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**SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR
DEVELOPMENT PROGRAM
TWENTY-SIXTH QUARTERLY REPORT
FEBRUARY-APRIL 1968**

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

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SODIUM-COOLED REACTORS
FAST CERAMIC REACTOR DEVELOPMENT PROGRAM

Twenty-Sixth Quarterly Report
January—April 1968

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Prepared for the
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SECTION I

INTRODUCTION

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

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

Progress during the period November 1, 1967 to April 30, 1968 on the currently active tasks of this program is described in subsequent sections.

This is the twenty-sixth 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—85, from July 1959 through April 1968;

GEAP-3888	FCR Development Program - First Quarterly Report, October—December 1961.	GEAP-4158	FCR Development Program - Fifth Quarterly Report, October—December 1962.
GEAP-3957	FCR Development Program - Second Quarterly Report, January—March 1962.	GEAP-4214	FCR Development Program - Sixth Quarterly Report, January—March 1963.
GEAP-3981	FCR Development Program - Third Quarterly Report, April—June 1962.	GEAP-4300	FCR Development Program - Seventh Quarterly Report, April—June 1963.
GEAP-4080	FCR Development Program - Fourth Quarterly Report, July—September 1962.	GEAP-4382	FCR Development Program - Eighth Quarterly Report, July—September 1963.
		GEAP-4480	FCR Development Program - Ninth Quarterly Report, October—December 1963.
		GEAP-4601	FCR Development Program - Tenth Quarterly Report, January—March 1964.
		GEAP-4640	FCR Development Program - Eleventh Quarterly Report, April—June 1964.
		GEAP-4723	FCR Development Program - Twelfth Quarterly Report, July—September 1964.
		GEAP-4795	FCR Development Program - Thirteenth Quarterly Report, October 1964—January 1965.
		GEAP-4860	FCR Development Program - Fourteenth Quarterly Report, February—April 1965.
		GEAP-4931	FCR Development Program - Fifteenth Quarterly Report, May—July 1965.
		GEAP-4982	FCR Development Program - Sixteenth Quarterly Report, August—October 1965.
		GEAP-5098	FCR Development Program - Seventeenth Quarterly Report, November 1965—January 1966.
		GEAP-5158	FCR Development Program - Eighteenth Quarterly Report, February—April 1966.
		GEAP-5198	FCR Development Program - Nineteenth Quarterly Report, May—July 1966.
		GEAP-5292	FCR Development Program - Twentieth Quarterly Report, August—October 1966.
		GEAP-5433	FCR Development Program - Twenty-First Quarterly Report, November 1966—January 1967.
		GEAP-5491	FCR Development Program - Twenty-Second Quarterly Report, February—April 1967.

- GEAP-5522 FCR Development Program - Twenty-Third Quarterly Report, May-July 1967.
- GEAP-5541 FCR Development Program - Twenty-Fourth Quarterly Report, August-October 1967.
- GEAP-5584 FCR Development Program- Twenty-Fifth Quarterly Report November 1967-January 1968.

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

In addition, the following topical reports have been issued:

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- 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;
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- 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 PuO_2-UO_2 Fuel; G. D. Collins (January 24, 1962).
- GEAP-3833 The Post-Irradiation Examination of a PuO_2-UO_2 Fast Reactor Fuel; J. M. Gerhart (November 1961).
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Part I - Technical Considerations; K. M. Horst and B. A. Hutchins (February 15, 1962).
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- GEAP-3885 Experimental Fast Ceramic Reactor Design, Status Report as of October 31, 1961; Edited by K. M. Horst (April 24, 1962).
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- GEAP-4028 A Fuel Reprocessing Plant for Fast Ceramic Reactors; H. W. Alter (February 1, 1962).
- GEAP-4058 Analytical Studies of Transient Effects in Fast Reactor Fuels; R. B. Osborn and D. B. Sherer (August 1962).
- GEAP-4090 FORE - A Computational Program for the Analysis of Fast Reactor Excursions; P. Greebler and D. B. Sherer (October 1962).
- GEAP-4092 Doppler Calculations for Large Fast Ceramic Reactors, Effects of Improved Methods and Recent Cross Section Information; P. Greebler and E. Goldman (December 1962).
- GEAP-4130 Experimental Studies of Transient Effects in Fast Reactor Fuels. Series I, UO_2 Irradiations; J. H. Field (November 15, 1962).
- GEAP-4226 Conceptual Mechanical Design of 565 MWe Fast Ceramic Reactor; A. G. Silvester (April 1963).
- GEAP-4271 Measurement of Oxygen-to-Metal Ratio in Solid Solutions of UO_2 and PuO_2 ; W. L. Lyon (May 31, 1963).
- GEAP-4283 Experimental Studies of Sodium Logging in FCR Fuels; G. L. O'Neill, P. E. Novak, M. L. Johnson, and W. E. Baily (September 1963).
- GEAP-4350 Fast Reactor Meltdown Excursions - Analysis and Computer Program FARM I; B. C. Luci, D. R. Riley, and N. F. Friedman (December 5, 1963).

