50
6/4/68
FAST REACTOR
GEAP-5584
AEC RESEARCH AND
DEVELOPMENT REPORT
MARCH 1968

A Distr. Journal
72R

MASTER

**SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR
DEVELOPMENT PROGRAM
TWENTY FIFTH QUARTERLY REPORT
NOVEMBER 1967-JANUARY 1968**

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

ADVANCED PRODUCTS OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

Twenty-Fifth Quarterly Report

November 1967—January 1968

R. E. Skavdahl

R. E. Skavdahl, Project Engineer
Fast Ceramic Reactor Development Program

E. L. Zebroski

E. L. Zebroski, Manager
Sodium Reactor Technology

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

*Printed in U.S.A. Available from the
Clearing House for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia
Price: \$3.00 per copy*

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

ADVANCED PRODUCTS OPERATION

GENERAL ELECTRIC

SUNNYVALE, CALIFORNIA

5448-APO-116
225-AT-4/68

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

See

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation; expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or*
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.*

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

TABLE OF CONTENTS

	<u>Page</u>
SECTION I INTRODUCTION	1
SECTION II SUMMARY	4
2.1 Task B—Defected Fuel Behavior	4
2.2 Task C—Safety Experimentation	4
2.3 Task E—Fuel Performance Evaluation	5
2.4 Task F—Integral Fast Flux Irradiations	5
2.5 Task G—Core Neutronics, Thermal-Hydraulics, and Safety Analysis	6
2.6 Task L—Cladding Development	7
SECTION III TASK B—DEFECTED FUEL BEHAVIOR	13
3.1 General	13
3.2 Defected Fuel Testing (Series B9, B8, and Miniloop)	13
3.3 Fission Product Distribution (Series B3 and B4)	17
SECTION IV TASK C—SAFETY EXPERIMENTATION	17
4.1 General	17
4.2 Axially-Restrained Fuel Experiments	18
4.3 Axial Fuel Relocation Experiments	23
4.4 Defected or Sodium-Bonded Fuel Experiments	23
4.5 Flow Blockage Experiments	23
4.6 RML Sodium Filling System	24
SECTION V TASK E—FUEL PERFORMANCE EVALUATION	25
5.1 General	25
5.2 Center Temperature Limits	25
5.3 High Burnup Irradiations	28
5.4 Plutonium and Fission Product Distribution	28
5.5 Physical Property Studies	34
5.6 Experimental Fuel Fabrication	38
SECTION VI TASK F—INTEGRAL FAST FLUX IRRADIATIONS	43
6.1 General	43
6.2 Medium Power, Long-Term Irradiations	43
6.3 Low Power Fuel Swelling Irradiation	44
6.4 High Power, Intermediate Burnup Irradiation	44
6.5 Elevated Cladding Temperature Tests (Group Fb)	49
6.6 Bare Pin Irradiation	56
SECTION VII TASK G—CORE NEUTRONICS, THERMAL HYDRAULICS, AND SAFETY ANALYSIS	56
7.1 General	56
7.2 Nuclear Data and Group Constants	57
7.3 Core Analysis	64
7.4 Transient and Safety Analysis	73
7.5 Thermal Hydraulics and Fuel Analysis	74

TABLE OF CONTENTS (Continued)

	<u>Page</u>
SECTION VIII TASK L—CLADDING DEVELOPMENT	85
8.1 General	85
8.2 Materials Irradiation Performance	86
8.3 Fuel-Cladding Compatibility	92
8.4 Cladding Optimization Studies	97
REFERENCES	106
CONTRIBUTORS	108
DISTRIBUTION	109

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
3-1	Gross Gamma Scan of Capsule B4D	15
4-1	Task C Series IV Specimen C4J (Powder, Mixed-Oxide Fuel) Post-Transient Neutron Radiograph	19
4-2	Task C Capsule C4C (Specimen C4D) Transient Irradiation Results, 2nd Shot (TT1078) Temperature Histories, Inner and Outer Sodium Annulus	21
4-3	Task C Capsule C4C (Specimen C4D) Transient Irradiation Results, 2nd Shot (TT-1078), THTD Calculated Radial Temperature Profile	22
5-1	Transverse Cross Section of E1F Fuel (76,000 MWd/Te)	26
5-2	Longitudinal Section of E1F Showing Fission Product Ingot near Bottom of Fuel Column	27
5-3	FCR Capsule E1AA—Direct Measurement of In-Pile Thermal Conductivity	29
5-4	FCR Capsule E1AA—Fuel Pin	30
5-5	Microprobe Analysis of Mixed Oxide Fuel Irradiated in FCR Capsule E2D to 120,000 MWd/Te	31
5-6	Photomicrographs of Unirradiated and Irradiated Fuel from Capsule E1E	33
5-7	Schematic Drawing of Mechanical Properties Test Furnace	35
5-8	Furnace for Melting Point Measurements of Irradiated Mixed Oxide Fuel	36
5-9	Melting Points of Irradiated Fuel	37
5-10	Photomicrographs of Fuel in Bridges across Center Void of Pellet Fuel	39
5-11	Comparison of Powder and Pellet Fuel Irradiated to 129,000 MWd/Te in Capsule E5B	40
5-12	FCR Capsule E5F—Linear Fuel Movement	41
5-13	FCR Capsule E5F—Fuel Pin	42
6-1	Gamma Scan, Gamma Radiograph, and Photographs of Capsule F2F	45
6-2	Cross Section of F2F at Peak Flux Position	47
6-3	Neutron Radiographs of Capsules F2F, F4D, and F4F	51
6-4	Gross Gamma Scan of F4D	53
6-5	Gross Gamma Scan of F4F	54
6-6	F4D Capsule Cladding—Lower Capsule Region	55
7-1	Comparison of Pu-239 Alpha Values for BAS02 and BAS03 with U. K. Differential Measurements and Swedish Values	58
7-2	Pu-239 $\sigma_f(E)$ Values Used in BAS02 and BAS03	60
7-3	U-238 $\sigma_c(E)$ Values Used in BAS02 and BAS03	61
7-4	FCR Reactor Zones for BISYN versus CRAM Calculations—Power Distribution	65
7-5	Axial Power Density Distribution—BISYN versus CRAM	68
7-6	Radial Power Density Distribution—BISYN versus CRAM	69
7-7	Reactor Zones for BISYN versus CRAM Calculations, Sodium Void (Cross-Hatched Region)	70
7-8	Control Rod Worth (Δk) of Flux-Trap Natural B ₄ C Control Rods (Diameter = 3.0 inches)	72
7-9	Vibration Stability Factor versus Unsupported Length, Grid Spacers	76
7-10	Vibration Stability Factor versus Unsupported Length, Grid Spacers	77
7-11	Vibration Stability Factor versus Unsupported Length, Grid Spacers	78

LIST OF ILLUSTRATIONS (Continued)

<u>Figure</u>	<u>Title</u>	<u>Page</u>
7-12	Design Curve for Vibrational Stability—10-mil Cladding, Grid Spacers	79
7-13	Design Curve for Vibrational Stability—15-mil Cladding, Grid Spacers	80
7-14	Design Curve for Vibrational Stability—10-mil Cladding, Wire Wrap Spacer	81
7-15	Design Curve for Vibrational Stability—15-mil Cladding, Wire Wrap Spacer	82
7-16	Nusselt Number	84
8-1	Uniform Tensile Strain versus Neutron Fluence for Austenitic Stainless Steels and Incoloy 800 Irradiated at 1000°—1200°F and Tested at 1300°F	88
8-2	Yield Stress versus Neutron Fluence of Austenitic Stainless Steels and Incoloy 800 Irradiated at 1000°—1200°F and Tested at 1300°F	89
8-3	Tensile Stress versus Neutron Fluence for Austenitic Stainless Steels and Incoloy 800 Irradiated at 1000°—1200°F and Tested at 1300°F	90
8-4	Rupture Characteristics of Type 304 Burst Specimens	93
8-5	Effect of Test Temperature on the Hoop Strength of Irradiated Type 304 Stainless Steel Tubing	94
8-6	Effect of Test Temperature on the Relative Ductility of Control and Irradiated Type 304 Stainless Steel Tubing	94
8-7	Sodium-Fuel Reaction Product in Type 316 Stainless Steel-Clad Capsule Containing Hyperstoichiometric Fuel	98
8-8	Interaction of Austenitic Steel with Mixed Oxide Fuel in the Presence of Sodium. Couples Annealed for 1000 hours at 1400°F. All Photomicrographs: As-Polished, 500×	99
8-9	Best Etching of Grid—Burst Specimen 2	100
8-10	Tensile Specimen "A" before Test, Showing Area of Flow on Right	102
8-11	Tensile Specimen "A" after Failure	102
8-12	Local Strain versus Position Along the Gage Length of a Tensile Specimen of Type 316, 75% Cold Worked	103
8-13	Circumferential Strain and Wall Thickness Around Circumference of Burst Specimen No. 3	105

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
2-1	Task B—Summary of Current Mixed Oxide Capsule Irradiations	8
2-2	Task C—Current Experimental Test Series	9
2-3	Task E—Summary of Current Mixed Oxide Capsule Irradiations	10
2-4	Summary of Task F Fast Flux Fuel Testing Program in EBR-II	11
2-5	Summary of Task L Materials Irradiations in EBR-II	12
3-1	B9A Experimental Parameters	14
4-1	Summary of Series 5 Capsule Operation in GETR	23
4-2	Tentative Design Parameters for the Series 6 Capsules	24
5-1	Irradiated Mixed Oxide Fuel Parameters	34
6-1	Characteristics of Failed Task F Pins	48
6-2	Task F Group F5 Fuel and Cladding Parameters	50
7-1	Pu-239 Correction Factors Representing Changes from BAS02 to BAS03 Cross Section File	62
7-2	U-238 Correction Factors Representing Changes from BAS02 to BAS03 Cross Section File	63
7-3	Effects of Data Uncertainties on LMFBR Fuel Costs	64
7-4	BISYN Calculational Procedures—Power Distribution	66
7-5	BISYN versus CRAM Results—Power Distribution	66
7-6	BISYN Calculational Procedures—Sodium Void	67
7-7	BISYN versus CRAM versus EXTERMINATOR-2	71
8-1	Post-Irradiation Tensile Testing at 1300°F and A Strain Rate of 0.02 Minute ⁻¹	87
8-2	Burst Data on F2 Capsule Tubes—Irradiated and Control	95
8-3	Post-Burst Diametral Changes in Control and Irradiated Type 304 Stainless Steel Tubing	96

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 November 1, 1967 to January 31, 1968 on the currently active tasks of this program is described in subsequent sections.

This is the twenty-fifth 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, Numbers 1—82, from July 1959 through January 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.

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. Bailly (September 1963).
- GEAP-4350 Fast Reactor Meltdown Excursions - Analysis and Computer Program FARM I; B. C. Luci, D. R. Riley, and N. F. Friedman (December 5, 1963).
- GEAP-4410 Calculations of Doppler and Sodium Reactivity Effects; P. Greebler (November 1963).
- GEAP-4414 Reactor Safety and Fuel Cycle Economics, K. P. Cohen (November 1963).

- GEAP-4418 Liquid Metal Fast Breeder Reactor Design Study, 1000 MWe PuO_2 - UO_2 Fueled Plant, Vols. I and II; M. J. McNelly, et al. (January 1964).
- GEAP-4420 Influence of Doppler Effect on Meltdown Accident; N. F. Friedman (December 1963).
- GEAP-4466 Experimental Studies of Plutonium and Fission Product Redistribution in Fast Reactor Fuels; T. Lauritzen, et al., (July 1966).
- GEAP-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. Baily (July 1966).
- GEAP-4770 A Technical and Economic Evaluation of Vented Fuel for Sodium-Cooled Fast Ceramic Reactors; G. L. O'Neill, J. Duffy, J. C. Gilbertson, F. W. Knight, and D. B. Sherer (May 1965).
- GEAP-4787 An Analysis of Fast Reactor Transient Response and Safety in Selected Accidents; W. Meinhardt, Jr. (August 1966).
- GEAP-4804 Experimental Studies of Transient Effects in Fast Reactor Fuels, Series II; J. E. Hanson, J. H. Field, and S. A. Rabin (June 1965).
- GEAP-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-4859 Demonstration of Fission Gas Venting; G. L. O'Neill, et al., (June 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. Baily (May 1966).
- GEAP-4897 Irradiation Behavior of Hyper- and Hypo-Stoichiometric Plutonia-Urania Fuel at 70,000 MWd/T; E. L. Zebroski, W. E. Baily, and D. P. Hines (March 1965).
- GEAP-4922 BISYN - A Two Dimensional Synthesis Program; M. D. Kelley (December 1966).
- GEAP-4939 Safety and Economic Characteristics of a 1000 MWe Fast Sodium Cooled Reactor Design; K. P. Cohen and G. L. O'Neill, (April 1967).
- GEAP-5272 Evaluation and Computation of Pu-239 Cross Section Data for the ENDF/B Files; P. Greebler, P. Aline, B. Hutchins (January 1967).
- GEAP-5273 FORE II, A Computational Program for the Analysis of Steady-State and Transient Reactor Performance; B. E. Lawler, et al., (September 1967).
- GEAP-5340 Neutron Spectrum Measurements in a Fast Plutonium Assembly ZPR-III Assembly 48; P. S. Brown and A. Weitzberg (April 1967).
- GEAP-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. Bailly (February 1968).

GEAP-4469 Experimental Studies of Transient Effects in Fast Reactor Fuels, Series III, Pre-Irradiated Mixed Oxide ($\text{PuO}_2\text{-UO}_2$), Irradi-

ations 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).

SECTION II

SUMMARY

2.1 TASK B—DEFECTED FUEL BEHAVIOR

Post-irradiation examination of the long vented fuel specimen B4D indicates that it operated successfully to a peak burnup of 47,500 MWd/T. A minimal examination is continuing.

Correlation of measured fuel temperatures and power in sodium-bonded fuel specimen B8B indicates that the reaction product in the fuel/cladding gap of the specimen with $\text{O/M} = 2.00$ has a thermal conductivity intermediate between sodium and helium.

Fuel pin fabrication and capsule assembly was completed for fuel/sodium/cladding compatibility test D8C. Fabrication of the fuel specimen and capsule parts for capsule B8D is continuing.

Design continued on the forced circulation capsule to permit in-pile testing of defected fuel in flowing sodium of prototypic velocity. Both single and multiple fuel pin tests were identified. The first test, B9A, will investigate the performance of defected fuel under typical FCR conditions, with fuel stoichiometry the major parameter.

Design was initiated on an out-of-pile, radioactive miniloop to study the distribution of dissolved and eroded fuel from defected, irradiated, mixed oxide in dynamic sodium.

2.2 TASK C—SAFETY EXPERIMENTATION

The transient irradiation program for the Series 4 zero-burnup, axially-restrained fuel specimens was completed with the irradiation of specimens C4I and C4J. Both specimens contained $\text{UO}_2\text{-20% PuO}_2$ powder fuel of 84% TD. Preliminary neutron radiographic examination indicated cladding failure and fuel dispersal had occurred in C4J, which had experienced a molten fuel volume of ~70%. Destructive examination of this series continued with deencapsulation of specimens C4A and C4I, and sectioning of C4A.

Fast flux pre-irradiation of capsules F7C and F7D (specimens C4G and C4H, respectively) continued to a burnup of ~27,000 MWd/Te. The remaining four capsules (F7A, B, E, and F) are nearing completion and will be shipped to EBR-II in the near future.

Transient heat transfer correlation has been improved with the use of the THTD computer code. Detailed analysis of potential failure mechanisms is in progress, particularly for failed, pellet-fuel specimens C4A and C4B.

Two 24-inch long specimens (C5A and C5B) have completed pre-irradiation exposure in GETR to burnups of ~22,000 and 21,000 MWd/Te, respectively. Reencapsulation of these specimens for subsequent TREAT irradiation has been initiated.

2.3 TASK E—FUEL PERFORMANCE EVALUATION

Metallographic examination of the fuel-cladding interface of the two fuel pins from capsule E5B (high burnup comparison of powder and pellet fuels) revealed that the cladding had been attacked. Chemical and microprobe analyses have been initiated to identify the reaction products.

Irradiation of capsule E2J (high burnup comparison of solid and annular pellet fuels) was successfully completed to a burnup of 125,000 MWd/Te at 22 ± 1 kW/ft. Capsules E5C, D, E (solid pellet, powder, annular pellet fuels, respectively) continued their irradiation at 19.5 kW/ft to a current burnup of 35,000 MWd/Te.

Capsule E5F (linear fuel movement) began its irradiation during GETR cycle 93. All thermocouples, and the linear motion transducer inside the fuel pin, are operating successfully.

The fuel, cladding, and all capsule parts for capsule E1AA (direct measurement of in-pile thermal conductivity) are fabricated and ready for assembly. Fabrication of capsules E2K-N (basic swelling capsules) was deferred pending resolution of FY-1968 funding.

The final design of capsules E2P and E2Q (variable radial restraint) was completed. Fuel and cladding have been fabricated and are ready for assembly.

Melting point measurements of irradiated UO_2 - PuO_2 were initiated. Preliminary measurements suggest a 50° to 100°C decrease in the melting point at 55,000 MWd/Te.

A creep rate of 1.63 in./in./h was observed on a 95% TD cylindrical pellet of UO_2 , 0.482-inch diameter and 0.746 inch long, under a stress of 8800 psi at 1450°C .

2.4 TASK F—INTEGRAL FAST FLUX IRRADIATIONS

Destructive examination of failed fuel pin F2F, from S/A XG06, is continuing. Diametral increases of up to 0.053 inch were observed on the fuel pin. An estimate of the maximum cladding (Incoloy 800) strain at failure, obtained by measuring the cladding perimeter, amounts to 2%. Metallographic examination indicates that the cladding temperature did not exceed 1850° – 1900°F for any significant period of time.

Fuel pin F4D, removed from S/A X011 with failed fuel pin F4F, was found to have leaked fission gas to the space between the fuel pin and the capsule, despite the absence of any indications of failure in the neutron radiographs or gamma scans. Profilometer measurements of the fuel pin showed diametral increases of 0.8 mil and 0.4 mil over the fuel and blanket regions, respectively.

Helium leak testing did not indicate any evidence of a rupture in the capsule containing failed fuel pin F4F. Profilometer measurements of the capsule wall indicated a diametral increase of 1.75 mils at a position 7 inches from the bottom of the fuel column, which corresponds to the position of an intense peak on the gamma scan.

Xe-131 activity was detected in the cover gas of the EBR-II during the December shutdown for refueling. Two suspect subassemblies, XG05 and XA08, were transferred from the core to the S/A storage rack and will remain out of the core during the initial full power operation after startup. It is planned to detect the leaker by a process of elimination. Subassembly XG05 contains nine Group 2 capsules which have accumulated 73,000 MWd/Te burnup.

The accumulated burnups to date on Groups 0, 2, 3B, 4, 6, 7, and 8A are calculated to be 26,400; 73,000; 3,600; 35,300; 30,500; 27,000; and 11,400 MWd/Te, respectively. Groups 2 (S/A XG05) and 4 (S/A X011) are currently out of the core, while the remaining groups continue under irradiation.

