Nb+Ta	Ti	Cu	Fe	Al	N**	ASTM gs	Homoge- neity Zone on ID	Inclu- sion
304	20184	<u>0.05</u> 0.05	<u>1.59</u> 1.59	<u>0.017</u> 0.011	<u>0.011</u> 0.014	<u>0.47</u> 0.47	<u>9.15</u> 9.12	<u>18.85</u> 18.95	† 0.35	† <.001	Nil	† <.01	Balance	Nil	† 852	5 to 5-1/2	Uniform gs	-
316	20303	<u>0.06</u> 0.05	<u>1.48</u> 1.54	<u>0.018</u> 0.017	<u>0.012</u> 0.012	<u>0.57</u> 0.60	<u>13.23</u> 13.24	<u>17.95</u> 17.82	2.26 2.25 †1.20	† 0.03	† 0.01	† 0.06	Balance	† 0.10	† 513	6-1/2 to 7	Uniform gs	-
321	28967	<u>0.06</u> 0.06	<u>1.79</u> 1.83	<u>0.017</u> 0.019	<u>0.014</u> 0.011	<u>0.52</u> 0.55	<u>11.23</u> 11.15	<u>17.45</u> 17.80	† 0.26	† 0.07	<u>0.44</u> 0.49 †0.41	† 0.10	Balance	† 0.015	† 273	8	Fine Grain Zone on ID Surface	-
347	24772X	<u>0.06</u> 0.05	<u>1.77</u> 1.81	<u>0.013</u> 0.005	<u>0.006</u> 0.014	<u>0.54</u> 0.48	<u>11.65</u> 11.60	<u>18.29</u> 18.16	† 0.28	<u>0.753</u> 0.72 †0.80	† 0.08	† 0.21	Balance	† 0.20	† 453	9-1/2 to 10	Uniform gs	-
I-800	HH2051A	<u>0.08</u> 0.06	<u>0.88</u> 0.81		<u>0.007</u> 0.009	<u>0.35</u> 0.42	<u>31.09</u> 31.20	<u>20.30</u> 20.10	† 0.23	† 0.08	<u>0.44</u> 0.41 † 0.55	<u>0.41</u> 0.43 † 0.24	<u>46.86</u> Balance	<u>0.096</u> 0.30 Nil	† 235	7 to 7-1/2	Uniform gs	-

- * All materials are relatively clean.
- * All materials are relatively clean: inclusions are random.
- ** ppm by weight.
- † Chemistries provided by Metallurgical Testing Corporation.

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FIGURE 8-7. DIMENSIONAL INSPECTION RESULTS, 316 SS SEAMLESS TUBING, 0.250 inch o.d., 0.015 inch WALL, HEAT No. 20303

FIGURE 8-8. ECCENTRICITY IN TYPE 304 TUBING PROCURED ON PO No. K-0666-G

68 PX 1000

TUBE SECTION N1

68 PX 1001

68 PX 1002

TUBE SECTION N2

68 PX 1003

N3 68 PX 1005

N4 68 PX 1006

N5 68 PX 1007

N6 68 PX 1008

FIGURE 8-9. NATURALLY-OCCURRING SURFACE DEFECTS ON SEAMLESS 316 SS (X40)

X40

1.5 mil SURFACE DEFECT

X500

X40

1.7 mil SURFACE DEFECT

X500

FIGURE 8-10. SURFACE DEFECTS ON AS-RECEIVED SEAMLESS TUBING TYPE-316

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