- GEAP-4410 Calculations of Doppler and Sodium Reactivity Effects; P. Greebler (November 1963).
- GEAP-4414 Reactor Safety and Fuel Cycle Economics, K. P. Cohen (November 1963).
- GEAP-4418 Liquid Metal Fast Breeder Reactor Design Study, 1000 MWe PuO_2 - UO_2 Fueled Plant, Vols. I and II; M. J. McNelly, et al. (January 1964).
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- 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).
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SECTION II

SUMMARY

2.1 TASK B—DEFECTED FUEL BEHAVIOR

Analysis and review with GETR/ETR/MTR personnel indicate that the forced convection capsule (FCC) concept is suitable for use as a flux-filtered experiment in the ETR and GETR cores, and as a nonfiltered experiment in the GETR pool and the ETR and MTR reflectors.

The design effort on the small FCC has been focused on a capsule with three-pin testing capability. The first test (B9A) will utilize one fueled pin and two dummy pins.

Design analysis was continued on the seven-pin FCC test B9B (center pin designed to rupture as it first approaches full power). Bench-scale tests indicate it is feasible to rupture cold-worked cladding (via differential thermal expansion) by cutting a longitudinal groove in the outside surface and sizing the fuel-cladding gap so that a thermal expansion differential of approximately 0.001 inch is generated between the fuel and cladding.

Irradiation of sodium/fuel/cladding compatibility capsule B8C was initiated. The capsule is operating at a peak power of 21 kW/ft and has accumulated an exposure of 19,000 MWd/T on the top pin and 9,500 MWd/T on the bottom pin.

Sodium-bonded fuel pin B8D-1, containing low (85% TD) and high (94% TD) density, hyperstoichiometric mixed-oxide fuel (O/M 2.03) showed cladding diameter increase of up to 0.005 inch over the low density pellets after heating out-of-pile to 730°F for 3-1/2 hours. The pin was refabricated using UO₂-40% PuO₂ with O/M = 1.97.

Results from vented pin B4D (47,500 MWd/Te) indicate that significant amounts of Cs-137 can be volatilized out through the vent if the temperature of the top blanket is sufficiently high.

Examination of fuel pin B3C (45,000 MWd/Te), which operated with cladding temperatures of 1325°—1350°F, indicates that

cladding degradation from fuel- (and possibly fission product) cladding reactions is more affected by increasing cladding temperature than by the presence of sodium.

The preliminary design was completed on a miniature out-of-pile sodium loop to study the distribution of dissolved and eroded fuel from defected, irradiated, mixed-oxide fuel in flowing sodium.

2.2 TASK C—SAFETY EXPERIMENTATION

Post-irradiation examination of failed specimen C4A (90.4% smeared density pelleted fuel) revealed that: (1) molten fuel had contacted the inner surface of the specimen cladding at one end of the fuel pin for a short period (< 3 seconds) without causing cladding failure at that location, (2) at another location in the fuel pin, melting of the cladding had occurred because of the contacting of expelled molten fuel with the outer surface of the cladding, and (3) a resolidified fuel formation in the lower portion of the specimen presents the possibility that fuel relocation can occur without requiring the driving force provided by transiently released fission gas.

Destructive examination continued on the two transiently-irradiated specimens (C4I and C4J) of powder fuel.

The hazards analysis for the axial fuel relocation experiments (pre-irradiated capsules C5A and C5B) was forwarded to Idaho for review.

The design parameters and fuel pin fabrication specifications for the defected or sodium-bonded experiment (series C6) were finalized.

The design of the EM flowmeter, intended for use in Mockup 1A, Mockup 2, and the multipin-forced convection capsule, was completed. Pressure transducers were

ordered, and design of the electrically heated fuel simulators for Mockup 1A was completed. The initial steady-state thermal analysis of Mockup 1A has been completed and detailed design is under way.

2.3 TASK E—FUEL PERFORMANCE EVALUATION

A value of 42.8 ± 6.4 watts/cm was calculated for the integral thermal conductivity (from 1020°C to melting temperature) at 77,000 MWd/Te burnup (capsule E1F).

Destructive examination of the two fuel pins from capsule E2J (high burnup comparison of solid and annular pellet fuels) continued. Neutron radiographs and gamma scans indicated no unusual fuel movement or gamma activity in either fuel pin.