Eighteen of the Group 3B (E-1H) specimens were assembled in subassembly X027, inserted into EBR-II core position 4B3, and irradiation initiated during November.

The Group 3A unencapsulated fuel pins have been inspected at the EBR-II site and are physically acceptable for irradiation in EBR-II. Preparation of the hazards analysis is in progress.

Fuel pin fabrication is continuing for the Group 5 encapsulated tests at elevated ($\sim 1250^\circ\text{F}$) cladding temperature. Ten of the 16 pins are completed and ready for encapsulation. The selection of a satisfactory subassembly design is yet to be resolved with ANL.

Fabrication of the fuel pins for unencapsulated subassemblies F9A and F9B is continuing. Preparations are under way at ANL to flow test the subassembly design chosen for these tests.

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

Computations of self-shielding factors from the ENDF/B data for iron and chromium indicated a lack of detail in the cross sections in the range from 500 keV to 1 MeV. It was recommended and approved at the January meeting of the Cross Section Evaluation Working Group that alternative ENDF/B information for iron and chromium be issued which has increased detail over this energy range.

The FCC-IV code, a zero-dimensional diffusion theory code which uses cross sections in the Russian format, was converted to the GE-635 computer.

Modifications are being made to the GE 60-group cross section data set BAS02 to incorporate the results from new data and

recent data evaluations. The principal departures involve the Pu-239 capture cross section between 0.6 and 20 keV, the Pu-239 fission cross section from 0.6 to 150 keV, and the U-238 capture cross section from 100 keV to 2 MeV.

The sensitivity of fuel costs to uncertainties in nuclear data was computed for two LMFBR designs. The total uncertainty range for the advanced design (low core leakage) was computed to be 0.25 mil/kWh, while that for the early design (high core leakage, BeO) was 30% greater. For both designs, the fuel cost uncertainty was dominated by the uncertainty in the α for Pu-239 below 15 keV.

Power distribution calculations using BISYN were found to give excellent results. It was concluded that only one initial solution (through the core in the radial direction) should be taken and shape function solutions need be calculated only in the reactor regions with significantly different compositions.

The code EXTERMINATOR-2 was placed in operation on the GE-035 computer.

Control rod worth calculations were performed using transport and diffusion theories. The latter was found to consistently overestimate control worth.

Several errors were found and corrected in FORE-II, and a new tape entered in the program library.

An initial study of fuel failure propagation examined a fuel pin, on the outer edge of a bundle, bowed against the channel wall. The results indicated that the bowed pin would probably receive sufficient cooling to avoid sodium boiling or cladding melting.

All of the components for the velocity profile hydraulic flow test were completed and stored pending loop availability.

A study of flow-induced vibration was performed which permits an estimate of the required spacing for both wire wrap and grid spacer designs.

A parametric study was completed, using the VELVET code, of the temperature distribution around a sodium-cooled fuel pin in a close-packed rod bundle.

2.6 TASK I.—CLADDING DEVELOPMENT

Tensile testing at 1300°F, after exposure in EBR-II to fluences of $1.3 - 6.7 \times 10^{21}$ nvt ($E > 1$ MeV) yielded uniform strains of 3.0—4.5% for Types 304 and 316L stainless steels, and 0.7—3.0% for Type 347 stainless steel and Incoloy 800, with all alloys having been irradiated in the mill-annealed condition. Evidence obtained from yield stress measurements indicates persistence of radiation strengthening at high fluences and irradiation temperatures $\geq 1/2 T_m$.

Elevated temperature burst testing of weld drawn Type 304 stainless steel tubing revealed a marked increase in hoop strength of material irradiated to 6.7×10^{21} nvt ($E > 1$ MeV) at 800°—900°F compared with that of as-received material. Disparity between irradiated and as-received tubing decreased with increasing test temperature. A pronounced reduction in elevated temperature ductility, relative to as-received tubing, was also noted.

Materials capsules L-2, L-6, L-10, and L-14 are currently in EBR-II. Their present total fluences, based on EBR-II core physics, are 4.7, 1.8, 1.2, and 0.6×10^{22} nvt, respectively.

All of the seamless tubing (~10,000 feet) ordered in the Spring of 1967 has been received and normal incoming tube inspection completed. Included are 304H, 316H, 321H, 347H, and Incoloy 800, all in the solution-anneal condition; and 316H and Incoloy 800 in the 25% cold-worked condition.

Metallographic examination of fuel-cladding and fuel-sodium-cladding compatibility couples was completed. This work has shown that austenitic stainless steel cladding is fully compatible with stoichiometric and substoichiometric fuel at 1400°F for periods as long as 1000 hours, but is attacked in similar systems containing hyperstoichiometric fuel. Cladding attack was somewhat more severe in systems containing sodium.

Preliminary tests using microhardness marks and etched surface grids show potential for detailed measurement of local strain in experiments with cladding materials.

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 (Smeas) (% TD)	Fuel-Cladding Gap (Diameter inch)	Composition		Cladding Material						
						PuO ₂ %	O/M							
B3C	Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	88	0.003	25	2.00	Type 347 stainless steel	V-Raft GETR	1250	21	45,000	Examination in progress.	
B3D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	85	0.006	25	2.00	Type 316 stainless steel	V-Raft GETR	1100	18	6,700	Stored—AEC directive.	
B3E	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.213	23	83	0.008	25	2.00	Type 316 stainless steel	V-Raft GETR	1100	18	8,700	Stored—AEC directive.	
B3F	Vented Fuel. Power profile simulates fuel in peripheral core position. Natural convection cooling design.	0.212	23	84	0.008	25	2.00	Type 316 stainless steel	V-Raft GETR	1100	15	23,500	Examination in progress.	
B4D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	84	0.007	25	2.00	Type 316 stainless steel	V-Raft GETR	1100	15	47,500	Examination in progress.	
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	850	15	14,500	Examination in progress.	
B8C	Fuel-sodium compatibility versus burnup. Fission gas release versus burnup. Fuel-sodium-clad compatibility. Two pins.	0.144 0.204	2.2 2.25	86 84	0.006 0.017**	25 25	1.97 1.97	Type 316 stainless steel	V-Raft GETR	1050	22	100,000* 50,000*	In fabrication.	
B8D	Fuel-sodium compatibility versus density and stoichiometry. Three pins.	0.214 0.214 0.214	1.75 1.75 1.75	80-88 80-88 80-88	0.0065 0.0065 0.0065	25 25 25	2.03 1.97 1.94	Type 316 stainless steel	V-Raft GETR	1050	22	50,000* 50,000* 50,000*	In fabrication.	

*Target

**Large gap required to accommodate sampling device

CEAP-5584

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		P	C	C	C	I	Axial restraint at fuel melting (~ 5000°F).
C4B	Identical to C4A	C		P	C	C	C	NC	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		P	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		P	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		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		P	Identical to C4E.
C4G (F7C)	UO ₂ -28% PuO ₂ fuel 90% smeared density pelleted fuel	C	I		P	P		P	Identical to C4E.
C4H (F7D)	Identical to C4G	C	I		P	P		P	Identical to C4E.
C4I ⁺⁺	UO ₂ -20% PuO ₂ fuel 85% dense powder fuel	C		P	C	C	NC	P	Axial restraint similar to C4A. Test transient is intended to be similar to that of C4D.
C4J	Identical to C4I	C		P	C	C	I	P	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		P	Identical to C4E.
C4L (F7F)	Identical to C4K	C	P		P	P		P	Identical to C4E.
C5A	10% enriched UO ₂ 90% smeared density pelleted fuel	C	C		P	P		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		P	P		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
		Diameter (inch)	Fuel Length (inch)	Fuel		Composition		Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)		
				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	Fabrication in process.	
E1B-1	Gap conductivity. Four pins	0.218	1.25	95.9	92.5	20	1.98	Type 347	GETR	—	20	500	Report in preparation.	
2	with diametral gaps ranging	0.218	1.25	94.2	94.2	20	1.98	stainless	GETR	974	20	500		
3	from 2 to 10 mils.	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 in publication.	
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 in publication.	
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 in publication.	
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			
E2K	Fuel swelling. Four separate pins at three power levels.	0.099	3	85.0	84.0	25	2.00	Type 304 stainless steel	GETR	600	8.0	0	Fabrication stopped pending contract negotiations.	
L		0.099	3	85.0	84.0	25	2.00			700	10.0			
M		0.099	3	85.0	84.0	25	2.00			800	12.0			
N		0.099	3	95.0	84.0	25	2.00			800	12.0			
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	Irradiation completed.	
2		0.218	5	96.0	88.5	25	1.98			1100	22			
E5A-1	Powder and pellet fuel comparison at low burnup.	0.216	4.9	86.8	84.3	25	2.00	Type 347 stainless steel	GETR	1100	27	2,000	Report in review.	
2		0.216	4.9	84.1	84.1	25	2.00		GETR	1100	27	2,000		
E5B-1	Powder and pellet fuel comparison at high burnup.	0.216	4.9	85.8	84.1	25	2.00	Type 347 stainless steel	GETR	1100	20	100,000	Destructive examination in progress.	
2		0.216	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	1,000	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	1,000	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	1,000	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	0	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	0	Fabrication 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	0	Fabrication in progress.	

-10-

GEAP-5584

TABLE 2-4

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

January 31, 1968

Group	Purpose and Major Parameters	Row	Number		Power (kW/ft)	Burnup (1000's MWd/Te)		Irradiation	
			SA	Fins		Target	Calculated	Finish*	Status
F-0	Low power swelling rate of high density fuel at various burnup levels.	7	XG02	1	5-6	40	26.4	November 1968	In reactor.
			XG03	2		60	26.4	March 1970	
			XG04	2	—	100	26.4	June 1972	
F-2	Medium power scoping study. Major parameters include smear density, stoichiometry, cladding coolant interactions, and burnup.	4	XG06	12	11-16	50	51.1	February 1967	Post-examination in progress.
			XG05	9		75	73.0	February 1968	Discharged from core for fission gas release test.
F-3A	Bare-pin irradiations to evaluate fuel cladding and cladding coolant interactions.	5	1	16	15-18	50	—	November 1968	Fuel at EBR-II Site.
F-3B (E-1H)	Density and stoichiometry effects for encapsulated fuel pins.	4	XO27	18	13-16	40, 80, 120	3.6	October 1968, August 1969, and July 1970	In reactor.
F-4	Investigate low density fuel behavior at high heat ratings.	2	XO11	9	15-18	50	35.3	April 1968	Discharged from core; SA released gas.
F-5	Fuel performance at FCR peak cladding temperatures.	4	1	16	15-16	100	—	January 1970	Fuel fabrication in process.
F-6	Fuel-cladding interactions under accelerated swelling.	7	XO10	4	7-9	50	30.5	December 1968	In reactor.
F-7	Pre-irradiate axially restrained fuel prior to transient test. Main variable is fuel density.	4	XO15	4	12-15	50	27.0	November 1968	In reactor.
F-8A	Evaluate decrease in density on swelling rate.	6	XO19	7	6-9	50	11.4	September 1969	In reactor.
			XO20	9		100	11.4	August 1971	
F-8B	Effect of void deployment method on fuel cladding interactions.	7	1	19	6-9	50	—	November 1969	Fuel pin encapsulation in process.
F-9A	Irradiation of bare pins to high burnup.	4	1	37	11-16	100	—	February 1970	Fuel fabrication in process.
F-9B	Irradiation of bare pins to high burnup.	4	1	37	10-13	100	—	July 1970	Fuel fabrication in process.

*Based on EBR-II operation at 45 MW and 50% load factor.

GEAP-5584

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		Projected Completion Date(2)
				Biaxial	Tensile		December 1967		
							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
<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	0.47	0.34	October 1969
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	0.18	0.11	September 1971
L-10	4—XO16	10	One of Type 321 stainless steel; two of Type 304 stainless steel; three each of Type 316 stainless steel, and Incoloy 800.	70	60 ⁽¹⁾	0.20	0.12	0.09	October 1968
L-14	4—XO25	19	Five each of Type 316 stainless steel and Incoloy 800; four of Type 304 stainless steel; three of Type 321, two of Type 347 stainless steels.	133	114 ⁽¹⁾	0.20	0.06	0.04	December 1968
L-15		4	All five alloys.	None	288	0.20	-	-	May 1970

1. Includes specimens of experimental alloy of base composition 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.

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 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. Activity release to the system limited to Kr-85 only appears attainable with vented fuel 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 B9, B8, AND MINILOOP)

3.2.1 Capsule B9A (Failed Fuel Performance in a Forced Convection Capsule)

Design continued on the B9 series of experiments to investigate the performance of fuel specimens intentionally defected to

simulate failed fuel exposed to typical FCR core coolant conditions. The tests planned for this series fall into two categories:

(1) failed fuel pin performance and, (2) failure propagation and interaction with adjacent pins. These tests require in-pile testing of both single and multiple fuel pins in dynamic sodium. The forced convection capsule is being developed to permit this testing in water cooled test reactors.

The first experiment in this series (B9A) contains an initially defected fuel pin to be operated for from 5 to 20 days under typical FCR power and core coolant conditions. The specimen would be compartmented into two axial regions to permit the differential testing of swelling, washout, and distortion characteristics of two fuel stoichiometries. Table 3-1 shows the experimental parameters. A helical induction pump was selected for the FCC because of its head and flow characteristics, light weight, compactness, and ease of adaptation to the in-pile testing requirements of double containment of the experiment. In addition, the pump stator is reusable, thus reducing both cost and handling problems before and after testing. (See subsection 4.5 under Task C for out-of-pile testing in support of the forced convection capsule development.)

3.2.2 Miniloop (Out-of-Pile Radioactive Fission Product/Sodium Loop)

Design was initiated on a miniature sodium loop to study out-of-pile the distribution of dissolved and eroded fuel from defected, irradiated, mixed oxide fuel in dynamic sodium. The objective of this work is to identify critical problem areas and support the in-pile defected fuel tests.

TABLE 3-1

B9A EXPERIMENTAL PARAMETERS

Number of specimens	2
Orientation of specimens	axial
Fuel	mixed oxide
Stoichiometry	1.98 (upper) 1.93 (lower)
Cladding	Type 316 stainless steel
Outside diameter	0.250 in.
Wall thickness	0.015 in.
Defect	axial slit (upper and lower)
Coolant	sodium
Direction of flow	up
Velocity	15—20 feet/second
Exit temperature from test section	1150°F (maximum)

The design provides for a section of irradiated, mixed oxide fuel to be inserted into the miniloop and exposed to a sodium temperature of $\geq 1000^{\circ}\text{F}$. A temperature gradient of $\geq 200^{\circ}\text{F}$ will be established in the loop to simulate the temperature gradient in the primary coolant loop of an FCR. After ≥ 100 hours of operation, the sodium will be sampled and the loop destructively examined to determine the fuel and fission product distribution in the system as functions of local velocities and temperatures in the loop.

3.2.3 Capsule B8B (Central Fuel Temperature Measurement and Compatibility of Sodium-Bonded Fuel)

Post-irradiation examination and analysis of capsule B8B continued through the report period. The capsule contained two axially aligned, sodium-bonded, mixed oxide fuel pins with fuel O/M's of 1.93 and 2.00.

Correlation of capsule heat transfer and calorimetry with physics and chemical burnup analyses indicate that the measured

central fuel temperature of 1000°C corresponds to a linear power generation rate of 14 kW/ft. Thus, the observed central temperature and linear power for the specimen with O/M = 2.00 suggests a larger temperature drop in the fuel-cladding gap than can be accounted for by simple sodium bonding. This may be caused by the presence of sodium oxide or sodium/uranium/plutonium oxides in the gap with conductivities intermediate between sodium and helium.

3.2.4 Capsules B8C and B8D (Fuel, Sodium, Cladding Compatibility Studies)

Fabrication continued on capsules B8C and B8D. These experiments are to evaluate the effect of stoichiometry, density, and burnup on sodium/fuel/cladding compatibility and support in-pile defected fuel tests.

Fuel pin fabrication and capsule assembly of capsule B8C was completed. The experiment is awaiting pre-irradiation neutron-radiographic calibration of the fission gas pressure bellows on the high burnup specimen ($\sim 100,000$ MWd/T target) prior to insertion into the reactor.

FIGURE 3-1. GROSS GAMMA SCAN OF CAPSULE B40

Fabrication of the fuel specimen and capsule parts for capsule B8D is continuing.

3.3 FISSION PRODUCT DISTRIBUTION (SERIES B3 AND B4)

3.3.1 Capsule B3F [Low Burnup, Typical FCR Length (23 inches), Sodium Ingress, Vented Fuel]

Examination and analysis of this experiment are now complete. Final interpretation and reporting will be done in conjunction with the companion capsule B4D.

3.3.2 Capsule B4D [Intermediate Burnup, Typical FCR Fuel Length (20 inches), Vented Fuel]

Figure 3-1 shows a gross gamma scan of the fuel column and cold trap region of

capsule B4D. Analysis of this scan plus additional multichannel analyses in both the fueled and cold trap regions of the capsule confirm that this vented fuel specimen operated successfully to a peak burnup of 47,500 MWd/T.

Capsule disassembly is under way to permit minimal examination of the specimen.

3.3.3 Capsule B3C (Medium Burnup, Typical FCR Length, Sealed Fuel, Natural Convection Cooled)

The post-irradiation examination of B3C was delayed pending completion of higher priority work on this program.

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 safety testing. 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 1968, 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 transients 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

The transient irradiation program for the Task C Series 4 zero-burnup, axially-restrained fuel specimens has been completed with the irradiation of specimens C4I (third transient) and C4J (second transient). Both of these specimens contain 84% TD, UO_2 -20% PuO_2 powder fuel.

The transient irradiation of specimen C4I was intended to be a repeat of that performed on specimen C4D (80% smeared density pelleted fuel). The C4D transient resulted in 50% of the fuel volume entering, but not passing through the heat of fusion.

Although the actual C4I transient was somewhat below that of C4D (128 MW peak versus 164 MW peak power, and 156 MW-sec versus 159 MW-sec integrated power), preliminary analysis indicates that approximately 50-55% of the fuel volume reached the heat of fusion. There was apparently no cladding failure.

The transient irradiation of specimen C4J was intended to be a repeat of that performed on specimen C4C (identical to specimen C4D), which produced a fuel volume melt of 67-69% and moderate fuel movement, without cladding failure (reported in the twenty-fourth quarterly).

The actual C4J transient was slightly below that of C4C (206 MW peak versus 230 MW peak power, and 177 MW-sec versus 186 MW-sec integrated power), yet cladding failure did occur (based on thermocouple reading during the transient, and post-transient neutron radiographs taken at TREAT and Vallecitos Nuclear Laboratory). The neutron radiographs indicate an extensive failure, with fuel dispersal extending over 8 inches of the 13.5-inch fuel column length (compares to 5 inches for the failed C4B and 3 inches for the failed C4A). The extent of fuel dispersal outside of the cladding can be seen in Figure 4-1, which

presents a 0-degree orientation neutron radiograph (taken at Vallecitos) of specimen C4J (compare with that of specimen C4B presented in the Twenty-First Quarterly Report, GEAP-5433).

Destructive examination of this series of specimens has continued with the deencapsulation of specimens C4A (failed specimen, 90% smeared density pellet fuel) and C4I (84% smeared density powder fuel), and sectioning of specimen C4A. Profilometer measurements have been obtained for both specimens, and preliminary analysis indicates some cladding deformation in the fueled region for both cases, with ridging occurring in the C4A pelleted specimen.

The EBR-II fast-flux irradiation of capsules F7C and F7D (fuel specimens C4G and C4H) has continued to a burnup of 26,000 MWd/Te. The reworking and preshipment examination (weld penetration measurements and capsule x rays and neutron radiographs) of the remaining four capsules have been completed. The irradiation proposal has been sent to ANL.

The post-transient, heat-transfer analysis of the zero-burnup, axially-restrained specimens has been improved by the adoption of the THTD transient heat transfer computer program⁽¹⁾ in place of the previously used ARGUS program.⁽²⁾ The implicit solution form of the THTD code tends to avoid stability problems characteristically associated with the explicit solution form. Specimen C4D transient irradiation results and calculated fuel temperatures are shown in Figures 4-2 and 4-3. It is significant to note that the THTD results predict the physically measured melt radius to within 1 mil of the actual value (i. e., within less than 1% of the actual value).

In an attempt to understand and define better the failure mechanisms associated with the failed pelleted-fuel specimens C4A and C4B, the following problems are under study:

FIGURE 4-1. TASK C SERIES IV SPECIMEN C4J
(POWDER, MIXED-OXIDE FUEL)
POST-TRANSIENT NEUTRON
RADIOGRAPH

FIGURE 4-2. TASK C CAPSULE C4C (SPECIMEN C4D) TRANSIENT IRRADIATION RESULTS, 2ND SHOT (TT1078) TEMPERATURE HISTORIES, INNER AND OUTER SODIUM ANNULUS

FIGURE 4-3. TASK C CAPSULE C4C (SPECIMAN C4D) TRANSIENT IRRADIATION RESULTS, 2ND SHOT (TT-1078) THTD CALCULATED RADIAL TEMPERATURE PROFILE

- a. A simplified, one-dimensional calculation has indicated that the presence of an argon bubble (capsule cover gas is argon) on the cladding surface might lead to a cladding melt-through. An investigation is under way, using the THTD code and capsule C4A-B intact geometry, to study the problem in detail (with bubbles of varying size and three-dimensional heat flow).
- b. An investigation has also been initiated in an attempt to correlate fuel thermal expansion and phase-change volumetric expansion to interior cladding volume and released fuel porosity. The intent is to predict measured cladding deformation, and to determine the resultant fuel-cladding hydrostatic pressures up to the estimated time of cladding failure. A time-share program has been developed for this purpose, using THTD output fuel volumetric temperatures and melt fractions.

4.3 AXIAL FUEL RELOCATION EXPERIMENTS

The Series 5 capsules (C5A, C5B) have completed steady-state irradiation (cycles 87 through 90) in GETR pool positions. Irradiation data is summarized in Table 4-1.

TABLE 4-1
SUMMARY OF SERIES 5
CAPSULE OPERATION IN GETR

Capsule	Cycle	Power (kW/ft)	Burnup (MWd/Te)
C5A	87	12.5	5930
C5B	87	12.0	5700
C5A	88	11.6	5460
C5B	88	12.3	5800
C5A	89	12.4	5590
C5B	89	11.3	5090
C5A	90	11.3	5050
C5B	90	10.9	4860

Reencapsulation of the capsules for transient irradiations in the TREAT facility was

initiated. Nondestructive examination (neutron radiographs and gross gamma scans) of the capsules revealed no apparent discrepancies. The schedule for completion of reencapsulation has been adjusted to concur with the revised fabrication and licensing schedule for the new TREAT shipping cask. Transient irradiations are tentatively scheduled to be initiated in March.

4.4 DEFECTED OR SODIUM-BONDED FUEL EXPERIMENTS

The design study to review capsule fabrication and assembly techniques continued. Preliminary drawings of the capsules have been completed and were reviewed. Final detail drawings have been initiated and will incorporate modifications resulting from the review.

A brief test program to develop sodium-bonding techniques for powder fuel has been initiated. Sodium filling tests with 60 and 80% smeared density powder pins (≈ 5 inches long) have been completed. Results of preliminary examinations indicate that the sodium-filled powder pins are acceptable. Design of a sodium-fill system to accommodate the longer length Series 6 pins (≈ 24 -inch active fuel) is in progress.

Table 4-2 summarizes tentative design parameters for the Series 6 capsules:

4.5 FLOW BLOCKAGE EXPERIMENTS

Design of the initial Forced Convection Capsule (FCC) mockup (1A) is under way. Operation is scheduled for mid-May. Power will be provided by three electrical resistance heaters, spaced with spiral wrapped wires to simulate a small portion of the close-packed LMFBR core. Instrumentation will include flow and pressure transducers as well as thermocouples.

Tests proposed for mockup 1A include sodium remelt, decay heat removal by

natural convection, instrumentation checkout and finally, behavior during a loss of flow accident. (See Task B for information on the in-pile portion of the Forced Convection Capsule program.)

4.6 RML SODIUM FILLING SYSTEM

The possibility of using eddy current techniques to detect voids in the sodium bond of the Task C reencapsulated, pre-irradiated fuel pins was investigated briefly. It was

concluded that eddy current tests could be used to detect voids in a relatively simple capsule cross section, but the complexities of the Task C capsules would severely limit the ability of this technique to produce useful results (especially in the critical area between spacers and the fuel). Because of this, it was decided to eliminate the eddy current test and rely primarily on carefully controlled filling techniques to provide a continuous sodium bond.

TABLE 4-2
TENTATIVE DESIGN PARAMETERS FOR THE SERIES 6 CAPSULES

	Gas-Bonded		Sodium-Bonded	
	Pellets C6A-3, C6B-3		Pellets C6A-1, C6B-1	Powder C6A-2, C6B-2
Composition	UO ₂	←————→		
Enrichment	5% (U-235)	←————→		
O/M	2.00	←————→		
Length	24-in. (active fuel)	————→		
Fuel/Cladding Gap	0.008-in. (diametral)	←————→		-
Density				
Smear	80% TD (8.776 gm/cc)	←————→		
Pellet	86% TD (9.44 gm/cc)	←————→		-
Fuel Diameter	0.212 in.	←————→		-
Blanket	Natural UO ₂	←————→		
Length	15 in. (2-in. insulator at bottom)	←————→		
Density	93% TD pellets	←————→		

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 has been possible so far primarily in thermal test reactors.

5.2 CENTER TEMPERATURE LIMITS

5.2.1 Capsules E1C, E1D, E1G
(Smeared Thermal Conductivity
at Low Burnup

The report on these capsules is in publication.

5.2.2 Capsules E1E and E1F (Smeared
Thermal Conductivity at High
Burnup

Metallographic examination of capsule E1F was completed. A radial cross section of the fuel is shown in Figure 5-1. Evidence of melting* was observed to 58% of the diameter of the fuel. The fuel adjacent to the cladding appeared to be more porous than the unirradiated fuel in a zone

extending in about 0.012 inch. A similar effect was observed in the E1E, E2C, E2D, and E2E fuel. (3-6) Fission products were observed in regions of the fuel which did not melt. In the melted region very few fission products were observed. A longitudinal section of the fuel near the bottom of the fuel pin is shown in Figure 5-2. A large ingot of fission products which was highly beta-gamma active was found in this section. It is hypothesized that all of the metallic fission products in the molten region agglomerated and formed the ingot. This is substantiated by the beta-gamma radiograph of the transverse cross section of the fuel shown in Figure 5-1. The dark area in the center of the radiograph indicates an absence of beta-gamma activity. The boundary of the dark area also corresponds to the molten boundary. Further examination of the fuel and ingot are to be carried out by microprobe analysis.

Physics calculations to determine the operating power and flux depression in the E1E and E1F fuel as a function of burnup were completed. Calculation of the fission product distribution in the fuel and the integral conductivity to melting were initiated and will be reported in the next quarterly report.

5.2.3 Capsule Series E1H (Effect of Void
Deployment on Fuel Performance

Capsule series E1H is being reported under Task F as Group F3B.

*Molten boundary defined as the porosity limit in the fuel.

CLADDING

MOLTEN
BOUNDARY

SUBSTRUCTURE
BOUNDARY

HIGH POROSITY
REGION

BETA-GAMMA RADIOGRAPH

MACROGRAPH

ALPHA RADIOGRAPH

FIGURE 5—1. TRANSVERSE CROSS SECTION OF EIF FUEL (76,000 MWd/Te)

BETA—GAMMA RADIOGRAPH

ALPHA RADIOGRAPH

← FUEL UO_2 INTERFACE

GEAP—5584

FIGURE 5—2. LONGITUDINAL SECTION OF EIF SHOWING FISSION PRODUCT INGOT NEAR BOTTOM OF FUEL COLUMN

5.2.4 Capsule E1AA (Direct Measurement of In-Pile Thermal Conductivity)

The fuel, cladding, and all capsule parts for capsule E1AA are fabricated and ready for assembly.

Fabrication of the W-Re in-fuel thermocouples has delayed assembly of the fuel pin and capsule. One W-Re thermocouple has been completed and the second thermocouple is 50% complete.

Capsule and fuel pin drawings of E1AA are shown in Figures 5-3 and 5-4. The fuel and test parameters for capsule E1AA were presented in the twenty-third quarterly report.⁽⁶⁾

5.3 HIGH BURNUP IRRADIATIONS

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

Irradiation of capsule E2J was completed on December 15, 1967. The capsule operated successfully to a burnup of 125,000 MWd/Te at a power of 22 ± 1 kW/ft. The calculated neutron exposure (>1 MeV) on the cladding was 1.7×10^{20} nvt.

The capsule is being cooled in the GETR canal prior to examination.

5.3.2 Capsules E2K, E2L, E2M, and E2N (Basic Swelling Studies)

Work on these capsules has been deferred. To date, three capsule bodies were accepted for fuel loading. Fabrication of the fourth body was in progress. Powder had been prepared for fuel fabrication.

5.3.3 Capsules E2P and E2Q (Variable Radial Restraint)

The final design of capsules E2P and E2Q was completed. Fuel and cladding have been fabricated and are ready for loading.

These capsules will provide information on the effect of cladding restraint on fuel swelling under axial restraint. The fuel pins will be removed from their capsules at burnup intervals of 10,000 to 15,000 MWd/Te, examined, and reencapsulated. Each interim examination of the fuel pins will include neutron radiography, gamma scan, and cladding profile measurement.

Design and fabrication of in-cell equipment to weld and NaK fill the capsules during reencapsulation was completed. Checkout of the equipment is in progress.

5.4 PLUTONIUM AND FISSION PRODUCT DISTRIBUTION

5.4.1 Radial Fission Product Distribution

Analysis of microcores from fuel pins E2C and E2D for Zr-95, Ce-144, Cs-137, Ru-106, and Pu-239 are in progress. The cores were drilled from radial fuel sections of each pin, dissolved, and the activity of the fission products determined using solid state counters. Mass spectrometer analysis of the core sample solutions is under way to determine the ratio of each fission product to U-238. The relative concentration of each fission product across the radius of the fuel will then be determined.

5.4.2 Electron Microprobe Examination of Irradiated Fuel

Examination of a microcore of irradiated fuel from pin E2D showed that it could be successfully examined in an unshielded microprobe. The elements technetium and ruthenium were detected during the cursory examination but deterioration of the carbon coating on the surface of the sample made it difficult to determine their location in the sample. X-ray pulse and specimen current image pictures showing a region of the fuel near the edge of the microcore which could be identified are shown in Figure 5-5. The light areas in the x-ray pulse pictures show

- 8 INDICATES WELD IDENTIFICATION ONLY.
7. ALL PRESSURE CONTAINMENT WELDS TO BE RADIOGRAPHED IN DUPLICATE TO DETERMINE DEPTH OF PENETRATION, RADIOGRAPH IN 4 ORIENTATIONS, 45° APART.
6. NO NICKS OR SCRATCHES GREATER THAN 10% OF THE WALL THICKNESS WILL BE ACCEPTABLE ON P3.
5. ALL WELDS AND BRAZES TO BE HELIUM LEAK CHECKED WITH HELIUM MASS SPECTROMETER. A LEAK RATE GREATER THAN 2×10^{-8} cc/sec. S.T.P. SHALL NOT BE ACCEPTABLE.
4. CHECK CONTINUITY OF T.C.'s BEFORE AND AFTER WELDING.
3. COVER GAS IN FUEL PIN TO BE HELIUM AT ONE ATMOSPHERE.
2. PRIOR TO ASSY. CLEAN ALL STN. STL. PARTS PER APED SPEC. P4C-YP3 AND PASSIVATE PER APED SPEC. P4A-YP1.
1. ALL WELDS SHALL HAVE A MINIMUM DEPTH OF PENETRATION OF 90%.
- NOTES:

FIGURE 5-4. FCR CAPSULE E1AA FUEL PIN

X-RAY PULSE IMAGE PICTURE OF RUTHENIUM
 L_{α} X-RADIATION

SPECIMEN CURRENT IMAGE PICTURE NEAR
EDGE OF FUEL

X-RAY PULSE IMAGE PICTURE OF TECHNETIUM
 L_{α} X-RADIATION

PHOTOMICROGRAPH OF AREA EXAMINED BY
MICROPROBE

FIGURE 5-5. MICROPROBE ANALYSIS OF MIXED OXIDE FUEL IRRADIATED IN
FCR CAPSULE E2D TO 120,000 MWd/ T_e

that both technetium and ruthenium are present in the same inclusions in the fuel matrix. The inclusions containing Tc and Ru are circled in the specimen current image and photomicrograph of the fuel. Preparation of a second sample from the same fuel pin is now in progress.

5.4.3 Electron Microscopy of Irradiated Fuel

Results of the examination of replicas from fuel irradiated in capsule E1E were reported in the twenty-fourth quarterly.⁽⁷⁾ Briefly summarizing the results, at the fuel periphery the fuel grains appeared to be fractured and second phase material or voids were observed at subgrain boundaries. In the equiaxed grain growth region, voids and second phase particles were found in the grain boundaries and smaller voids and second phase particles were observed inside the grains. In the columnar grain growth region larger second phase particles were observed, but very little porosity was observed inside the grains.

Examination was performed of replicas of an unirradiated fuel sample prepared from the same fuel batch as that for E1E. Optical and electron micrographs of the fuel are shown in Figure 5-6. Photomicrographs of the control sample showed that the as-fabricated fuel consisted of equiaxed grains 20 to 40 microns in diameter. The fuel also contained grey and white second phase inclusions. An electron micrograph of second phase material observed at grain boundary triple points is shown in Figure 5-6. The phase stands out in relief indicating that it is either harder or more resistant to etching than the fuel matrix. An electron micrograph of the rosette structure visible in the optical photomicrograph is

also shown in Figure 5-6. It is hypothesized that the rosettes result from small composition gradients in the fuel, or from grain boundary migration.

Comparison of the grain size of the irradiated and unirradiated fuel confirmed that the grain size of the as-fabricated fuel (20–40 μ) was larger than that of the irradiated fuel near the periphery (1–2 μ). A similar effect has been observed by Padden⁽⁸⁾ in highly irradiated UO₂. Even more surprising was the fact that the grain size in the unirradiated fuel was greater than that observed in the equiaxed grain growth region of the irradiated fuel. Further work will be carried out to determine if the rosette structure observed in the unirradiated fuel is the result of plutonium or uranium gradients and to identify the second phase material observed in the grain boundaries.

5.4.4 Lattice Parameter Measurements of Irradiated Fuel

Samples of mixed oxide fuel were analyzed using the double crystal x-ray spectrometer at Battelle Northwest Laboratories. Six of the samples examined were irradiated fuel and four were unirradiated control samples. Of the irradiated fuel samples, five were irradiated in thermal flux (GETR) and one in fast flux (EBR-II). The irradiated fuel parameters are listed in Table 5-1. The variables being investigated are oxygen-to-metal ratio, burnup, and sodium-bonded versus gas-bonded fuel.

The lattice parameter calculations have not been completed, but the following qualitative conclusions have been reached:

- a. The mixed oxide fuel retained its crystalline structure to burnups of 280,000 MWd/Te.
- b. The lattice order, indicated by line broadening, of F2E, which was irradiated to 51,000 MWd/Te in a fast flux is

PHOTOMICROGRAPH OF UNIRRADIATED FUEL,
ETCHED 10 MINUTES WITH $10\text{HNO}_3 - 20\%\text{H}_2\text{O}_2 - 70\%\text{H}_2\text{O}$ (500X)

ELECTRON MICROGRAPH OF ROSETTE STRUCTURE
IN UNIRRADIATED FUEL (4500X)

(REDUCED TO 75% FOR REPRODUCTION)

ELECTRON MICROGRAPH OF SECOND PHASE
OBSERVED IN GRAIN BOUNDARY TRIPLE
POINTS OF UNIRRADIATED FUEL (12,000X)

ELECTRON MICROGRAPH OF OUTER PERIPHERY
OF FUEL IRRADIATED IN CAPSULE EIE (4500X)

**FIGURE 5-6. PHOTOMICROGRAPHS OF UNIRRADIATED AND IRRADIATED FUEL
FROM CAPSULE EIE**

TABLE 5-1
IRRADIATED MIXED OXIDE FUEL PARAMETERS

Sample	wt % PuO ₂	O/M	Burnup MWd/Te	Neutron Flux	Remarks
B8B-1	24.6	2.00	14,000	Thermal	Sodium bonded
B8B-2	24.6	1.93	14,000	Thermal	Sodium bonded
E2D-1	24.9	2.00	127,000	Thermal	Gas bonded
E2D-2	24.9	1.96	127,000	Thermal	Gas bonded
E2H	22.1	2.00	280,000	Thermal	Gas bonded
F2E	20.4	1.98	51,000	Fast	Gas bonded

approximately equivalent to that of E2H, which reached a burnup five times greater in a thermal flux.

- c. No second phases were detected in the sodium-bonded pins B8B-1 and B8B-2. The unit cell size of the B8B-1 fuel (O/M = 2.00) appears to have increased, suggesting a possible reduction in stoichiometry as a result of reaction with sodium. Calculation of the lattice parameters of the fuel samples will be completed and reported in the next progress report.

5.5 PHYSICAL PROPERTY STUDIES

5.5.1 Mechanical Properties of Ceramic Fuels

The creep and plasticity furnace was completed and tests initiated to checkout and calibrate the system. A drawing of the apparatus is shown in Figure 5-7.

Initial tests showed that the high thermal conductivity of the tungsten plungers, which are in direct contact with the test specimens, created an axial temperature gradient along the specimens. Sapphire and UO₂ discs were fabricated to insulate the plungers from the specimens and decrease the temperature gradient.