Analysis of microcores from two high burnup (> 100,000 MWd/Te) fuel pins (E2C and E2D) showed an increase in concentration at the fuel periphery for Zr-95, Ce-144, Cs-137, Ru-106, and Pu-239.

Capsules E5C, E5D, E5E (high burnup comparison of solid pellet, powder, and annular pellet fuels) continued their irradiation at 19.5 kW/ft to a present burnup of 53,000 MWd/Te.

Capsule E5F (in-pile axial fuel thermal expansion) continued irradiation to 12,400 MWd/Te after a short delay from initial calibration difficulties.

Fabrication of capsules E2P and E2Q (effect of radial restraint on fuel swelling) was completed and their irradiation initiated. When thermocouple readings indicated that design power (15 kW/ft) could not be obtained in either capsule, the capsules were removed from the reactor and placed in storage pending resolution of the anomalous power indications.

Fabrication was completed of capsule E1AA (in-pile thermal conductivity).

Preliminary design drawings were initiated for capsule series E1J (effect of fuel restructuring on smeared thermal conductivity).

The melting point of $U_{0.75}Pu_{0.25}O_{2.00}$ irradiated to 250,000 MWd/Te was observed to decrease $130^{\circ}C$ to $\sim 2640^{\circ}C$.

Lattice parameter measurements of irradiated fuel specimens with burnups from 14,000 to 255,000 MWd/Te were completed. These measurements indicated that crystallinity was retained at high burnups with lattice disruption being more severe at 50,000 MWd/Te in a fast flux, compared to 255,000 MWd/Te in a thermal flux.

Out-of-pile experiments of oxygen migration in a thermal gradient were completed on specimens of UO_2 - PuO_2 with initial stoichiometries of 1.93, 1.98, and 2.00. The tests indicated that O/M profiles in mixed-oxide fuel can be predicted under steady-state operation if the temperature profile is known, and that steady-state O/M profiles are probably established in times very short compared to the lifetime of the fuel.

Initial creep and plasticity data were obtained at 1500° – $1660^{\circ}C$ on 95% dense UO_2 with a grain size of 12–18 microns and an O/U ratio of 2.008 ± 0.005 . The data can be described by an equation of the form $\dot{\epsilon} = A\sigma m_e (-Q/RT)$, where the activation energy (Q) is 97.4 kcal/mole.

2.4 TASK F—INTEGRAL FAST FLUX IRRADIATIONS

The irradiation of Groups 0, 3B, 6, 7, and 8A was temporarily discontinued when EBR-II released fission gas periodically during run No. 27. Attempts to isolate the source of fission gas resulted in the removal of the subassemblies containing these experiments from the EBR-II core. The accumulated

burnups to date on Groups 0, 2, 3B, 4, 6, 7, and 8A are calculated to be 28,300, 74,000, 8,900, 35,800, 33,500, 28,700, and 14,500 MWd/Te, respectively.

Irradiation of Group 2 in subassembly XG05 was terminated after receiving a peak burnup calculated to be 74,000 MWd/Te. This subassembly was removed from the core as a suspected leaker; however, fission gas activity was also detected in the EBR-II cover gas with S/A XG05 out of the core. The capsules were removed from the subassembly at the EBR-II site and neutron radiographed using the TREAT reactor for a neutron source. The results indicate that eight fuel pins are intact and that one (F-20) failed, but is not the source of fission gas in EBR-II.

The Group 4 capsules in subassembly XO11 which were removed from the reactor last summer are scheduled to be incorporated into a new subassembly (XO39) for reinsertion into EBR-II (run No. 29) for an additional 15,000 MWd/Te burnup.

The nineteen capsules in Group 8B and four Group 7 capsules were delivered to EBR-II and passed pre-irradiation inspection in preparation for irradiation testing in the EBR-II.

Examination of failed fuel pin F2F (58,000 MWd/Te) continued. A fission gas release of 33% was measured, which is lower than that observed (50%–60%) in the unfailed pins from S/A XG06 which operated at comparable power and burnup levels. A crack extending through the thickness of the cladding to the interface between the cladding and the bottom end cap was observed which may indicate a possible location and initiating mode of pin failure.

Continued examination of failed fuel pin F4F (34,000 MWd/Te) revealed localized zones of melted cladding, but no indications of significant fuel-sodium reaction, consistent with expected behavior from gas bubbles passing in the pin-capsule annulus.

Flow testing was initiated on a Mark A subassembly incorporating cold sodium bypass flow for possible use as the test vehicle for the Group F5 elevated cladding temperature tests. Fabrication of 14 fuel pins was completed and assembly of the remaining two is proceeding.

The revised hazards analysis for the first unencapsulated fuel test, Group 3A, was submitted to the EBR-II staff for their review.

Fabrication of the unencapsulated fuel pins for the 37-pin Group 9A is continuing. Fabrication of Group 9B has been deferred until completion of Groups 9C and 9D, the design of which was completed and fabrication initiated.

The effect of fully-enriched oxide fuel in Rows 1, 2, and 3 on the EBR-II power coefficient of reactivity was calculated, suggesting that even with cladding-controlled axial expansion of the oxide fuel, power coefficients comparable to the all-metal fuel core were obtained.

Hydraulic flow tests were initiated at water flow velocities of 4 to 14 ft/sec, and pitch/diameter ratios of 1.1 and 1.3. Some skewing of the velocity profile was observed from small dimensional asymmetries.

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

Modified cross sections for iron, which show more detail in total and elastic scattering than those presently on ENDF/B, have been sent to Brookhaven National Laboratory for distribution to members of the Cross Section Evaluation Working Group.

Updated cross sections for U-238 and Pu-239 have been incorporated into the CROFO3 code, and are being used for current LMFBR computations.

Checkout of the ENDRUN code has been completed through the smooth data averaging and scattering matrix generation. The resonances-averaging routines are being modified to provide increased accuracy.

The study of the sensitivity of fast reactor power costs to nuclear data uncertainties has been extended to include possible effects on LMFBR design and the total reactor economics.

A study of group-constant condensation from 60 to 16 groups, as used with the BISYN code, indicates that with careful selection of the 16-group structure, adequate accuracy for scoping and most design calculation is achieved in k_{eff} , breeding ratio, safety coefficients, and power distribution.

Transport and diffusion theory calculations have been extended, for two fast reactor models, to include anisotropic scattering. The results indicate that the errors in diffusion theory calculations due to inadequate treatment of flux anisotropy and scattering anisotropy are nearly independent.

Fuel cycle costs were reestimated for sodium-bonded fuel assuming ~24 and 30 kW/ft linear power. Using the RAPFU code to refine the previous calculations resulted in a somewhat greater indicated economic incentive (~0.05 mill/kWh) than previously reported (0.02 mill/kWh).

A nodal computer model has been set up, using the THTD system, to study the progress of fuel failure initiating from a local blockage. The model has been checked under steady-state conditions against VELVET slug flow results, and appears satisfactory.

An examination of the thermal effects of spacers on the pins on the outer edge of a bundle has been initiated.

2.6 TASK L—CLADDING DEVELOPMENT

Irradiation of L-2, L-6, L-10 and L-14 materials capsules is continuing in the EBR-II. The total neutron fluences, as of April 30, 1968, are 4.6, 2.0, 1.3, and 0.7×10^{22} n/cm², respectively.

Mechanical evaluation at 1300°F of irradiated ($\sim 4 \times 10^{21}$ nvt, >1 MeV) specimens indicated that carbide-agglomerated 321 and 347 have larger uniform elongations, at comparable fluences, than mill-annealed structures. Carbide-agglomerated 316 and Incoloy 800 have lower uniform elongations than comparably-exposed mill-annealed structures. Carbide-agglomerated 316 and Incoloy 800 indicated an increase in total elongation with decreasing strain rate, while the mill-annealed structures showed the reverse. Carbide-agglomerated 304 showed anomalous strain behavior, the uniform strain increasing (from 8 to 13%) with increasing fluence in the range 0.8 to 3×10^{21} nvt ($E > 1$ MeV).

From the stress-strain history of materials capsule L-2Q (Type 304), it was determined that, due to the localized deformation resulting from in-reactor biaxial creep, the hoop stress was a function of the deformation within each test section.

The successful preparation of thin wafers of highly radioactive mixed oxide fuel for electron microprobe analysis was demonstrated. An activity survey of the shielded microprobe, performed with a cobalt-60 source in the specimen chamber, demonstrated the acceptable gamma attenuation capability of the instrument.

An analytical study of the fuel-cladding system has shown the dependence of compatibility on the oxygen activity of the system.

Evaluation of the barreling mandrel-ring ductility test has been completed demonstrating the reproducibility of this technique, with small samples typical of fuel cladding sections.

In-depth characterization of the five cladding alloys presently being irradiated in EBR-II materials capsules has been initiated. This program will define the effects of thermal exposure on tensile properties and microstructure of all five alloys at thermal conditions similar to those being experienced by the irradiated specimens.

An experimental study to evaluate the effect various levels of surface defects on the effective mechanical properties of tubing at elevated temperatures has been initiated.