Testing of a 95% dense sample of UO₂ under a stress of 8800 psi at 1450°C was completed. A steady-state creep rate of 1.63 in./in./h was observed on a cylindrical pellet 0.482 inch in diameter and 0.746 inch long. Post-test measurements showed that the sample was barrel shaped. The diameter increase at the middle of the sample was about 0.010 inch.

Checkout will continue on UO₂ and Al₂O₃ samples to calibrate the apparatus and establish experimental procedures prior to installation of the apparatus in a plutonium glove box.

5.5.2 Melting Point of Irradiated Fuel

Installation and checkout of the melting point furnace in an alpha cell was completed. Figure 5-8 shows the equipment installed in the alpha cell. The furnace is of the Mendenhall V-filament type and can be operated with a gas atmosphere or in a vacuum. No problems were encountered in the remote operation of the furnace and the cell achieved "hot" status on January 5, 1968.

Measurements of the melting point of microcores of fuel ranging in burnup from 55,000 to 93,000 MWd/Te were completed. Results of the measurements are shown in Figure 5-9. Data for PuO₂ concentrations of

FIGURE 5-7. SCHEMATIC DRAWING OF MECHANICAL PROPERTIES TEST FURNACE

FRONT VIEW OF FURNACE THROUGH CELL WINDOW

SIDE VIEW OF FURNACE

FIGURE 5—8. FURNACE FOR MELTING POINT MEASUREMENTS OF IRRADIATED MIXED OXIDE FUEL

GEAP—5584

FIGURE 5-9. MELTING POINTS OF IRRADIATED FUEL

20 and 25 wt% are shown. The melting point of stoichiometric 20 wt% PuO_2 after irradiation was found to decrease 100°C at 55,000 MWd/Te. The 25 wt% samples showed a 50°C decrease in melting point from the unirradiated value at 85,000 MWd/Te. All measurements were carried out in cryogenic grade argon with an overpressure of 10 psi. Additional samples will be melted which will extend the burnup range to over 200,000 MWd/Te.

5.6 EXPERIMENTAL FUEL FABRICATION

5.6.1 Capsule E5B (High Burnup Comparison of Powder and Pellet Fuels)

Metallographic examination of samples of the E5B fuel and cladding is in progress. Neutron radiographs, gamma scans and profilometer measurements of these fuel pins were presented in the twenty-third and twenty-fourth quarterly reports. ^(6, 7)

The physical appearance of the fuel structures in each pin is very similar. In particular, the powder fuel shows no evidence of the as-fabricated fuel structure.

The structure of fuel bridges across the central void (observed in the neutron radiographs) is shown in Figure 5-10. Fuel having a similar structure was observed in pins from capsule E2C. ⁽⁶⁾ This material has a higher beta-gamma and lower alpha activity than the surrounding fuel, indicating it is highly enriched in fission products.

Examination of the fuel-cladding interface revealed that the cladding had been attacked. Transverse cross sections from the center of each fuel pin are shown in Figure 5-11. The reaction product at the fuel-cladding interface is visible in the pellet fuel.

High magnification (500 $\times$) photomicrographs of the fuel-cladding interface of each fuel pin are shown in Figure 5-11.

The effective thickness of the cladding of the pin containing powder fuel was decreased by from 0.001 to 0.0015 inch and the pellet fuel cladding 0.002 to 0.004 inch. Chemical and microprobe analyses of the samples have been initiated to identify the reaction products.

5.6.2 Capsules E5C, E5D and E5E (High Burnup, High Power, Reference Length Fuel Tests)

Capsules E5C, E5D and E5E are continuing their irradiation in GETR. All three capsules are operating at a power of 19.5 kW/ft. The average burnup as of January 30, 1968 was calculated to be 35,000 MWd/Te. All instrumentation in the capsules is operating normally.

5.6.3 Capsule E5F (Linear Fuel Movement)

Capsule E5F was inserted into GETR on January 20, 1968 at the start of reactor cycle 93. All thermocouples and the linear motion transducer inside the fuel pin are operating successfully. The capsule was cycled from a power of 7 to 17 kW/ft four times at the beginning of the reactor cycle and is presently operating at 10 ± 1 kW/ft. Analysis of the expansion data is in process. Data on the expansion of the fuel will be obtained as a function of burnup by cycling the fuel from low to high powers at burnup intervals of approximately 10,000 MWd/Te.

Drawings of the capsule and fuel pin are shown in Figures 5-12 and 5-13.

STRUCTURE OF FUEL IN BRIDGE

100X

STRUCTURE OF FUEL IN BRIDGE

500X

FIGURE 5-10. PHOTOMICROGRAPHS OF FUEL IN BRIDGES ACROSS CENTER VOID OF PELLET FUEL

FIGURE 5-11. COMPARISON OF POWDER AND PELLET FUEL IRRADIATED TO 120,000 MWd/Te IN CAPSULE E5B

AS BUILT

10. Nicks or scratches deeper than .002 inches or 10% of wall thickness shall not be allowed.
11. After spiral wrapping wire (P15) test with gage .332 ± .001 I.D. by 3" long.
12. Weld number in diamond ~~XXXX~~ indicates weld identification only.
13. Radiograph fuel pin orientation full length in duplicate. This radiograph to be made in vertical position, with bellows end up.
14. Fuel pellets to be loaded through end opposite the transducer. P15 spiral wire shall be clamped at that end until and placed next weld is made and locked test. P15 will then be welded to the end of the fuel pin and the fuel pin of P15 another. Final split test shall be performed.
15. All welds to be leak checked with Helium mass spec. A leak greater than 2×10^{-6} cc/sec. S.T.P. shall not be acceptable.
16. All radiographs to be reviewed by responsible engineer.
17. All weld penetrations to be reviewed by responsible engineer.
18. All fusion welds shall have at least 90% of penetration.
19. All stainless steel parts to be cleaned per spec. #PM-CPY1 and passivated per spec. #PM-YP1.
20. White glove cleanings shall be maintained following cleaning and passivation.

NOTES:

FIGURE 5-13. FCR CAPSULE E5F-FUEL PIN

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 flux spectra. Important fuel performance characteristics, such as fission gas release, fuel cladding interactions, and fuel thermal performance, must be 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 have been conducted so far only with encapsulated specimens and recently unencapsulated bare pins have been built

and tentatively been approved for testing. It is planned to increase the proportion 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

The removal of failed pin F2F (S/A XG06) from its capsule was described in the previous quarterly report.⁽⁷⁾ Features of the pin, which reached 59,000 MWd/Te, are summarized in Table 6-2 of that report. Examination of the pin indicated that the rupture essentially followed a spiral path along the spacer wire over a 16-inch length, extending above and below the 14.21-inch fuel stack. This is shown in Figure 6-1. Longitudinal views in four angular positions are shown in the upper view, while the lower view shows a composite of all four mounted in single planar view. Gross gamma scan of this pin (Figure 6-1) shows a disrupted fuel column with peaks at positions approximately 5-1/2, 6, and 11 inches from the pin bottom. These peaks apparently correspond to nuggets of solid fission products, observed in the pinhole camera gamma radiograph.

Profilometer traces were taken where possible, and show substantial diametral increases of up to 53 mils. A diametral increase of 2—2.5 mils was recorded over the blanket region.

A metallographic specimen positioned 6-1/4 inches from the bottom of the rod is

reproduced in Figure 6-2. Significant features are listed below:

- a. A center void and the presence of a solid fission product agglomerate in the pellet center, previously observed in the pinhole gamma radiograph, imply high temperatures. This is also concluded from temperature calculations which indicate a radius of equiaxed grain growth equivalent to a gap conductance of 1500—1600 Btu/h-ft²-°F, typical for gas-bonded pins.
- b. A sodium-fuel reaction phase 4—5 mils thick, which is α -active, appears at the fuel-cladding interface. Considerable intergranular fuel attack took place with more attack noted at the fuel perimeter than at the center. Grain boundary attack is also more prominent in the outer region than in the center, possibly because grain boundaries occupy a larger area.
- c. The fuel expanded 28 mils in the X-plane of Figure 6-2. The 28-mil diametral increase does not necessarily represent ductile cladding strain, but rather a measure of fuel expansion after failure, with the expanded cladding "hinged" at the observable crack at the bottom. An estimate of the maximum Incoloy clad strain at failure, obtained by measuring the cladding perimeter, amounts to 2%.
- d. Cladding separations appear as brittle, intergranular cracks as shown by oxalic acid etching. No apparent clad grain growth is noted, so that it may be assumed that the cladding temperatures did not exceed 1850°—1900°F for any significant period of time.

A possible explanation of the spiral rupture configuration could be the upward motion of a gas bubble along the spacer wire,

as would be the case for a gas bubble in the sodium bond or a fission gas bubble released at the time of failure, with resultant gas blanketing and localized overheating.

A computer calculation was performed in which cladding temperatures were calculated assuming a sodium void over a 60-degree arc, in keeping with the extent of gross deformation indicated by the metallographic view of Figure 6-2. Using input characteristics of F2F operation, an average cladding temperature of 1950°F was calculated at the blanketed location, while temperatures of adjacent sections declined to 950°F.

Subassembly XG05, containing nine Group F2 encapsulated fuel pins, was removed from the EBR-II core during a refueling shutdown in December when Xe-131 activity was detected in the cover gas. Two suspect subassemblies, XG05 and XG08, were transferred from the core to the S/A storage rack. Both subassemblies will remain out of the core during the initial full power operation after start-up. It is planned to detect the leaker by a process of elimination.

6.3 LOW POWER FUEL SWELLING IRRADIATION

Groups 0, 6, and 8A are continuing their irradiation in EBR-II and their current status is presented in Table 2-4.

6.4 HIGH POWER, INTERMEDIATE BURNUP IRRADIATION

The Group F4 capsules from S/A XO11, which was removed from the reactor last summer, are still awaiting incorporation into a new subassembly for reinsertion into the reactor. Peak burnup in this cluster is estimated to be 35,300 MWd/Te, with a target of 50,000 MWd/Te. The previous quarterly report⁽⁷⁾ noted the failure of the Incoloy 800 clad fuel pin F4F. This pin

FIGURE 6-1. GAMMA SCAN, GAMMA RADIOGRAPH, AND PHOTOGRAPHS OF CAPSULE F2F

FIGURE 6-2. CROSS SECTION OF F2F AT PEAK FLUX POSITION

GEAP-5504

TABLE 6-1

CHARACTERISTICS OF FAILED TASK F PINS

Pin	Cladding	wt % PuO ₂	O/M	Density, Smeared	Gap mils	Fast Flux Exposure >1.0 MeV	Heat Rating kW/ft	Burnup $\times 10^3$ MWd/Te	Estimated Temperature °C ⁽¹⁾		Remarks
									Surface	Center	
F2F	Modified Incoloy 800	20	1.98	94.8%	0.4—5.6	1.1×10^{22}	16.8	59	880	2715	Failure indicated by neutron radiograph and gamma scan.
F4F	Incoloy 800	30	2.00	86.3%	2.1—4.4	0.7×10^{22}	17.3	34.2	980	2855	Failure indicated by neutron radiograph and gamma scan.
F4D	Incoloy 800	20	2.00	86.8%	1.8—4.5	0.7×10^{22}	17.2	34.0	910	2785	Failure indicated only by presence of fission gas within capsule. Failure not indicated by neutron radiograph or gamma scan.

(1) Assuming no sodium is present.

was removed from S/A XO11 along with the apparently unfailed, Incoloy 800 clad pin F4D, which had similar design and operating parameters. Both of these specimens have cladding exposures of 0.7×10^{22} nvt (>1 MeV) and cladding surface temperatures of 950°F . Significant parameters of these pins are noted in Table 6-1. Neutron radiographs of these capsules, along with capsule F2F, are shown in Figure 6-3. Gamma scans of F4D and F4F are shown in Figures 6-4 and 6-5, respectively. The concentration of fuel in the bottom of capsule F4F (Figure 6-3) appears as a peak in the gross gamma scan (Figure 6-5).

6.4.1 Capsule F4D

Puncturing the capsule prior to removing the pin for examination revealed the presence of fission gases, despite the absence of any indications of failure in the radiographs or gamma scans. Subsequent puncturing of the pin showed a high concentration of fission gases, at a pressure that was about three times that of the gases in the capsule. A total gas release rate of 51% was measured for this pin. This value is consistent with previous gas release values for fast flux tests. The location of the pin failure has not yet been determined. The inability, at room temperature, to detect a leak in the pin suggests that the failure is quite small.

The metallographic examination of the outer containment capsule cladding at three locations has been completed. It was found that no microstructural or dimensional degradation of the tubing had occurred as a result of the EBR-II irradiation and subsequent handling. A recommendation was made that other capsules originally in bundle XO11 be reinserted in EBR-II for further irradiation based on these observations. Typical views of the sensitized cladding are shown in Figure 6-6.

Profilometer measurements over the capsule showed no significant dimensional changes. Profilometer traces of the fuel pin itself show a maximum diametral increase of 0.8 mil over the fuel region, with an average increase of 0.7 mil. These measurements are consistent with ΔD values of the F2 pins at 50,000 MWd/Te. An average increase of 0.4 mil was measured over the blanket region.

6.4.2 Capsule F4F

Helium leak testing of this capsule has not indicated any evidence of a rupture in the capsule body. Profilometer traces of the capsule indicate a diametral increase of 1.75 mils at a position 7 inches from the bottom of the fuel column. Diametral increases to about 1 mil were also measured at a location about 9 inches from the fuel bottom. The former position is at the approximate center of the fuel column and corresponds to an intense peak of gamma emitting fission products. This diametral increase represents a condition of ovality, as it is only present in the 0/180 degree plane. A possible explanation of this configuration could be uniform swelling around the pin circumference, with the wire wrap causing the additional, localized diametral increase leading to out-of-roundness. This is the condition of the transverse metallographic specimen of specimen F2F, Figure 6-2. A diametral increase to 1 mil was recorded in the 90/270 degree plane along a 3.5-inch length above the previously mentioned midheight position. No diametral increase was recorded in this plane at the midheight position.

6.5 ELEVATED CLADDING TEMPERATURE TESTS (GROUP F5)

Fuel pin fabrication is continuing, with ten pins completed and ready for encapsulation. The fuel is prepared for the remaining six pins. The fuel and cladding parameters for these pins is presented in Table 6-2.

TABLE 6-2

TASK F GROUP F5 FUEL AND CLADDING PARAMETERS

Group	Designation	Pellet Density (%)	Fuel-Cladding Gap (mils)	Composition		Fuel Powder Fabrication Process	O/M Ratio	Cladding	Fuel	Blanket
				$\frac{\text{Pu}}{\text{U+Pu}}$ (%)	U-235 (%)					
F5	F 5A03	94±1.00	3 +2.00 -0.00	25	92-93	Co-ppt	1.960-1.980	Incoloy 800	SP	UO ₂ (natural)
	F 5B03		3 +2.00 -0.00					Incoloy 800		
	F 5C07		7 ± 1.00					Incoloy 800		
	F 5D05	94±1.00	5 ± 2.00			Co-ppt		Incoloy 800	SP	
	F 5E00	84±1.00**	0			P-mix***		Incoloy 800	VCP	
	F 5F63	94±1.00	3 +2.00 -0.00			Co-ppt		316	SP	
	F 5G63		3 +2.00 -0.00					316		
	F 5H67		7 ± 1.00					316		
	F 5I65	94±1.00	5 ± 2.00			Co-ppt		316	SP	
	F 5J60	84±1.00**	0			P-mix***		316	VCP	
	F 5K45	94±1.00	5 ± 2.00			Co-ppt		304	SP	
	F 5L45							304		
	F 5M15							321		
	F 5N15							321		
	F 5O75							347		
F5	F 5P75	94±1.00	5 ± 2.00	25	92-93	Co-ppt	1.960-1.980	347	SP	UO ₂ *

*Upper blanket contains two monitoring pellets—one depleted UO₂ and one fully enriched UO₂.
Positions will be specified on drawings.

**Powder (Dyna-Packed).

***P-mix designates physically mixed UO₂ and PuO₂ powder was used to make pellets.

GEAP-5584

FIGURE 6—3. NEUTRON RADIOGRAPHS OF CAPSULES F2F, F4D, AND F4F

FIGURE 6-4. GROSS GAMMA SCAN OF F4D

FIGURE 6-5. GROSS GAMMA SCAN OF F4F

OXALIC ACID ETCH
TYPICAL CLADDING

NEG. No. 9787-06
200X

OXALIC ACID ETCH
WELD ZONE

NEG. No. 9787-05
200X

FIGURE 6-6. F4D CAPSULE CLADDING — LOWER CAPSULE REGION

The selection of a satisfactory EBR-II subassembly design for this test is yet to be resolved. Flowtests will be required to resolve a potential mixing problem between hot and cold sodium in the proposed bypass flow modification to the 19-capsule Mark A subassembly. Should the mixing be judged inadequate, flow mixers will be incorporated into the subassembly and tested. Hydraulic analysis and conceptual design of several staggered tube systems to promote mixing of the coolant has been initiated (at GE).

6.6 BARE PIN IRRADIATION

6.6.1 Group 3A (High Power)

These sixteen 0.290-inch outside diameter fuel pins are at the EBR-II site. They have been inspected and are physically acceptable for irradiation testing in EBR-II. ANL is preparing additional hazards analysis requirements for unencapsulated pin

irradiations, which are a prerequisite to their insertion in the EBR-II. In the meantime, a hazards analysis is being prepared based on the draft criteria and the results of a meeting at EBR-II site on January 25.

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

Fabrication of the unencapsulated fuel pins (0.250-inch outside diameter) for these two 37-pin subassemblies is continuing. The modified EBR-II Mark B-37 subassembly design has been accepted for this test. Preparations are under way at ANL to fabricate and flowtest the design.

6.6.3 Pre-Irradiation Transient Tests

Two of the Group 7 capsules are currently operating in the EBR-II at 12 to 15 kW/ft and have accumulated 27,000 MWd/Te burnup. The status of the remaining four fuel pins in this group is reported under Task C.

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 requires integration of results of several disciplines, including core neutronics, thermal hydraulics, and safety and economic analysis. To accomplish this, improved analytical methods and computer programs are being developed and used in analytical studies of fast reactor neutronics, safety characteristics, and both analytical and experimental thermal hydraulics. These studies, together

with a continual evaluation and surveillance of the results of experimental programs, form a basis for establishing current design and safety criteria. These criteria, in turn, define the range of reactor characteristics and test parameters upon which the experimental programs must focus.

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

ENDF/B tapes 114, 115, 116 and 117, containing category 1 materials, have been received from Brookhaven National Laboratory. Copies were made for use on the GE-635 computer.