TABLE 2-1

TASK B—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

		Design Parameters							Operating Conditions						
		Fuel													
						Composition									
Designation	Purpose and Major Parameters	Diameter (inch)	Length (inch)	Density (Smear) (% TD)	Fuel-Cladding Gap (Diameter inch)	PuO ₂ %	O/M	Cladding Material	Irradiation Facility	Cladding Surface Temperature (°F)	Peak Rating (kW/ft)	Burnup (MWd/Te)	Status		
B3C	Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	88	0.003	25	2.00	Type 347 stainless steel	V-Raft GETR	1250	21	45,000	Examination completed.		
B3D	Vented Fuel. Power profile simulates fuel in central core position. Natural convection cooling design.	0.217	20	85	0.006	25	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 completed.		
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 completed.		
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	19,000 9,500	Irradiation in progress.		
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 80-83	0.0065 0.0065 0.0065	40 25 25	1.97 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

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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	C	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 -23% 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	C	I	Axial restraint similar to C4A. Test transient is intended to be similar to that of C4D.
C4J	Identical to C4I	C		P	C	C	C	I	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

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TABLE 2-3

TASK E—SUMMARY OF CURRENT MIXED OXIDE CAPSULE IRRADIATIONS

		Design Parameters							Operating Conditions					Status
		Fuel						Irradiation Facility	Cladding Surface Temperature (°F)	Peak Power (kW/ft)	Burnup (MWd/Te)			
		Diameter (inch)	Fuel Length (inch)	Density, % TD		Composition								
				Pellet	Smeared	PuO ₂ %	O/M	Cladding Material						
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 complete.	
E1B-1	Gap conductivity. Four pins with diametral gaps ranging from 2 to 10 mils.	0.218	1.25	95.9	92.5	20	1.98	Type 347 stainless steel	GETR	—	20	500	Report in preparation.	
2		0.218	1.25	94.2	94.2	20	1.98	GETR	974	20	500			
3		0.218	1.25	96.1	85.0	20	1.98	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	0.218	1.9	94.7	94.5	20	2.00	Type 347 stainless steel	MTR	875	20	85,000	Data analysis in progress.	
E1F	high burnup and rabbit test E1F.	0.218	1.9	94.6	94.8	20	2.00	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				
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.	0.218	4.9	86.8	84.3	25	2.00	Type 347 stainless steel	GETR	1103	27	2,000	Report in review.	
2		0.218	4.9	84.1	84.1	25	2.00	GETR	1103	27	2,000			
E5B-1	Powder and pellet fuel comparison at high burnup.	0.218	4.9	85.8	84.1	25	2.00	Type 347 stainless steel	GETR	1103	20	100,000	Destructive examination in progress.	
2		0.218	4.9	84.1	84.1	25	2.00	GETR	1103	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	1053	19.5	53,000	Irradiation in progress.	
E5D-1	Powder, pellet, and annular fuel comparison at high burnup.	Powder	16.0	81.8	81.8	25	1.97	Type 316 stainless steel	GETR	1053	19.5	53,000	Irradiation in progress.	
E5E-1	Powder, pellet, and annular fuel comparison at high burnup.	0.215 (Annular)	16.0	90.5	83.3	25	1.96	Type 316 stainless steel	GETR	1053	19.5	53,000	Irradiation in progress.	
E5F	Linear fuel movement versus burnup.	0.214	10.0	92.0	87.1	25	1.97	Type 316 stainless steel	GETR	850	15	17,000	Irradiation in progress.	
E2P	Effect of radial restraint on fuel swelling.	0.219	2.0	92.0	91.0	25	1.98	Type 316 stainless steel	GETR	1160	15	500	Irradiation in progress.	
E2Q	Effect of radial restraint on fuel swelling.	0.229	2.0	92.0	91.0	25	1.98	Type 316 stainless steel	GETR	1160	15	500	Irradiation in progress.	

TABLE 2-4

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

April 30, 1968

Group	Purpose and Major Parameters	Row	Number		Power (kW/ft)	Burnup (1000's MWd/Te)		Irradiation	
			SA	Pins		Target	Calculated	Finish*	Status**
F-0	Low power swelling rate of high density fuel at various burnup levels.	7	XG02	1	5-6	40	28.3	March 1969	In reactor.
			XG03	2		60	28.3	June 1970	
			XG04	2		100	28.3	March 1973	
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. Discharge from core, test terminated.
			XG05	9		75	74	April 1968	
F-3A	Bare-pin irradiations to evaluate fuel cladding and cladding coolant interactions.	5	1	16	15-18	50	—	August 1969	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	8.9	February 1969, February 1970, and March 1971	In reactor.
F-4	Investigate low density fuel behavior at heat ratings.	2	XO11	9	15-18	50	35.8	August 1968	Discharged from Core; SA released gas.
F-5	Fuel performance at FCR peak cladding temperatures.	4	1	16	15-16	100	—	January 1971	Fuel fabrication in process.
F-6	Fuel-cladding interactions under accelerated swelling.	7	XO10	4	7-9	50	33.5	April 1969	In reactor.
F-7	Pre-irradiated axially strained fuel prior to transient test. Main variable is fuel density.	4	XO15	4	12-15	50	28.7	January 1969	In reactor. Fuel at EBR-II Site.
			1	4	12-15	50	—	October 1969	
F-8A	Evaluate decrease in density on swelling rate.	6	XO19	7	6-9	50	14.5	March 1970	In reactor.
			XO20	9		100	14.5	October 1972	
F-8B	Effect of void deployment method on fuel cladding interactions.	7	XO36	19	6-9	50	—	November 1970	Fuel at EBR-II Site.
F-9A	Irradiation bare pins to high burnup.	4	1	37	11-16	100	—	January 1971	Fuel fabrication in process.
F-9B	Irradiation bare pins to high burnup.	4	1	37	10-13	100	—	January 1971	Fuel fabrication in process.