Computations of self-shielding factors from the ENDF/B smooth data for iron indicate that there is insufficiently fine detail of cross sections in the range from 500 keV to 1 MeV. To test the need for more detail, the ENDF/B iron data were modified to be consistent with the more detailed curves recommended by Schmidt.⁽⁹⁾ These modified data were then averaged with the ENDRUN code and showed a definite increase in the degree of self-shielding in the energy range from 400 to 800 keV, although less self-shielding than obtained with the Russian cross-section set.⁽¹⁰⁾ Chromium data on ENDF/B were found to have a similar lack of detail, and it was recommended and approved at the January meeting of the Cross Section Evaluation Working Group, that alternative ENDF/B iron and chromium information be issued which has increased detail over this energy range.

7.2.2 Code Conversion and Development

The FCC-IV code, a zero-dimensional diffusion theory code which uses cross sections in the Russian format, has been converted to the GE-635 computer. Means for expanding it to handle 70 energy groups and

50 downscattering groups have been considered. Since this full expansion would require extensive modifications, changes will be limited to a more moderate expansion (~50 groups and 20 downscattering groups) until such time that the multigroup cross section adjusting study may indicate that more groups are necessary.

Small service routines for ENDF/B data tapes have been written for the GE-635 computer. These routines will create, modify, and read packed binary tapes, each of which has a library listing of the ENDF/B materials contained on the tape. These routines will permit storage of the complete ENDF/B data on a single tape, with rapid access to the data of any material.

7.2.3 Analysis of Critical Experiments

The effects of adjusting cross sections for Pu-239 and U-238 over several energy ranges have been studied by recomputing some of the quantities measured in ZPR-III, Assembly 48. Higher α values (by 50 to 110%) for Pu-239 between 600 eV and 15 keV were indicated in early 1967 by the analysis of Sowerby in the U.K., using high precision $\sigma_t(E)$ and $\sigma_f(E)$ measurements. Several papers^(11, 12, 13) at the recent IAEA meeting on Fast Reactor Physics and Related Safety Problems, in Karlsruhe, Germany, tend to substantiate higher values, with the data of Schomberg, et al., being considerably higher than the computer values of Sowerby (see Figure 7-1). In addition, measurements made at Karlsruhe by Pönitz, et al.,⁽¹⁴⁾ indicate lower U-238 σ_c values from 100 to 500 keV.

The higher α values for Pu-239 of Sowerby and the lower U-238 σ_c data of Pönitz were considered, along with 10 to 15% higher σ_f values for Pu-239 between 20 and 100 keV (where ENDF/B values are perhaps too low), as modifications to the present ENDF/B data for these materials. The

FIGURE 7-1. COMPARISON OF Pu-239 ALPHA VALUES FOR BAS02 AND BAS03 WITH U.K. DIFFERENTIAL MEASUREMENTS AND SWEDISH VALUES

spherical model of Assembly 48 suggested for use in testing ENDF/B data⁽¹⁵⁾ was used in the calculation of the computed quantities. While ENDF/B data, used directly, under-computes k_{eff} by about 1%, using the combination of one-half the Sowerby increase in Pu-239 α (an increase of 30 to 60%), the U-238 data of Pönitz and the higher Pu-239 σ_f , brings the k_{eff} agreement to within 0.5%. This combination also generally improves the agreement with experiment for reaction rate ratios and central worths of the more important fast reactor materials. While these results do not provide conclusive evidence for determining the correct values of the α for Pu-239, they do indicate that, with reasonable modifications in the other data used, substantially higher α values can be incorporated into the data sets without producing prohibitively poor agreement between calculations and experiment.

The indications described above, for a substantially higher Pu-239 α value between 600 eV and 20 keV than now used in the BAS02 file, and the results of these studies on Assembly 48 have resulted in the decision to modify the basic data being used at GE for fast reactor computations. These modifications are described in subsection 7.2.4, Cross Section Files.

7.2.4 Cross Section Files

Modifications are being made to the GE 60-group cross section data set BAS02 to incorporate the results from new data and recent data evaluations. The new file will be titled BAS03, and will serve as the standard for fast reactor computations.

The principal departures from BAS02 involve the Pu-239 capture cross section between 0.6 and 20 keV, the Pu-239 fission cross section from 0.6 to 150 keV, and the U-238 capture cross section from 100 keV to 2 MeV. Essentially the same

changes resulted in good agreement between calculations and measurement for ZPR-III, Assembly 48 (see subsection 7.2.3, Analysis of Critical Experiments).

Figures 7-1 through 7-3 show the data changes in graphical form, with Figure 7-1 indicating the increase in $\alpha(E)$ of Pu-239 which results from the changes in capture and fission cross sections between 0.6 and 20 keV. Because relatively accurate measurements of the total cross sections have been made in this energy range, the higher α values were achieved by a combination of an increase in σ_c and decrease in σ_f . The open crosses and circles in Figure 7-1 are differential values reported by the U.K.⁽¹¹⁾ Both sets have the possibility of large errors. Systematic errors, particularly in the liquid scintillator measurements, may mean that these values are much too high. The crosses with circles are values used in fast reactor calculations in Sweden,⁽¹²⁾ as of November 1967. BAS03 values are slightly larger than the Swedish values down to 1 keV and remain high to 600 eV, whereas the Swedish values drop at 1 keV.

The increase in σ_f of Pu-239 shown in Figure 7-2 is indicated by critical experiments, as well as by differential data. The many measurements of σ_f in this range are too numerous to include in Figure 7-2, but the BAS03 values essentially follow the measurements of White, et al.,⁽¹⁶⁾ except for the dip between 20 and 150 keV, which White observed, and was included in the BAS02 values. Figure 7-3 shows the modification in σ_c of U-238. The primary change results from the lower values reported by Pönitz, et al.,⁽¹⁴⁾ above 100 keV.

Tables 7-1 and 7-2 list the correction factors, for the standard GE 60-group structure, which are applied to the BAS02 file cross sections to arrive at the BAS03 values.

FIGURE 7-2. Pu-239 σ_f (E) VALUES USED IN BAS02 AND BAS03

FIGURE 7-3. U-238 σ_c (E) VALUES USED IN BAS02 AND BAS03

TABLE 7-1

PU-239 CORRECTION FACTORS REPRESENTING
CHANGES FROM BAS02 TO BAS03 CROSS SECTION FILE

Group	Group Lower Energy (keV)	Correction Factor	
		Fission	Capture
1—8	170	Unchanged	Unchanged
9	150	1.005	↑
10	140	1.011	
11	120	1.016	
12	110	1.020	
13	100	1.025	
14	91	1.029	
15	82	1.034	
16	74	1.045	
17	67	1.048	
18	61	1.057	
19	55	1.066	
20	50	1.073	
21	45	1.078	
22	41	1.089	
23	37	1.083	
24	33	1.075	
25	30	1.054	
26	27	1.008	
27—28	21	Unchanged	Unchanged
29	18	Unchanged	1.01
30	15	0.99	1.06
31	12	0.975	1.14
32	11	0.96	1.23
33	9.1	0.945	1.32
34	7.7	0.985	1.47
35	6.4	0.94	1.46
36	5.5	0.915	1.47
37	4.6	0.93	1.53
38	3.9	0.93	1.48
39	3.3	0.935	1.435
40	3.0	0.955	1.44
41	2.7	0.975	1.46
42	2.5	0.985	1.47
43	2.2	0.98	1.47
44	2.0	0.965	1.45
45	1.6	0.94	1.41
46	1.3	0.91	1.35
47	1.0	0.895	1.30
48	0.8	0.92	1.36
49	0.63	0.935	1.42
50	0.50	0.995	1.08
51—60	Thermal	Unchanged	Unchanged

TABLE 7-2
U-238 CORRECTION FACTORS REPRESENTING
CHANGES FROM BAS02 TO BAS03 CROSS SECTION FILE

<u>Group</u>	<u>Group Lower Energy (keV)</u>	<u>Correction Factor for Capture</u>
1—2	2200	Unchanged
3	1350	0.83
4	825	0.94
5	500	0.99
6	300	0.97
7	180	0.92
8	170	0.89
9	150	0.90
10	140	0.91
11	120	0.94
12	110	0.96
13	100	0.97
14	91	0.98
15	82	0.99
16—60	Thermal	Unchanged

7.2.5 Multigroup Cross Section Adjusting

Good progress is continuing on the nuclear data processing program ENDRUN. The routines for averaging ENDF/B smooth data and for creating scattering matrices are running in the linked form. Averaging routines for resolved and unresolved data are in the checkout stage.

7.2.6 Fast Reactor Sensitivity to Cross Section Uncertainties

The sensitivity of fuel costs to uncertainties in nuclear data has been computed for two LMFBR designs, one representative of an advanced design with low core leakage and the other of an early design with high core leakage, and containing BeO. Uncertainty limits and energy ranges have been chosen from a number of data evaluations, ^(9, 17, 18) as well as from basic data measurements, to provide from 75 to 95% expectancy between the limits and with consideration for error correlation in the data uncertainties.

The primary results are shown in Table 7-3. It is seen that the total uncertainty range for the advanced reactor is

0.25 mill/kWh, while that of the early design is 30% greater, because of the safety requirements of this design and the subsequent changes required in BeO volume fraction. For both reactors, the fuel cost uncertainty is dominated by the uncertainty in the α for Pu-239 below 15 keV. In all cases, the fuel cost uncertainties are primarily caused by changes in plutonium credit and inventory costs. Changes in fission cross section had the greatest effect on inventory costs, while changes in parasitic and U-238 capture data affected the credit costs nearly exclusively.

These results have been used to estimate what nuclear data accuracies are required to reduce the combined fuel cost uncertainty of the advanced reactor to ± 0.03 mill/kWh. Foremost among these requirements are a 3 to 5% accuracy for Pu-239 α , a 2% accuracy for Pu-239 σ_f , a 0.5% accuracy for Pu-239 $\bar{\nu}$, and a 2 to 5% accuracy for U-238 σ_c .

The results of this study were presented in a paper at the Winter Meeting of the American Nuclear Society, Chicago, Illinois, November 5—9, 1967, ⁽¹⁹⁾ and will be published in the May or June 1968 issue of Nuclear Applications.

TABLE 7-3
EFFECTS OF DATA UNCERTAINTIES ON LMFBR FUEL COSTS

	Fuel Cost Ranges (mill/kWh)	
	High Leakage Reactor	Low Leakage Reactor
Base Values	0.70	0.50
Data Uncertainty		
Pu-239 α (0.2 - 15 keV)	0.66 to 0.90	0.48 to 0.62
Pu-239 σ_f (15 - 300 keV)	0.68 to 0.70	0.46 to 0.50
Pu-239 $\bar{\nu}$ (0 - 10 MeV)	0.65 to 0.78	0.45 to 0.58
U-238 σ_c (1 - 100 keV)	0.65 to 0.75	0.45 to 0.55
Fission Products (0 - 10 MeV)	0.66 to 0.76	0.48 to 0.54
All Data Uncertainties	0.61 to 0.94	0.41 to 0.66

7.3 CORE ANALYSIS

7.3.1 Two-Dimensional Neutron Diffusion Calculations

Several of the 13-group BISYN⁽²⁰⁾ calculations reported in Reference (7) have been analyzed for their accuracy in calculating power distribution. In each BISYN calculation on core 24C of Reference (6), a different combination of initial and shape function solutions were used. The 13-group comparison calculation was made using CRAM,⁽²¹⁾ a two-dimensional diffusion code.

The results show a very satisfactory low sensitivity to variations in the details of the BISYN solution. For this particular reactor model, it can be concluded that only one initial solution (through the core in the radial direction) should be taken and shape function solutions need be calculated only in the reactor regions with significantly different compositions. These conclusions are consistent with those made in Reference (7) where only k_{eff} was considered.

The reactor configuration considered is shown in Figure 7-4, and Table 7-4 gives the combinations of initial and shape function solutions considered for this analysis.

In the first four of these, the initial solution was taken in the radial direction; in the fifth, the initial solution was taken in the axial direction.

In Table 7-5, it is seen that Case 3 with one radial initial solution through the core region and five shape solutions (four of them in core regions) gives the closest agreement to CRAM for the peak-to-average core power, peak power in Core Zone 1 to peak power in Core Zone 2 ratio, and fraction of power produced in the core zones. Using two fewer shape solutions in the core zones (Case 2) increases the error only slightly. This indicates that if two adjacent core regions have nearly the same composition, only one shape solution is necessary for them. A comparison of Cases 1 and 2 shows that a shape function should be calculated in the radial blanket region. From Case 4, it is evident that for this reactor model, initial solutions should not be made in the axial blanket regions. Case 5, for which the solutions were run transverse to the other cases, does not give as good results as obtained for Cases 2 and 3.

**FIGURE 7-4. FCR REACTOR ZONES FOR BISYN VERSUS CRAM CALCULATIONS—
POWER DISTRIBUTION**

TABLE 7-4

BISYN CALCULATIONAL PROCEDURES—POWER DISTRIBUTION

Case Number	Direction of Initial Solution	Solutions Used (See Figure 7-4.)									
		Initial			Shape						
		1	2	3	1	2	3	4	5		
1	Radial	X			X	X					
2	Radial	X			X		X		X		
3	Radial	X			X	X	X	X	X		
4	Radial	X	X		X	X	X	X	X		
5	Axial	X		X	X	X					

TABLE 7-5

BISYN VERSUS CRAM RESULTS—POWER DISTRIBUTION

ZONE	CRAM	BISYN CASE				
		1	2	3	4	5
	<u>POWER FRACTION</u>					
1	0.4815	0.4739	0.4785	0.4790	0.4898	0.4851
2	0.3791	0.3811	0.3820	0.3812	0.3671	0.3767
3	0.0186	0.0190	0.0159	0.0159	0.0176	0.0189
4	0.0023	0.0025	0.0025	0.0025	0.0024	0.0033
5	0.0285	0.0283	0.0286	0.0286	0.0286	0.0246
6	0.0152	0.0151	0.0151	0.0151	0.0158	0.0149
7	0.0032	0.0040	0.0047	0.0048	0.0038	0.0044
8	0.0126	0.0127	0.0108	0.0108	0.0119	0.0136
9	0.0017	0.0021	0.0018	0.0018	0.0017	0.0035
10	0.0027	0.0049	0.0036	0.0036	0.0031	0.0048
11	0.0367	0.0381	0.0385	0.0382	0.0375	0.0316
12	0.0178	0.0181	0.0180	0.0182	0.0204	0.0186
	<u>PEAK POWER TO CORE AVERAGE POWER RATIO</u>					
	1.436	1.427	1.435	1.435	1.493	1.456
	<u>CORE ZONE 1 TO CORE ZONE 2 PEAK POWER RATIO</u>					
	1.0257	0.9998	1.0042	1.0044	1.0966	1.0434

Figures 7-5 and 7-6 show power distributions from CRAM and BISYN Cases 3 and 4 along the radial and axial midplanes, respectively. Note the excellent agreement between CRAM and BISYN Case 3.

Thirteen-group BISYN sodium void reactivity calculations were also made to determine the best regions through which initial and shape function solutions should pass in a problem representing a partially voided core. Again, the 13-group comparison cal-

culation was made using CRAM. The results indicate that, for this partial sodium void calculation, excellent agreement is obtained for several combinations of initial and shape function solutions.

The combinations of initial and shape function solutions considered for this analysis are given in Table 7-6; Cases 1 and 2 are based on the reactor configuration in Figure 7-4. Cases 3-6 are based on the full core reactor model shown in Figure 7-7.

TABLE 7-6
BISYN CALCULATIONAL PROCEDURES—SODIUM VOID

Case	Problem	Solutions Used				
		Initial		Shape		
		1	2	1	2	3
1	Na In	X		X	X	
2	Na Out	X		X	X	
3	Na Void	X			X	X
4	Na Void		X	X	X	
5	Na Void		X	X	X	X
6	Na Void	X	X	X	X	X

Two considerations were present in the choice of a representative partial sodium void. First, it had to be a typical void; that is, one which could be expected to occur in a real situation. Second, the void had to be of the size and in the location that would best test BISYN's capability to handle the increased axial leakage and spectrum hardening. The sodium void pattern selected was nonsymmetric about the radial midplane. Its size corresponds to voiding in the first 10 centrally-located fuel bundles.

From Table 7-7, it is seen that taking the void into account in all the solutions gives the closest agreement to the CRAM reactivity result. Finally, for total sodium loss, the BISYN and CRAM reactivity calculations are in excellent agreement.

The computer code EXTERMINATOR-2⁽²²⁾ has been placed in operation on the GE-635. Two of the cases above were recalculated with EXTERMINATOR-2. As shown in

FIGURE 7-5. AXIAL POWER DENSITY DISTRIBUTION — BISYN VERSUS CRAM

FIGURE 7-6. RADIAL POWER DENSITY DISTRIBUTION—BISYN VERSUS CRAM

FIGURE 7-7. FCR REACTOR ZONES FOR BISYN VERSUS CRAM CALCULATIONS - SODIUM VOID (CROSS-HATCHED REGION)

TABLE 7-7
BISYN VERSUS CRAM VERSUS EXTERMINATOR-2

Total Sodium Loss and Sodium Void Reactivities

<u>Case</u>	<u>Description</u>	<u>k_{eff}</u>	<u>Δk</u>
CRAM-1	Na In	0.99569	-
CRAM-2	Na Out	1.00256	0.00687
CRAM-3	Na Void	0.99749	0.00180
BISYN-1	Na In	0.99308	-
BISYN-2	Na Out	0.99996	0.00688
BISYN-3	Na Void	0.99504	0.00172
BISYN-4	Na Void	0.99533	0.00201
BISYN-5	Na Void	0.99515	0.00183
BISYN-6	Na Void	0.99509	0.00177
EXTERMINATOR-1	Na In	0.99294	-
EXTERMINATOR-2	Na Out	0.99987	0.00693

Table 7-7, the EXTERMINATOR-2 k_{eff} calculation is in excellent agreement with the BISYN results, but is low compared to CRAM.

7.3.2 Control Rod Calculations

A series of rod worth calculations was made on 3-inch diameter, ZrH_2 - B_4C flux-trap design control rods, using diffusion and S_8 discrete-ordinates transport theory. The model studied was a three-region cylindrical cell:

- a. Central region of moderator (ZrH_2)
- b. Annulus of absorbing material (B_4C , natural B)
- c. Annulus of typical sodium-cooled FCR core composition.

The cell outer diameter was calculated so that the control region was 0.67% of the total cell volume. The radius of the moderator region was varied to provide a range of moderator-to-absorber volume ratios from 0.0 (all B_4C rod) to 6.0.

All calculations were made in 13 groups using the transport theory code ANISN. The diffusion option of ANISN was used in the diffusion theory calculations. Anisotropic scattering was taken into account for all materials only through the use of a transport cross section.

Control rod worth was computed to be the difference between k_{eff} of the cell when only sodium was present in the two innermost regions and k_{eff} when the control materials were present. Diffusion and S_8 transport theory calculated the same k_{eff} when only sodium was present.

The results of the calculations are shown in Figure 7-8. Diffusion theory consistently overestimates control worth; the error reaches a maximum of ~13.4% at $V(ZrH_2)/V(B_4C) = 3.0$ and is ~11.5% at the point of peak control rod worth [$V(ZrH_2)/V(B_4C) = 1.0$]. Diffusion theory overestimates the all B_4C control rod worth by ~5.1%.

**FIGURE 7-8. CONTROL ROD WORTH (% Δk) OF FLUX-TRAP
NATURAL B_4C CONTROL RODS (DIAMETER = 3.0 inches)**

7.4 TRANSIENT AND SAFETY ANALYSIS

7.4.1 Space Dependent Reactor Kinetics

Programming modifications to the WIGL2 computer program⁽²³⁾ were completed which will allow the user to input tables of reactivity versus time and adjust the appropriate nuclear parameters. Region-dependent Doppler coefficients may also be used with modification of the group 2 absorption cross section to obtain the desired reactivity effect.

7.4.2 Core Meltdown Calculations

Calculation of excursions leading to core disassembly in a "typical" 1000 MWe LMFBR using the 2-D reactor excursion code MARS⁽²⁴⁾ gave good agreement for total energy release with previous 1-D calculations. Coding modifications were suggested to improve the calculational model:

- a. Provide a means of explicitly including the heat of fusion in excursions starting below fuel melting.
- b. Provide a means of changing the heat capacity temperature dependence at the melting point (tabular format).
- c. Eliminate pressure feedback below the fuel melting temperature.

7.4.3 FORE System

The following errors have been found and corrected in FORE II, and a new tape has been entered in the program library.

- a. Incorrect heat balance on the inner fuel ring.
- b. Incorrect calculation of fuel conductivity for the inner fuel ring in steady state.
- c. Temperature dependent feedbacks did not use the last calculated values of temperature.
- d. Calculation of the Doppler feedback was incorrectly coded for a one-channel calculation.

- e. The code would not calculate density feedback for the coolant and structure if additional material were not present in the core.

A new time step procedure for more accurate and efficient handling of the time step has been developed and checked out. Also, a new model for the fuel center temperature has been formulated and is being tested.

Studies of low flow transients from low power conditions have yielded instabilities in the transient cladding and coolant temperature profiles, and inaccuracy in the axial coolant temperature distribution. The numerics of these problems are now under consideration in an effort to find a more reliable method of computation.

7.4.4 Fuel Failure Propagation Study

A study of accidents in a fuel bundle that may possibly cause propagation to adjacent subassemblies has been undertaken. The initial case examined is that of a fuel pin, on the outer edge of a bundle, bowed against the channel wall.

To establish a reference case in the undisturbed geometry, the temperature distribution for an outer row fuel pin was calculated. The significant effects found were development of the thermal coolant profile within 3 to 4 inches from the bottom of the core and large circumferential gradients in the cladding caused by nonuniform cooling of the pin near the channel wall.

The analysis of the pin against the channel wall indicated that the additional rise in coolant temperature in the adjacent bundle would be on the order of 20°F per foot of core length when the pins were run at 20 kW/ft. This means that if the pin contacts the wall along its entire length, a 24-inch core would experience only a 40°F increase in average core ΔT in the adjacent channel.

Also, the results indicate that the bowed pin would probably receive sufficient cooling to avoid sodium boiling or cladding melting.

7.5 THERMAL HYDRAULICS AND FUEL ANALYSIS

7.5.1 Hydraulic Flow Tests

All of the components for the "Velocity Profile Tests" (semi-infinite array) have been completed. These components are now being stored until space in the test loop becomes available in early March. A 6-week test period will be required to complete this series of tests.

The velocity profile tests will use water to simulate sodium in a six-scale mockup of a portion of an LMFBR fuel bundle. Velocity profiles will be measured for pitch-diameter ratios between 1.1 and 1.3 and for equivalent velocities of 10 to 30 ft/sec.

A program plan for hydraulic flow testing was prepared and submitted.

7.5.2 Flow Induced Vibrations

As reported in the previous quarterly report (GEAP-5541), a brief analysis of flow induced vibration characteristics of LMFBR fuel pins has been completed. The design curves for vibrational stability presented allow one to estimate the required spacing of both grid and wire wrap spacers to prevent vibration problems (Figures 7-12 through 7-15). The results contained in these figures are conservative. However, they should be considered as the maximum permissible spacings.

7.5.2.1 Assumptions

- a. The natural frequency may be determined by using a single-span, simply-supported model.
- b. The virtual mass of the coolant may be based on the coolant volume in an annulus of width equal to one-half the fuel pin spacing.

- c. The coolant is assumed to be sodium at temperatures of 1000°F.
- d. The path of coolant flow is parallel to the axis of the fuel pin.
- e. The cladding-fuel gap is assumed to be negligible.
- f. The smeared fuel density is assumed to be 95% of theoretical density.
- g. The wire wrap pins are assumed to have two points of support in each of three directions for each complete wrap of wire.

7.5.2.2 Discussion of Assumptions

The foregoing assumptions will be discussed in the order of appearance.

- a. The use of a single-span, simply supported model is conservative. A multi-span analysis would yield higher natural frequencies as the stiffness of the structure would be increased. This would increase the structures resistance to vibratory forces. In reality, the spacers will not act as a simple support nor can they be expected to completely restrain the fuel pin. Since some movement of the fuel pin will be possible because of manufacturing tolerances, use of the simply-supported model maintains the conservative viewpoint desired in the analysis. Because of the large degree of conservatism in the single-span, simply-supported model, it is felt that the vibrational problems resulting from occasional loose spacers (greater than normal tolerance buildups) would be less severe than this analysis would predict.
- b. Use of an annulus width of one-half the fuel pin spacing ($P-D/2$) for virtual mass calculations is conservative. Because of the interaction between adjacent vibrating pins, calculation of the virtual mass analytically would be quite lengthy. Therefore, the above approximation was necessitated.

- c. Although the experimental tests upon which the vibration stability factor is based were performed using water as the coolant, the characteristics in sodium should be quite similar.
- d. With wire wrap spacers where the flow may not be parallel to the fuel pins, conservatism is maintained as credit is not taken for the fact that the fuel pin is supported in three directions.
- e. Neglecting the cladding-fuel gap (and replacing it with fuel) decreases the natural frequency which decreases the vibrational stability.
- f. The use of 95% of theoretical density for the smeared fuel density is also conservative.
- g. This assumption is not valid for edge pins or if some pins are unwrapped.

7.5.2.3 Results

The curves in Figures 7-9 through 7-11 show the variation of the vibration stability factor with unsupported length for various velocities. Figure 7-9 represents the "worst case" while Figures 7-10 through 7-11 show the improvement in the stability factor with increased cladding thickness and pitch-diameter ratio, respectively.

The design curves in Figures 7-12 through 7-15 allow one to determine the maximum spacing for the P/D, velocity, and cladding thickness under consideration. Separate curves are presented for grid-type spacers and wire wrap spacers. These curves represent the combinations which yield a vibrational stability factor of 3.0.

7.5.2.4 Discussion of Results

The use of the design curves (Figures 7-12 through 7-15) deserves some discussion. Because of the possibility of loose spacers being present in a bundle, the curves should not be considered as

being conservative. If the design conditions (P/D, velocity, and cladding thickness) lie between two lines of constant spacer pitch, the lowest pitch should be used.

The general results indicate that for wire wrapped bundles the wire wrap pitch for internal pins will not be determined by vibrational considerations. If vibration limits the spacing for edge pins, the wire pitch can be reduced by a factor of two on these pins. The vibrational characteristics of grid-type spacers are such that they may indeed control the spacer pitch.

7.5.2.5 Analysis

The vibrational stability factor V_f used as a design criteria in this analysis was proposed by E. P. Quinn. This parameter was determined from experimental work performed by Quinn. If $V_f < 3.0$, then experimental verification would be required.

The vibrational stability factor is defined by:

$$V_f = \frac{\omega_n L}{V} \quad (1)$$

where

ω_n = natural frequency of fuel pin

L = unsupported length of the fuel pin

V = velocity of coolant.

The natural frequency of the fuel pin is given by: (for simply-supported)

$$\omega_n = \frac{1.57}{L^2} \sqrt{\frac{EI_g}{M}} \quad (2)$$

where

E = modulus of elasticity of cladding

I = moment of inertia of cladding

M = mass of beam

g = gravitational constant.

FIGURE 7-9. VIBRATION STABILITY FACTOR VERSUS UNSUPPORTED LENGTH, GRID SPACERS

FIGURE 7-10. VIBRATION STABILITY FACTOR VERSUS UNSUPPORTED LENGTH, GRID SPACERS

FIGURE 7-11. VIBRATION STABILITY FACTOR VERSUS UNSUPPORTED LENGTH, GRID SPACERS

FIGURE 7-12. DESIGN CURVE FOR VIBRATIONAL STABILITY—10-mil CLADDING, GRID SPACERS

FIGURE 7-13. DESIGN CURVE FOR VIBRATIONAL STABILITY—15-mil CLADDING, GRID SPACERS

FIGURE 7-14. DESIGN CURVE FOR VIBRATIONAL STABILITY—10-mil CLADDING, WIRE WRAP SPACER

FIGURE 7-15. DESIGN CURVE FOR VIBRATIONAL STABILITY—15-mil CLADDING WIRE WRAP SPACER

The modulus of elasticity and moment of inertia are considered for the cladding only as the fuel does not add strength to the beam.

The moment of inertia is given by:

$$I = \frac{\pi}{64} (D_o^4 - D_i^4) \quad (3)$$

where

D_o = outer diameter of cladding

D_i = inner diameter of cladding.

7.5.2.6 Grid Spacers

The mass of the beam consists of the mass of the cladding, fuel, and the virtual mass of the coolant:

$$M = M_c + M_f + M_v \quad (4)$$

where

M_c = mass of cladding

M_f = mass of fuel

M_v = virtual mass of coolant.

The masses of the cladding and fuel are determined from:

$$M_c = \rho_c \frac{\pi}{4} (D_o^2 - D_i^2) \quad (5)$$

$$M_f = \rho_f \frac{\pi}{4} D_i^2 \quad (6)$$

where

ρ_c = density of cladding lb/in.³

ρ_f = density of fuel lb/in.³

The virtual mass associated with a rod vibrating in a circular tube is given by:

$$M_v = \left[1 + \frac{\text{Rod Diameter}}{2 (\text{Annular Gap})} \right] (\text{Mass of fluid displaced by rod})$$

$$M_v = \left[1 + \frac{D_o}{2G} \right] \left(\rho_\ell \frac{\pi}{4} D_o^2 \right) \quad (7)$$

where

G = width of annular gap

ρ_ℓ = density of coolant.

By substituting Equations (3) through (7) into Equation (2) the natural frequency is:

$$\omega_n = \frac{1.57}{L^2} \left\{ \left[\frac{E\pi}{64} (D_o^4 - D_i^4) g \right] \right. \\ \left. + \left[\frac{\rho_c \pi}{4} (D_o^2 - D_i^2) + \frac{\rho_f \pi}{4} D_i^2 + \left(1 + \frac{D_o}{2G} \right) \left(\frac{\rho_\ell \pi}{4} D_o^2 \right) \right] \right\}^{1/2} \quad (8)$$

Two time-sharing computer codes were written to evaluate the natural frequency ω_n and the vibration stability factor V_f . The results of these programs are presented in Figures 7-9 through 7-13.

7.5.2.7 Wire Wrapped Spacers

When considering wire wrapped spacers, one must include the effect of the wire as it vibrates with the fuel pin. For the purposes of this study, it was assumed that the wire adds no strength to the pin but does increase the mass. This additional mass is:

$$M_w = \rho_w \frac{\pi}{4} (P-D)^2 \quad (9)$$

This mass was added to Equation (4) and used in a time-sharing computer program to generate the curves in Figures 7-14 and 7-15.

7.5.3 VELVET Code

A parametric study of the temperature distribution around a sodium-cooled fuel pin in a close-packed rod bundle, using the VELVET code, has been completed. The calculations were done for various values of pitch-to-diameter ratio, Reynolds number and linear pin power. The calculational model and results of the analysis will be presented in a topical report.

FIGURE 7-16. NUSSELT NUMBER

Briefly, the results of the study indicated the following significant effects:

- a. At small pitch-to-diameter ratios, the velocity distribution has a strong effect on the cladding and coolant temperature distribution and thus on the resultant heat transfer coefficient. The need for a reliable knowledge of the velocity profile and turbulent mixing contributions is strongly indicated.
- b. A slug flow approximation will yield pessimistic results in an analysis of the circumferential cladding temperature profiles.
- c. For pitch-to-diameter ratios above 1.5, circumferential variations in the cladding temperature profile and cladding to coolant heat flux become insignificant in the turbulent flow regime.

Also, as part of the study, the convective heat transfer coefficient to the sodium

(Nusselt number) has been calculated, as a function of Peclet number and pitch-to-diameter ratio, and is presented in Figure 7-16.

For Peclet numbers above 300, the curves for all pitch-to-diameter ratios show an upward trend, as a result of increased turbulent mixing at higher flow velocities. For Peclet numbers below 300 and P/D on the order of 1.3 or greater, the curves are flat, because of the absence of mixing in the model used. However, for P/D less than 1.3, the Nusselt number keeps decreasing with decreasing Peclet number, because of the highly nonuniform circumferential flow distribution and absence of turbulent mixing. It is this region ($P/D < 1.3$ and $Pe < 300$) where slug flow becomes a poor approximation for calculating the temperature distribution in this geometry.

SECTION VIII

TASK I—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. However, no material 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 4 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 processes.
- 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 MATERIALS IRRADIATION PERFORMANCE

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

Materials capsules L-2, L-6, L-10, and L-14 are currently in EBR-II. As of December 1967, the approximate total fluences, based on EBR-II core physics, are 4.7, 1.8, 1.2, and 0.6×10^{22} nvt, respectively. Complete descriptions of these series and their experimental objectives have been given previously. (3, 5)

8.2.2 Series L-2' and L-4/L-4' EBR-II Materials Capsules*

Initial tensile results from five Series L-2' capsules (2.5×10^{22} n/cm², total) two Series L-4 capsules (1.5×10^{22} n/cm², total) and five Series L-4' capsules (1.1×10^{22} n/cm², total) were reported earlier. (7)

Further tensile testing, delayed because of redesign and fabrication of improved tensile grips, has been continued at 1100° and 1300°F. Preliminary results to date, on irradiated samples, are presented in Table 8-1 and Figures 8-1, 8-2, and 8-3.

Shown in Figure 8-1 is the effect of fast-neutron exposure on the uniform tensile elongations of four candidate alloys (Types 304, 316L, and 347 stainless steels, and Incoloy 800) in the mill-annealed condition. Tensile testing was conducted in duplicate at 1300°F and a crosshead rate of 0.02 inch/min (1-inch gage length) subsequent to a 15-minute soak at temperature.

Within the limited neutron fluence range investigated, it appears that all materials irradiated in the mill-annealed condition were embrittled, with the degree of embrittlement increasing with increasing fluence. Type 304 and Type 316 L seem to be embrittled less severely at the higher neutron exposures, exhibiting uniform strains of 3.0 to 4.5%. Type 347 and Incoloy 800 showed similar, but

*The total fluence values reported for these capsules were obtained from flux-wire activities using the (n, p) reactions of Fe-54 and Ni-58. The threshold energies associated with these (n, p) reactions are between 1 and 2 MeV. The total fluence was obtained by an extrapolation of these measurements over the entire energy spectrum utilizing a measured/computed EBR-II flux spectrum. The total neutron fluences calculated in this manner are significantly different (less) than those calculated by ANL-EBR-II using a power normalization method. Until this discrepancy has been rectified, reported fluences will be based on flux-wire measurements. Of the total fluence reported, 25–30% is considered to be greater than 1 MeV.

TABLE 8-1

POST-IRRADIATION TENSILE TESTING AT 1300°F AND A STRAIN RATE OF 0.02 MINUTE⁻¹

Alloy	Structure	Neutron Exposure (E>1 MeV), n/cm ²	Uniform Strain, %	Total Strain, %	Yield Stress (0.2%), ksi	Ultimate Stress, ksi
Type 304	Mill Annealed ⁽¹⁾	1.3×10^{21}	4.4—6.8 ⁽²⁾	4.9—7.0 ⁽²⁾	19.4—26.9 ⁽²⁾	34.3—35.8 ⁽²⁾
Type 304	Mill Annealed ⁽¹⁾	1.3×10^{21}	7.1—7.8	12.6—12.9	27.8—28.4	34.3—34.7
Type 304	Mill Annealed ⁽¹⁾	6.7×10^{21}	3.3—3.8	5.2—8.3	36.0—38.4	40.7—41.0
Type 316L	Mill Annealed ⁽¹⁾	1.3×10^{21}	4.0—7.1	6.4—9.1	30.1—39.2	39.5—44.3
Type 316L	Mill Annealed ⁽¹⁾	1.3×10^{21}	4.6—9.4	7.6—12.3	22.5—32.0	32.6—37.5
Type 316L	Mill Annealed ⁽¹⁾	6.7×10^{21}	3.1—4.6	4.4—7.7	39.9—41.7	44.1—47.3
Type 347	Mill Annealed ⁽¹⁾	1.3×10^{21}	4.0—4.1	6.8—7.3	33.1—35.5	39.2—40.7
Type 347	Mill Annealed ⁽¹⁾	6.2×10^{21}	0.7	1.4	43.8	44.2
Type 347	Mill Annealed ⁽¹⁾	6.7×10^{21}	0.9 ⁽²⁾	1.0 ⁽²⁾	40.5 ⁽²⁾	44.5 ⁽²⁾
Incoloy 800	Mill Annealed ⁽¹⁾	1.3×10^{21}	3.2—4.5	6.6—8.8	34.8—42.0	39.0—43.3
Incoloy 800	Mill Annealed ⁽¹⁾	1.3×10^{21}	3.0—3.8	4.3—4.9	41.0—44.3	45.1—48.5
Incoloy 800	Mill Annealed ⁽¹⁾	6.2×10^{21}	0.7—0.8 ⁽²⁾	0.8—1.3 ⁽²⁾	-	-
Incoloy 800	Mill Annealed ⁽¹⁾	6.2×10^{21}	2.8—2.7	4.5—5.8	38.7—40.7	40.3—41.0
Incoloy 800	Mill Annealed ⁽¹⁾	6.7×10^{21}	2.0—3.3 ⁽²⁾	2.2—3.9 ⁽²⁾	-	-

(1) Annealed at 1800°F for 15 minutes prior to irradiation.

(2) Test performed at a strain rate of 0.01 min⁻¹.

FIGURE 8-1. UNIFORM TENSILE STRAIN VERSUS NEUTRON FLUENCE FOR AUSTENITIC STAINLESS STEELS AND INCOLOY 800 IRRADIATED AT 1000°-1200°F AND TESTED AT 1300°F

FIGURE B-2. YIELD STRESS VERSUS NEUTRON FLUENCE OF AUSTENITIC STAINLESS STEELS AND INCOLOY 800 IRRADIATED AT 1000°—1200°F AND TESTED AT 1300°F

FIGURE 8-3. TENSILE STRESS VERSUS NEUTRON FLUENCE FOR AUSTENITIC STAINLESS STEELS AND INCOLOY 800 IRRADIATED AT 1000°—1200°F AND TESTED AT 1300°F

lower, ductilities, 0.7 to 3.0% in this neutron fluence range. (Unirradiated uniform strain values for all candidate materials were 9 to 17% at 1300°F.)