* Based on EBR-II operation at 45 MW as of April 1, 1968 and at 50% load factor.

** All experimental S/A have been removed from EBR-II core in attempt to isolate source of fission gas.

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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 1970
L-6	2—XO18	9	One each of Types 321 and 347 stainless steels; two each of Type 304 stainless steel and Incoloy 800; three of Type 316 stainless steel.	63	126 ⁽¹⁾	1.00	0.20	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 300.	70	60 ⁽¹⁾	0.20	0.13	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	0.07	0.05	December 1968
L-15	4—XA03	4	All five alloys.	None	288	0.20	-	-	May 1969
MT-3 ⁽⁵⁾ and 4		2	Thirty-two Incoloy-800, 32 Inconel-625, and 32 Hastelloy-X tens. samples.	None	96	1.00	0.39	0.29	December 1971

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

SECTION III

TASK B—DEFECTED FUEL BEHAVIOR

3.1 GENERAL

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

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

The technologies of defected, vented, and sodium-bonded fuels are closely related. Although directed primarily at studying the behavior of defected fuel, Task B also has in the past included development of vented and sodium-bonded fuels. Vented pins are used in prospective reference designs for future 300 MW and 1000 MW plants, but not for FFTF. Vented fuel testing has been terminated at AEC request.

3.2 DEFECTED FUEL TESTING
(SERIES B8 AND B9)3.2.1 Forced Convection Capsule

Analysis and review with GETR/ETR/MTR personnel indicate that the forced convection capsule (FCC) concept is suitable for use as a flux filtered experiment in the GETR and ETR cores and as a nonfiltered experiment

in the GETR pool and the ETR and MTR reflectors. No instrumented experiments are permitted in the MTR core. Testing in any of these reactors requires a hazard analysis examining the consequences of discharging the full capsule inventory of hot sodium into the reactor water. Such tests are presently covered by license in GETR for capsule sodium inventories of up to 1 kg. For inventories greater than 1 kg, all three reactors require a detailed hazards analysis backed up by further experimental studies.

A series of tests is presently being designed and fabricated to provide information for hazards analysis of the large FCC in a water-cooled test reactor. These tests are designated SODA POP and consist of injecting molten sodium at up to 1200°F from a ruptured FCC mockup into water which simulates the core of a reactor. The objective of the tests is to determine the extent of test reactor core damage occurring from the sodium-water reaction. In addition, possible methods of reducing the severity and consequences of such a failure are being studied.

Design and fabrication continued for the sodium remelt heater and flow coastdown tests for the forced convection capsule design. This is discussed in Task C.

The design effort on the small FCC has been focused on the design of a capsule with three-pin testing capability. A study was completed showing that three-pin testing appears feasible within present license limits in the GETR pool. However, power skewing from thermal flux depression in the pool would be severe for the three-pin test if all three are fueled. A three-pin test configuration with two dummy pins eliminates the power skewing in the thermal

flux. Figure 3-1 compares the cross sections of a three-pin and single-pin arrangement in the small FCC. In both cases, the sodium inventory is held below 1 kg. The coolant velocity through the test section of the three-pin FCC would be limited to approximately 12 ft/sec when using the same pump and helix as for the single pin FCC. Figure 3-2 shows this pump and characteristic capsule performance curve. Investigations are presently under way to increase the pump capacity by modifying the pump helix.

The design of an EM flowmeter for the small FCC is in progress.

A pressure transducer is required for flow reduction tests in Mockup 1A and the large FCC's used for flow reduction testing. A survey of available transducers is in progress.

Knowledge and control of the oxygen concentrations in the sodium coolant are required for a comprehensive analysis of the results of defected fuel behavior. Sampling and analytical procedures for pre- and post-irradiation oxygen analysis of the sodium are being evaluated. Design was initiated on the sodium-cleanup loop to be used in filling the FCC capsules. Dynamic cold trapping in an external loop will be used to purify the sodium initially in the FCC. In addition, a diffusion cold trap is being considered for oxide control during capsule operation.

3.2.2 Capsules B9A and B9B

Fuel defecting techniques were investigated during the report period for use in the forced convection B9A capsule experiment. The defect mechanisms being evaluated include: (a) mechanical needle on a plunger, (b) low melting or soluble plug in a machined slot or hole, (c) notched cladding stressed to rupture by differential thermal

expansion of fuel, and (d) pneumatic pressurization of notched cladding.

Tests were performed in which both annealed and fully cold-worked Type 316 stainless steel tubing specimens were longitudinally notched to a depth of 0.012 inch on the outside diameter surface (wall thickness 0.015 inch). These specimens were then stressed either by (1) placing a cold mandrel snugly inside of the heated tubular specimen and letting an interference fit develop as both pieces reached the same temperature, or (2) ballizing with progressively increasing ball diameters until failure occurred. Results of these tests indicated that notched, annealed cladding did not fail when subjected to the small strains expected from differential thermal expansion of fuel and cladding. Figure 3-3 shows a transverse view of a notched, annealed specimen after being stressed by shrink fitting over a steel plug.

Figure 3-4 shows the failure of a notched, cold-worked specimen after being ballized 0.0008 inch and indicates that cold-worked cladding may be required to fail fuel specimens with the small strains expected from differential fuel/cladding thermal expansion.

Preliminary design of the first seven-pin test to investigate the effects of defected fuel pin swelling on neighboring pins is in progress. B9B will consist of seven mixed oxide pins in hexagonal array cooled in sodium at typical FCR core temperatures and velocities. The center pin will be designed to rupture as it first approaches full power.

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

Capsule examination was completed this report period with receipt of fission gas and plenum leach sample analyses. Table 3-1 shows the results of the plenum leach sample

analysis. Absolute fission gas release data appears questionable, since significant amounts of oxygen and nitrogen were measured indicating possible in-leaking of air during sampling. The gas release results, assuming no loss of xenon and krypton, are shown in Table 3-2.

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

Irradiation of Capsule B8C to explore fuel/sodium/cladding compatibility as a function of burnup was initiated in March, 1968. The two sodium-bonded mixed oxide fuel pins (O/M = 1.97) of the capsule were operated at a peak power of approximately 21 kW/ft and have accumulated an exposure of 19,000 MWd/T for the top pin and 9,500 MWd/T for the bottom pin (the diameter of the top pin is adjusted to achieve approximately double the burnup of the bottom pin for the same exposure time).

Fuel pins B8D-1, B8D-2, and B8D-3, to be used to evaluate fuel/sodium/cladding compatibility as functions of O/M and density were heat soaked 3-1/2 hours at 730°F to improve the sodium-bond prior to final capsule assembly. Each pin contains

fuel of two densities separated by an insulator pellet. The top four pellets are 85% theoretical density and the bottom three pellets are 94% theoretical density. Fuel pins B8D-2 and B8D-3 containing mixed oxide fuel of 1.97 and 1.94 stoichiometry showed no significant change in cladding diameter after treatment. However, profilometer measurements of fuel pin B8D-1, which contains mixed oxide fuel of O/M = 2.03, showed increases in cladding diameter of up to 0.005 inch over the low density pellets. This change corresponds to approximately 14% swelling of the fuel in this region and may be indicative of extensive sodium-fuel reaction.

Figure 3-5 shows the pre- and post-heat soak profilometer traces made on this fuel pin. The maximum change in cladding diameter occurred at the bottom of the low density fuel column. Since a comparable or even greater volumetric expansion of the fuel would be expected at the top of the fuel column where the sodium is most accessible for reaction, the increase in cladding diameter from the top to the bottom of the low density fuel column appears to reflect the degree of axial restraint experienced by each of the successive fuel pellets.

TABLE 3-1
FISSION PRODUCT RELEASE TO PLENUM OF SODIUM-BONDED FUEL

	B8B-1	B8B-2
O/M	2.00	1.93
kW/ft	14	14
<u>% Fission Product Release to Plenum</u>		
Cs-134	1.5*	
Cs-137	0.48*	
Ce-141	1.39×10^{-4}	0.85×10^{-4}
Ce-144	0.67×10^{-5}	1.02×10^{-5}
Ru-103	1.3×10^{-4}	0.7×10^{-4}
Ru-106	0.45×10^{-5}	1.2×10^{-5}
Zr-95	**	0.25×10^{-4}

* Average for both pins.