The data for mill-annealed Type 304 agree favorably with the results obtained by Holmes and Irwin at PNL⁽²⁵⁾ at comparable conditions (1.7×10^{22} n/cm² total and 1300°F). Since the flux > 1 MeV in EBR-II is approximately 25 to 30% of the total, an equivalent measurement would be 4.8×10^{21} n/cm², E>1 MeV. PNL results indicate ~2-1/2% uniform tensile strain at these conditions compared to the ~4% strain reported here.

Kangilaski and Wullaert⁽²⁶⁾ of BMI have reported tensile properties for irradiated Type 347. From their curve, at 1300°F and 2.1×10^{22} n/cm²; E>1 MeV (ETR exposure), the uniform elongation is ~0.7%. An extrapolation of the Type 347 curve in Figure 8-1 to 2.1×10^{22} n/cm² (E>1 MeV) indicates a very low uniform strain, which is in agreement with the BMI data.

The effect of irradiation on the elevated-temperature yield stress (0.2% offset) is presented in Figure 8-2. Mill-annealed Types 304, 316L, and 347 all show higher yield stress than representative unirradiated values because of the fast neutron exposure and increase slightly with increased exposure. PNL has previously presented⁽²⁵⁾ data on the effects of radiation on the yield strength of mill-annealed Type 304 at 1.7×10^{22} total neutron fluence (4.8×10^{21} n/cm², E>1 MeV); these results indicated that radiation-induced strengthening is retained at this neutron exposure to a test temperature of 1300°F, which is in direct agreement with the data presented in Figure 8-2. PNL⁽²⁶⁾ reports for Type 347 ~36,000 psi yield strength at 1300°F and 2.1×10^{22} n/cm² (E>1 MeV). A disparity exists with the value extrapolated here (49,000 psi) for Type 347 irradiated in EBR II. The disagreement may be due to differences in the fast neutron energy spectrum.

Incoloy 800 exhibits nearly constant yield strength with increasing fluence. Therefore, at fluences to 6.7×10^{21} n/cm² (E>1 MeV) and elevated temperatures of irradiation ($\geq 1/2 T_m$), radiation strengthening not only persists but appears to be increasing with increasing fluence in mill-annealed Types 304, 316L, and 347. Incoloy 800 also shows evidence of radiation strengthening, but the effect does not increase with increasing fluence.

The same general trends were observed in the ultimate tensile strengths of the test materials (Figure 8-3) as were observed for the yield strengths.

Disparity again exists between the PNL data⁽²⁶⁾ on Type 347 irradiated in ETR: 37,000 psi at 2.1×10^{22} (E>1 MeV) compared to the ~49,000 psi value extrapolated in Figure 8-3.

8.2.3 Effect of Irradiation on Burst Properties of Type 304 Stainless Steel*

An experimental program to evaluate the effect of fast flux irradiation on the elevated temperature strength and ductility of austenitic stainless steel tubing has been performed. Test material represented core and blanket regions of the Type 304 stainless steel capsule tubes used in the F2 fuel irradiation test in EBR-II. The F2 capsule tubes—3/8-inch diameter, 0.020-inch wall, weld drawn—were irradiated in Row 4 in EBR-II subassembly XG06 and accumulated a fluence of 6.7×10^{21} nvt (E>1 MeV). The radiation temperature was estimated to have been 800°F at the outer (flowing sodium) surface. Metallographic examination of a representative F2 capsule tube yielded no positive evidence of structural change because of the radiation environment.⁽⁷⁾ However, it was observed

*See Footnote on page 86.

that the material was lightly sensitized throughout its structure, a somewhat unexpected occurrence at the temperature noted above, which suggests that carbide precipitation may have been enhanced by radiation.

Burst testing on as-received (annealed) control and irradiated specimens was performed at 900°, 1100°, 1300°, and 1500°F. The tubular sections were equilibrated at the test temperature for 30 minutes prior to pressurization with argon. The specimens were pressurized at a rate of approximately 200 psi/min. As shown in Figure 8-4, fracture appears to have occurred generally along the seam weld. The experimental and analytical data are listed in Table 8-2 and plotted in Figure 8-5. Thin wall analysis was applied to convert burst pressure to hoop strength.

The data show a marked increase in hoop strength with decreasing test temperature, relative to that of the unirradiated controls. In all cases, the specimens irradiated in the core region of EBR-II exhibited higher hoop strength than those exposed in the blanket. Extrapolation of the data show that the hoop strengths of irradiated and control specimens converge at approximately 1600°F, indicating complete recovery at this temperature. This work is in close agreement with that of Holmes and Irwin,⁽²⁵⁾ who observed that radiation hardening of Type 304 stainless steel in a fast flux is stable to $0.67 T_m$. Recent studies performed on Type 304L stainless steel irradiated to 1.4×10^{22} n/cm² in EBR-II,⁽²⁷⁾ while showing similar change in rupture strength with test temperature, indicate that radiation-induced strengthening in this modified alloy is completely removed at approximately 1400°F. This lower recovery temperature may be associated with the higher radiation temperature of the ANL cladding specimens.

The effect of irradiation on the elevated temperature ductility of Type 304 stainless steel is indicated in Figure 8-6. (Diametral fracture strain values, calculated from micrometer measurements adjacent to the fracture tip, are listed in Table 8-3. Although the spread in the data is somewhat large, the measurements show a very definite reduction in elevated temperature ductility with irradiation. The data further show an increase in ductility with test temperature to 1300°F with an apparent reduction above 1300°F. Although the range of test temperatures was insufficient to establish a minimum ductility temperature, previous work⁽²⁷⁾ has shown a minimum ductility at approximately 900°F for irradiated Type 304 stainless steel, with a slight recovery above this temperature. It is perhaps significant, in this regard, that no substantial radiation damage recovery could be detected in these tests, although all specimens were held at the test temperature for 30 minutes prior to the application of pressure.

8.3 FUEL-CLADDING COMPATIBILITY

8.3.1 Out-of-Pile Compatibility Studies

Quantitative analysis of the reaction zones formed on Type 304 stainless steel annealed at 1400°F for 1000 hours in contact with hyperstoichiometric mixed plutonia-urania fuel revealed that the outer metallic-appearing layer is virtually 100% iron while the metallic content of the inner gray layer is approximately 75% (iron + chromium).⁽⁷⁾ The fact that no other metallic constituent was detected that could constitute the balance in the gray layer suggested that the only other major constituent capable of reacting with the cladding is oxygen. It was assumed that, since the compatibility capsule was sealed in an inert environment, the only source of oxygen in this system was the fuel. This assumption was strengthened by the observation that no reaction occurred in identical capsules containing stoichiometric or sub-stoichiometric fuel.

a) F2E, BLANKET, 1300°F

c) F2S, BLANKET, 1500°F

b) F2E, CORE, 1300°F

d) F2S, CORE, 1500°F

FIGURE 8-4. RUPTURE CHARACTERISTICS OF TYPE 304 STAINLESS STEEL BURST SPECIMENS

FIGURE 8-5. EFFECT OF TEST TEMPERATURE ON THE HOOP STRENGTH OF IRRADIATED TYPE 304 STAINLESS STEEL TUBING

FIGURE 8-6. EFFECT OF TEST TEMPERATURE ON THE RELATIVE DUCTILITY OF CONTROL AND IRRADIATED TYPE 304 STAINLESS STEEL TUBING

TABLE 8-2

BURST DATA ON F2 CAPSULE TUBES* - IRRADIATED AND CONTROL

Specimen No.	Test Temperature (°F)	Time to Rupture (min)		Burst Pressure (psi)		Hoop Strength (psi)	
		Core	Blanket	Core	Blanket	Core	Blanket
Control 1	900		28.8		6450		57,000
Control 2	900		28.9		6250		55,400
Control 3	900		32.3		6250		55,400
F2P-X	900	>57.9**		>9235**		>81,800	
Control 4	1100		27.0		5300		47,000
Control 5	1100		27.3		5170		45,900
Control 6	1100		25.2		5300		47,000
F2A	1100	25.2	38.5	8020	7000	71,100	62,100
F2B	1100	30.4	42.8	8210	7025	72,800	62,300
F2Y	1100	43.1	21.5	8195	7435	72,700	65,900
Control 7	1300		16.7		3530		31,300
Control 8	1300		17.0		3440		30,500
Control 9	1300		18.0		3700		32,800
F2E	1300	23.3	21.1	4735	4060	42,000	36,000
F2U	1300	22.1	21.7	5020	4125	44,500	36,600
F2Z	1300	20.8	22.7	5175	4300	45,900	38,100
Control 10	1500		9.4		1820		16,100
F2N	1500	10.5	15.2	2910	2530	25,800	22,400
F2P	1500	16.3	12.2	2860	2485	25,400	22,000
F2S	1500	11.0	14.2	2660	2585	23,600	23,000

* Annealed Type 304 Stainless Steel.

** Specimen did not burst at pressure limit of equipment.

GEAP-5584

TABLE 8-3

POST-BURST DIAMETRAL CHANGES IN CONTROL AND IRRADIATED

TYPE 304 STAINLESS STEEL TUBING

Specimen No.	Position *	Temperature (°F)	D _{Top} ** (in.)	D _{Bottom} (in.)	D _{Average} (in.)	ΔD (in.)	ΔD/D ₀ (%)
1	Control	900	0.433	-	0.434	0.059	15.7
2	Control	900	0.438	-			
3	Control	900	0.432	-			
F2A-2	Core	1100	-	0.380	0.377	0.002	0.5
F2B-2	Core	1100	-	0.376			
F2Y-2	Core	1100	0.375	0.376			
F2A-4	Blanket	1100	0.396	0.400	0.390	0.015	4.0
F2B-4	Blanket	1100	-	0.389			
F2Y-4	Blanket	1100	0.386	0.379			
4	Control	1100	0.447	0.431	0.439	0.064	17.0
5	Control	1100	0.442	0.442			
6	Control	1100	0.432	-			
F2E-2	Core	1300	-	0.383	0.385	0.010	2.7
F2U-1B	Core	1300	-	0.391			
F2Z-2	Core	1300	0.388	0.379			
F2E-4	Blanket	1300	0.397	0.406	0.397	0.022	5.9
F2U-4B	Blanket	1300	0.394	-			
F2Z-4	Blanket	1300	0.394	0.392			
7	Control	1300	0.437	-	0.446	0.071	18.9
8	Control	1300	0.451	0.443			
9	Control	1300	0.442	0.456			
F2N-2	Core	1500	-	0.378	0.380	0.005	1.3
F2P-2	Core	1500	0.380	0.380			
F2S-2	Core	1500	0.378	0.383			
F2N-4	Blanket	1500	0.389	0.400	0.397	0.022	5.9
F2P-4	Blanket	1500	0.395	0.395			
F2S-4	Blanket	1500	0.402	0.402			
10	Control	1500	0.434	-	0.434	0.059	15.7

*Core fluence was 2.5×10^{22} total n/cm². Blanket fluence was 0.5×10^{22} total n/cm².

**Diametral measurements were made above and below the fracture region, at a point 1/2-inch from the crack tip.

Additional analytical work has now established that hyperstoichiometric mixed plutonia-urania fuel does indeed lose oxygen in this system and can account for the cladding oxidation observed. An electrical conductivity probe across the fuel, parallel to the diffusion direction, revealed a reduction in conductivity of the fuel in a surface zone 100 microns wide. This reduction represents a loss of oxygen since resistivity in mixed oxides is known to increase as the oxygen-to-metal ratio is decreased.⁽²⁸⁾

Since the outer cladding layer is metallic rather than an oxide of iron, as would be expected in austenitic stainless steel oxidation in air, it is proposed that in the absence of adequate oxygen activity or a reduction in activity with time, the iron oxide (in this case, Fe_3O_4) would decompose to FeO , thence to metallic iron. This suggests that the rate-controlling step in the oxidation of stainless steel in this isothermal system is the diffusion of oxygen to the surface of the fuel.

This experimental program is now completed. In summary, the work has shown that:

- a. Stoichiometric and substoichiometric mixed plutonia-urania fuels are fully compatible with austenitic stainless steel to temperatures as high as 1400°F and annealing times of 1000 hours in an isothermal environment, and
- b. Slight oxidation of these alloys occurs at 1400°F and 1000 hours in systems containing hyperstoichiometric mixed plutonia-urania fuels, the oxygen environment being provided in this case by the release of oxygen from the fuel.

8.3.2 Fuel-Sodium-Cladding Compatibility

The compatibility of mixed-oxide fuel and austenitic stainless steel cladding in the presence of sodium vapor at 1400°F

has been evaluated in a long-term (1000-hour) anneal. Results of a cursory examination of the several capsules were reported previously.⁽⁷⁾ Detailed metallographic examination of the capsules, performed on radial cross sections through the fueled portion, showed evidence of sodium intrusion or sodium-fuel reaction in all fuel stoichiometries tested. The reaction appeared to be most severe, however, in capsules containing hyperstoichiometric ($\text{O/M} = 2.04$) fuel. The product may be characterized as a homogeneous gray material, generally occurring in fissures at or near the fuel periphery. A reaction product typical of that observed in all fuel compositions is shown in Figure 8-7. The material was rapidly attacked by aqueous etching solutions, which resulted in severe grain pull-out, particularly in the fine grained hypo-stoichiometric ($\text{O/M} = 1.97$) fuel.

Attack of the inner cladding surface occurred to some extent in all capsules although, like the sodium-fuel reaction, the attack was most prominent in capsules containing hyperstoichiometric fuel. Representative metallography of all fuel-cladding combinations tested is shown in Figure 8-8. The apparent reaction between Type 321 stainless steel and stoichiometric fuel represents a very localized effect and is considered atypical.

Additional analytical work aimed at establishing the composition of the sodium-fuel reaction product and the cladding surface reaction product is currently being performed.

8.4 CLADDING OPTIMIZATION STUDIES

8.4.1 Experimental Strain Analysis

8.4.1.1 Background

Work is continuing on the determination and understanding of cladding and cladding specimen deformation. Previous work has

been involved in estimating strains in cladding specimens and test fuel pins with the aid of profilometer or micrometer measurements. The limited ability of this method of examination to measure local surface strains was discussed previously.⁽⁶⁾ The need to know the type and level of local strain patterns for determining and understanding deformation mechanisms has resulted in a search for a method of directly measuring local surface strain.

A review of the methods experimental strain measurements revealed the shortcomings and limitations of most methods that prevent their utilization in the environments of most cladding tests. A suitable method appears to be the simple technique of placing fiducial marks on the surface and measuring the change in such "gage lengths" by making before and after measurements between marks.

Two methods of marking appear suitable for the severe environments encountered in

program tests. One method is to indent the surface and the other is to deposit a durable substance on the surface. The first method has received the most attention.

Indentations used in the course of micro-hardness tests were investigated for marking. The Vickers diamond point mark when properly oriented on the diagonal appeared to be the most suitable. Typically a 1 kg load produces a mark 2-3/4 mils across the diagonal and 0.4 mil deep.

Measuring microscopes are suitable for locating and measuring distances on such a small scale. The bifilar measuring system of the Tukon hardness testing instrument, which was used to make the indentations, was found to be the most satisfactory instrument readily available.

The moire method of strain analysis uses a fine accurate grid that can be

(REDUCED TO 75%
FOR REPRODUCTION)

AS-POLISHED

500X

FIGURE 8-7. SODIUM-FUEL REACTION PRODUCT IN TYPE 316 STAINLESS STEEL-CLAD CAPSULE CONTAINING HYPERSTOICHIOMETRIC FUEL

(REDUCED TO 64% FOR REPRODUCTION)

FIGURE 8-8. INTERACTION OF AUSTENITIC STAINLESS STEEL WITH MIXED OXIDE FUEL IN THE PRESENCE OF SODIUM. COUPLES ANNEALED FOR 1000 HOURS AT 1400°F. ALL PHOTOMICROGRAPHS: AS-POLISHED, 500X

etched into the specimen surface and used as a premeasured fiducial system for microscope measurement. Also, it appears that quantitative strain measurements may be obtainable by moiré fringe interpretation while the least that could be expected from moiré fringe analysis is a qualitative indication of areas of high strain to be examined in detail under a microscope.

To provide initial evaluation and demonstration of these methods, four areas of interest were investigated.

8.4.1.2 Analysis of Etched Grid

A 200-line-per-inch grid was etched on each of two Type 304 stainless steel tubes, $1/4 \times 0.015$ inch, 3 inches long; and a Type 304 stainless steel pressurized pin. One of the tubes was fabricated into three burst specimens and the other was retained for future analysis and experimentation.

Figure 8-9 shows the grid at 250 $\times$ magnification in its best condition. The etching process removed metal from the "lines" of the grid, leaving a pattern of raised squares. Depth of etch ranged from 0.23 to 0.64 mil in the clearest areas. The quality of etching was quite variable between tubes and between various locations on the surface of any given tube.

It was determined that in the best regions of the grid $\pm 20\%$ accuracy at 2% strain could be obtained without premeasurement, if three measurements were made over a distance of 6 squares (gage length about 30 mils). Accuracy would, of course, be better with more observations, a longer gage length, or at a higher strain level.

While accurate etching of grids on flat surfaces is common practice, this is the first known attempt to etch on a surface of this sharp curvature. The vendor experienced

FIGURE 8-9. BEST ETCHING OF GRID—BURST SPECIMEN 2

difficulty in achieving the results obtained but it seems reasonable to expect useable results after even small improvements in equipment and/or techniques.

8.4.1.3 Effects of Sodium Exposure on Fiducial Systems

Two pressurized pins of Type 304 stainless steel were marked with various fiducial markings and are now immersed in flowing sodium at 1300°F. The marking systems were documented and will be compared with post-test conditions after a 1000-hour test period. One pin was etched with a 200-line-per-inch grid as described in the previous section while the other specimen will be used to evaluate the durability of various types and weights of microhardness indents through sodium exposure.

8.4.1.4 Effect of Microhardness Indentations on Tensile Failure

To determine initially the effects of indents on performance, the most notch sensitive and readily available material was used. Five identical tensile specimens were stamped from Type 316 stainless steel sheet, 75% cold worked, 16 mils thick with a gage section 0.1 inch wide. These specimens all had a smooth, shallow, nearly invisible flaw on one edge (see Figure 8-10). Past experience with similar specimens indicated that failure would occur at this position; consequently, a row of microhardness indentations was made across the width of the specimen at this point, typically as shown in Figure 8-10. Various depths were used with a spacing of about 10 mils. The most severe condition contains two rows of marks about 3 millimeters apart made under 1 kilogram load, which produced marks about 0.4 mil deep.

It was observed that although failure occurred very near the lines of marks across the specimen (as dictated by the

presence of the flaw), it was found that, of the four specimens tested in this manner, the fracture did not pass through even one indentation. Therefore, it is concluded that for these specimens the marking did not affect failure. Figures 8-10 and 8-11 show typical before and after photographs.

The specimen shown in Figures 8-10 and 8-11 yielded an interesting by-product of this study. On this specimen were located a series of indents along the length, enabling a plot of strain versus length to be made (Figure 8-12). This plot demonstrates the ability of the fiducial system to measure localized strain on a tensile specimen with a gage length of less than 0.010 inch.

8.4.1.5 Use of Fiducial Systems in Measuring Expansion of Tubing

The purpose of this task was to conduct a demonstration test of the fiducial systems, while investigating the suspected nonuniform circumferential deformation in expanding tubing.

A burst specimen was fabricated by welding end plugs and threaded connector tubes on a short length of tubing grid etched as previously described.

It was found that welding can affect the dimensional stability of bench mark patterns placed prior to assembly. Slight changes in measurements were noted. It is highly possible that this effect is felt only near the weld region and may not even need consideration in the center sections of longer specimens. Any serious problem can be avoided by etching or marking after assembly.

To investigate nonuniform strain around the circumference of a tube, a pattern of 29 1-kg marks was made and premeasured at the middle of the specimen, one mark every 12.6 degrees around its entire circumference. The specimen was then hydraulically

50X

FIGURE 8-10. TENSILE SPECIMEN "A" BEFORE TEST, SHOWING AREA OF FLAW ON RIGHT

50X

FIGURE 8-11. TENSILE SPECIMEN "A" AFTER FAILURE

FIGURE 8-12. LOCAL STRAIN VERSUS POSITION ALONG THE GAGE LENGTH OF A TENSILE SPECIMEN OF TYPE 316, 75% COLD WORKED

pressurized to 8600 psi internal (which represents a hoop stress of approximately 143,000 psi) in two increments, then depressurized.

Remeasurement of the bench marks after depressurization indicated the irregular strain of Figure 8-13, the top curve of which is a plot of the wall thickness for comparison. The two curves confirm that circumferential deformation is an inverse function of the wall thickness at any point. The standard deviation of strain measurement was determined to be 0.4% strain.

The average $\Delta D/D$ strain obtained by micrometer measurements of the tube in eight directions, was found to be 7.67%. This compares quite favorably with the average circumferential strain of 7.74% obtained by averaging all of the individual strains from the bench mark measurements.

At six locations around the circumference, the etched grid was surveyed (four readings each direction, two and six squares at a time) to assess the usefulness of direct grid measurement in strain calculations. The strains so measured are shown as squares on Figure 8-13. They appear to correspond fairly well with those strains calculated from the bench marks.

By increasing grid contrast by an inking procedure, it was possible to make some measurements by the moiré fringe technique. A replica of the original grid master used in etching was obtained on a photographic negative. Wrapping this grid around the specimen in the proper orientation made it possible to produce fringes with a spacing of approximately 1/16 inch. The average of twelve random readings produced a strain value of 5.8%, which is about 1.9% below the averages obtained above. This degree of error is not surprising in view of the rather primitive preliminary nature of the

equipment used in etching and measurement. For example, it was found that the replica grid on the negative varied from 196 lines per inch to 204 lines per inch, depending on the region examined, which alone could account for the discrepancy.

Considerable refinement in the methods above is possible and required to make the moiré method usable in a quantitative way on the strain level expected in most experiments. Increasing the number of lines per inch, improved grid application method, more accurate replica grid, and better means of measuring fringe spacing are areas of possible improvement.

8.4.1.6 Conclusions and Applications

It has been demonstrated that the microsurveying method can be used to measure accurately local deformation patterns on 1/4-inch tubing. Microhardness bench marks are the most accurate, yet etched grids are acceptable, particularly in view of possible improved methods.

It can be concluded so far that the specific method of surveying bench marks with a measuring microscope can be used to investigate variation in strain patterns locally on any area of interest, in tests conducted at elevated temperature out of sodium and without irradiation. At present, or in the near future, such tests include tensile tests and creep tests, and on tubing, burst and stress rupture tests. With little improvement in grid etching techniques, grids could be used extensively as a substitute for bench marks in these investigations.

8.4.2 Materials Management

An effort is continuing toward procurement of stainless steel tubing having selected microstructures currently being investigated

FIGURE 8-13. CIRCUMFERENTIAL STRAIN AND WALL THICKNESS AROUND CIRCUMFERENCE OF BURST SPECIMEN No. 3

for resistance to radiation damage as sheet specimens in Series L-10, L-14, and L-15. Negotiations are presently under way with a tube vendor for procurement of tubing which has received well-defined thermomechanical treatments.

All seamless tubing ordered for the P.A. 10 tasks has been received

and normal incoming tube inspection has been completed. Characterization of the tubing is in progress. This tubing includes Types 304H, 316H, 321H, and 347H stainless steels, and Incoloy 800, all in the solution-anneal condition. Also included in this order were Incoloy 800 and Type 316H tubing in the 25% cold-worked condition.

REFERENCES

1. Skirvin, S. C., "Users Manual for the THTD Computer Program," Nuclear Materials and Propulsion Operation, Nuclear Technology Department, General Electric Company, Cincinnati, Ohio, 1966, (G. E. R60FPD647).
2. Schoeberle, D. F., et al., "A Method of Calculating Transient Temperatures in a Multiregion Axisymmetric, Cylindrical Configuration. The ARGUS Program," Argonne National Laboratory, Argonne, Illinois, November 1963, (ANL-6654).
3. "Sodium-Cooled Reactor, Fast Ceramic Reactor Development Program, Twentieth Quarterly Report, August—October 1966," GEAP-5292.
4. "Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-First Quarterly Report, November 1966—January 1967," GEAP-5433.
5. "Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-Second Quarterly Report, February—April 1967," GEAP-5491.
6. "Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-Third Quarterly Report, May—July 1967," GEAP-5522.
7. "Sodium-Cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-Fourth Quarterly Report, August—October 1967," GEAP-5541.
8. Padden, T. R., and Schnizler, P. S., "Microstructural Changes Associated with Irradiation Burnup of Uranium Dioxide," Electron Microscopy: Fifth International Congress for Electron Microscopy, Academic Press, New York, 1962.
9. Schmidt, J. J., "Neutron Cross Sections for Fast Reactor Materials Part I: Evaluation," Kernforschungszentrum, Karlsruhe, Germany, February 1966, (KFK 120).
10. Abagyan, L. P., Bazazyants, N. O., Bondarenko, I. I., and Nikolaev, M. N., "Group Constants for Nuclear Reactor Calculations," Consultants Bureau, New York, N. Y. (1964).
11. Schomberg, M. G., Sowerby, M. G., and Evans, F. W., "A New Method of

REFERENCES (Continued)

- Measuring Alpha (E) for Pu-239," IAEA Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany, October 30—November 3, 1967, (SM 101/41).
12. Hellstrand, E., and Anderson, T. L., "Measurement of the Effective Capture to Fission Ratio in Pu-239 and U-235 in Different Fast Reactor Spectra," IAEA Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany, October 30—November 3, 1967, (SM 101/2).
 13. Campbell, C. G., et al., "A Comparison of Experiment with Prediction of the Variation of k-Inf with Coolant Density for a Plutonium Fueled Steam-Cooled Fast Reactor Lattice," IAEA Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany, October 30—November 3, 1967, (SM 101/43).
 14. Pönitz, W. P., Kompe, D., Menlove, H. O., and Beckurts, K. H., "Some New Measurements and Renormalizations of Neutron Capture Cross Section Data in the keV Energy Range," IAEA Symposium on Fast Reactor Physics and Related Safety Problems, Karlsruhe, Germany, October 30—November 3, 1967, (SM 101/9).
 15. Memo of January 16, 1967, from P. Greebler to Cross Section Evaluation Working Group Participants, "Testing of ENDF/B Cross-Section Data - Information on Simplified Model of ZPR-III Critical Assembly 48."
 16. White, P. H., Hodgkinson, J. G., and Wall, G. J., Proc. Salzburg Conference on Physics and Chemistry of Fission, IAEA Salzburg, Austria (1965).
 17. Stehn, J. R., Goldberg, M. D., Wiener-Chasman, R., Mughabghab, S. F., Magurno, B. A., and May, V. M., "Neutron Cross Sections," BNL-325, 2nd ed., Supplement No. 2, III, Brookhaven National Laboratory, February 1965.
 18. Greebler, P., Aline, P., and Hutchins, B. A., "Evaluation and Compilation of Pu-239 Cross Section Data for ENDF/B Files," GEAP-5272, December 1966.
 19. Greebler, P., Hutchins, B. A., and Linford, R. B., "Sensitivity of Fast Reactor Economics to Uncertainties in Nuclear Data," American Nuclear Society Winter Meeting, Chicago, Illinois, November 5—9, 1967.
 20. Greebler, P., et al., "BISYN - A Two Dimensional Synthesis Program," GEAP-4922, July 1965.
 21. Hassitt, A., "A Computer Program to Solve the Multigroup Diffusion Equations," 1962, TRG Report 229 (R).
 22. Fowler, T. B., Tobias, M. L., and Vondy, D. R., "EXTERMINATOR-2: A FORTRAN IV Code for Solving Multigroup Neutron Diffusion Equations in Two Dimensions," April 1967, (ORNL-4078).

REFERENCES (Continued)

23. Henry, A. F., and Vota, A. V., "WIGL2—A Program for the Solution of the One-Dimensional, Two-Group, Space-Time Diffusion Equations Accounting for Temperature, Xenon, and Control Feedback," October 1965, (WAPD-TM-532). 24. Hirokawa, N., "MARS, A Two-Dimensional Excursion Code," June 1967, (APDA-198).	25. Holmes, J. J., and Irwin, J. E., Trans. ANS, 10, 2 (1967). 26. Kangilaski, M., and Wullaert, R. A., August 15, 1967, (BNWL-532). 27. Kittel, J. H., December 1967, (BNWL-609). 28. Baily, W. E., and Lyon, W. L., "Some Electrical Properties of the Plutonium-Uranium Mixed Oxide System," GEAP-4675, September 1965.

CONTRIBUTORSTask B—Vented, Defected, and Sodium-Bonded Fuel Development

G. Collins	T. Lannin
R. Eymann	P. Novak
W. Faust	J. Peterson
K. Gregoire	H. Pfefferlen
R. Irvine	D. Plumlee
J. Jaklevick	D. Pomeroy

Task C—Safety Experimentation

P. A. Bohm	T. Lannin
D. Bracken	S. Lukezic
D. L. Brown	R. E. Murata
W. L. Conant	N. C. Shirley
R. H. Eymann	R. E. Smith
W. R. Faust	G. R. Thomas
J. H. Field	R. C. Thompson
J. D. Heald	F. R. Ulrech
T. Hikido	W. O. Wade
J. F. Jaklevick	O. A. Willson
W. H. Kelso	D. L. Zimmerman

Tasks E and F—Fuel Performance Evaluation and Integral Fast Flux Irradiations

E. Aitkin	E. Kurtz
R. Asamoto	T. Lannin
W. Baily	T. Lauritzen
P. Bohaboy	J. Lewis
W. Cowden	R. Nelson
C. Craig	K. Perry
W. Cummings	R. Pifferetti
E. Darlin	H. Rosenbaum
L. Davis	B. Rubin
R. Eymann	C. Ryer
J. Heald	N. Shirley
G. Hull	C. Spalaris
J. Jaklevick	M. Thompson
T. Koppel	R. Ulrech
J. Krankota	D. Zimmerman

Task G—Core Neutronics, Thermal-Hydraulics, and Safety Analysis

F. Aquino	S. Crick
D. Bender	J. Field
D. Bullington	D. Freeman

Task G (Continued)

P. Greebler	H. Pfefferlen
G. Gyorey	P. Pluta
T. Hersum	L. Price
D. Hines	R. Protsik
B. Hutchins	J. Stephen
M. Kelley	R. Stuart
R. Linford	R. Thompson
W. Meinhardt	J. Turner

Task L—Cladding Development

F. Acencio	J. Jaklevick
E. Aitken	E. Kurtz
C. Barrett	T. Lauritzen
R. Blood	A. Montes
H. Busboom	R. Peterson
F. Comprelli	H. Rosenbaum
W. Cowden	R. Smith
E. Darlin	W. Wagner
K. Evans	D. Weistein
R. Eymann	A. Withop
G. Ferguson	D. Zimmerman
D. Hincs	

DISTRIBUTION

	<u>Copies</u>		<u>Copies</u>
Director, Contracts Division U. S. Atomic Energy Commission San Francisco Operations Office 2111 Bancroft Way Berkeley, California 94704	2	J. A. Lieberman Assistant Director for Nuclear Safety	1
Chief, California Patent Group U. S. Atomic Energy Commission San Francisco Operations Office P. O. Box 808 Livermore, California 94551	1	J. W. Crawford Assistant Director for Plant Engineering	1
RDT Site Office U. S. Atomic Energy Commission c/o General Electric Company 310 De Guigne Drive Sunnyvale, California Attn: J. V. Levy	2	R. H. Steele Division of Naval Reactors	1
Division of Reactor Development and Technology U. S. Atomic Energy Commission Washington, D. C. 20545		RDT Senior Site Representative U. S. Atomic Energy Commission P. O. Box 2325 San Diego, California 92112	1
E. E. Sinclair Asst. Director for Reactor Technology (Attn: J. M. Simmons and I. Zartman)	2	RDT Senior Site Representative U. S. Atomic Energy Commission Canoga Park Area Office P. O. Box 591 Canoga Park, California 91305	1
Attn: Assistant Director for Reactor Engineering	2	RDT Senior Site Representative U. S. Atomic Energy Commission Argonne National Laboratory 9800 South Cass Avenue Argonne, Illinois 60439	1
M. J. Whitman, Assistant Director for Program Analysis	1	RDT Site Office U. S. Atomic Energy Commission Atomic Power Development Asso- ciates, Inc. 1911 First Street Detroit, Michigan 48226	1
A. Giambusso, Assistant Director for Project Management (Attn: R. Sweek, G. Wensch, J. Morabito)	3	RDT Senior Site Representative Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee 37830	1

	<u>Copies</u>		<u>Copies</u>
RDT Site Office U. S. Atomic Energy Commission Combustion Engineering, Inc. P. O. Box 500 Windsor, Connecticut 06095	1	Westinghouse Electric Corporation Bettis Atomic Power Laboratory P. O. Box 79 West Mifflin, Pennsylvania Attn: E. J. Kreh	1
RDT Senior Site Representative U. S. Atomic Energy Commission P. O. Box 550 Richland, Washington 99352	1	Oak Ridge National Laboratory P. O. Box X Oak Ridge Tennessee 37830 Attn: J. E. Cunningham	1
RDT Site Representative USAEC Post Office Box 2108 Idaho Falls, Idaho 83401	1	Brookhaven National Laboratory Upton, New York 11973 Attn: O. E. Dwyer	1
RDT Site Representative U. S. Atomic Energy Commission United Nuclear Corporation Grasslands Road Elmsford, New York 10523		Battelle Memorial Institute Columbus Laboratories 505 King Avenue Columbus, Ohio 43201 Attn: D. Keller	1
Attn: M. Napack	1		
A. Strasser	1	Dow Chemical Company Rocky Flats Division P. O. Box 888 Golden, Colorado 80401 Attn: R. D. Forest	1
LMFBR Program Office Argonne National Laboratory 9800 South Cass Avenue Argonne, Illinois 60439	1		
Attn: A. Amorosi, Director L. R. Kelman	1	Scientific Representative U. S. Atomic Energy Commission American Embassy APO San Francisco, California 96503	1
Argonne National Laboratory 9800 South Cass Avenue Argonne, Illinois 60439			
Attn: J. H. Kittel	3	U. S. AEC Scientific Representative	1
R. C. Vogel	1	United States Embassy Paris, France	
Pacific Northwest Laboratory P. O. Box 999 Richland, Washington 99352		Senior U. S. AEC Representative U. S. Mission to the European Communities United States Embassy Brussels, Belgium	1
Attn: E. A. Eschbach	1		
E. A. Evans	1		
Knolls Atomic Power Laboratory P. O. Box 1072 Schenectady, New York 12301 Attn: W. Cashin	1	Division of Technical Information Extension U. S. Atomic Energy Commission P. O. Box 62 Oak Ridge, Tennessee 37831	50
University of California Lawrence Radiation Laboratory P. O. Box 808 Livermore, California 94551 Attn: A. Rothman	1	Dr. John C. Woodhouse 1 Guest Lane Wilmington, Delaware 19809 Power Reactor Development Corp. 1911 First Street Detroit, Michigan 48226 Attn: W. J. McCarthy	1
Los Alamos Scientific Laboratory P. O. Box 1663 Los Alamos, New Mexico 87544		Argonne National Laboratory	1
Attn: R. D. Baker	1	Idaho Division	
D. B. Hall	1	P. O. Box 1096	
W. R. Wykoff	1	Idaho Falls, Idaho 83401	
Reports Librarian	1	Attn: F. A. Thalgot	

	<u>Copies</u>		<u>Copies</u>
Detroit Edison Company 1911 First Street Detroit, Michigan 48226 Attn: A. S. Griswold	1	Nuclear Materials & Equipment Corporation Plutonium Laboratory Leechburg, Pennsylvania 15656 Attn: William J. Ross	1
Atomic Power Development Asso. Associates 1911 First Street Detroit, Michigan 48226 Attn: A. A. Shoudy	1	Mr. L. W. Fromm, Manager 1000 MWe LMFBR Follow-On Study Project Building 208 Argonne National Laboratory 9800 South Cass Avenue Argonne, Illinois 60439	1
Combustion Engineering, Inc. Nuclear Division Prospect Hill Road Windsor, Connecticut 06095 Attn: W. P. Chernock	1	Mr. C. A. Anderson, Project Manager, 1000 MWe LMFBR Follow-On Study Westinghouse Electric Corporation Advanced Reactors Division Waltz Mill Site P. O. Box 158 Madison, Pennsylvania 15663	1
The Babcock & Wilcox Company Atomic Energy Division Lynchburg, Virginia 24501 Attn: H. S. Barringer S. H. Esleeck M. W. Croft	1 1 1	U. S. AEC Scientific Representative United States Embassy London, England	1
General Atomic, A Division of General Dynamics Corporation P. O. Box 608 San Diego, California 92112 Attn: D. B. Coburn	1	S. J. Stachura Villa Plein Aél Quartier Roussier Aix-en-Provence 13 - France	1
The Babcock & Wilcox Company Alliance Research Center Alliance, Ohio 44601 Attn: D. Koch	1	FFTF Project P. O. Box 220 Richland, Washington 99352 Attn: Configuration and Data Management	8
Westinghouse Electric Corporation Advance Reactors Division P. O. Box 217 Cheswick, Pennsylvania 15024 Attn: W. E. Ray	1	W. P. Staker, Project Manager 1000 MWe LMFBR Follow-On Study Combustion Engineering, Inc. P. O. Box 500 Windsor, Connecticut 06095	2
		S. Visner Nuclear Division Combustion Engineering, Inc. P. O. Box 500 Windsor, Connecticut 06095	2