** None detected.

FIGURE 3-1. COMPARISON OF CROSS-SECTIONAL ARRANGEMENT OF SINGLE- AND THREE-PIN FCC :

FIGURE 3-2. PUMP VERSUS THREE-PIN CHARACTERISTIC FLOW CURVE

FIGURE 3-3. TRANSVERSE VIEW OF NOTCHED, ANNEALED CLADDING SPECIMEN

FIGURE 3-4. VIEW OF NOTCHED, COLD-WORKED CLADDING
SPECIMEN AFTER BALLIZED FAILURE

FIGURE 3-5. PRE- AND POST-HEAT SOAKING PROFILOMETER TRACES OF SODIUM-BONDED MIXED OXIDE FUEL PIN B8D-1

TABLE 3-2

FISSION GAS RELEASE FROM SODIUM-
BONDED FUEL

	<u>B8B-1</u>	<u>B8B-2</u>
O/M	2.00	1.93
kW/ft	14	14
% Fission Gas Release		
Kr	2.9	1.2
Xe	5.0	1.2

Refabrication of fuel pin B8D-1 was initiated and is now complete except for sodium bonding and welding the fill tube covers. The pin was refabricated with the same external design but with fuel containing 40% PuO₂ by weight and 1.97 stoichiometry. This permits direct comparison between fuel pins B8D-1 and B8D-2 (fuel 25% PuO₂ by weight and 1.97 stoichiometry) on the effect of plutonium content of the fuel on the sodium-fuel-cladding compatibility.

3.3 FISSION PRODUCT DISTRIBUTION (SERIES B3 AND B4, MINILoop)

3.3.1 Capsule B4D (Intermediate Burnup, Typical FCR Fuel Length (20 inches), Vented Fuel)

Post-irradiation examination continued with disassembly of the capsule and examination of the vented fuel pin (47,500 MWd/T). Externally, the pin appeared in excellent condition with no measurable swelling or other dimensional changes. Upon sectioning, it was found that the vent had performed successfully and that none of the NaK coolant had entered the fuel region of the pin. Figure 3-6 shows the appearance of the fuel in the peak power region. B4D operated at kW/ft and had a smeared density of 84% theoretical. The fuel morphology is typical for these conditions.

The cladding was generally free of attack except for small localized "stringers," one of which is shown in Figure 3-7. The

maximum cladding inside diameter temperature in this specimen is estimated to have been approximately 1175°—1200°F.

The specific activity of Cs-137 in the NaK coolant of capsule B4D was measured and was found to be greater by a factor of 8-1/2 than the corresponding value for B3F, a similar specimen which was NaK-bonded. When allowance is made for the higher burn-up of B4D, it is found that the release fraction of Cs-137 in the case of B4D was 0.45% compared to the 0.1% observed in specimen B3F.

The unexpectedly high Cs-137 releases from B3F and B4D, therefore, appear caused by two different mechanisms. In B3F, a "swishing" of the NaK in and out through the vent probably took place. Each cycle mechanically carried out some of the highly soluble cesium. The results of B4D, however, must be interpreted in the light of the complete absence of NaK in the fuel pin. In this case the volatilization of the cesium (boiling point = 1238°F) in the blanket, diffusion transport in the vapor phase through the vent (down a concentration and temperature gradient) and subsequent condensation in the NaK may have occurred because of the higher temperature in B4D compared to B3F. In addition, the design interface temperature in B4D was 1550°C compared to 800°C in B3F. ⁽¹⁾

It is unlikely that escape of the 4-minute Xe-137 precursor of Cs-137 occurred directly through the vent and into the coolant by steady-state gaseous diffusion since negligible Cs-137 was found in B4A ⁽²⁾ where the ratio of void volume within the pin to fuel was approximately 40% less than in B4D.

It has been reported in the literature ⁽³⁾ that bursts of fission gas release occur during large decreases in power in UO₂ fuel. The possibility that significant amounts of Xe-137 might have been carried out through

the vent of B4D by such bursts was investigated. Calculations indicate that only 3% of the observed Cs-137 release could be accounted for by this mechanism. This follows from the short half-life of Xe-137, i.e., only small amounts are present within the fuel at any one time.

An explanation for the very low Cs-137 release from B4A can also be made now. The coolant, cladding, and therefore, blanket temperatures were approximately 200°F lower in the short, static capsule design used for B4A compared to natural circulation capsules B4D and B3F in spite of the higher heat rating of the B4A fuel. Volatilization of the Cs-137 did not occur at the lower temperature of the B4A blanket; thus, Cs-137 could not escape through the vent.

3.3.2 Capsule B3C [Medium Burnup, Typical FCR Length (28 inches), Sealed Fuel, Natural Convection Cooled]

The post-irradiation examination of this long, failed specimen (45,000 MWd/T) was completed during the report period. Metallography was performed on the fuel pin. Figure 3-8 shows the appearance of the fuel in the failed region.

The globule of metallic material attached to one edge of the cladding appears to have a high fission product content as indicated by the beta-gamma autoradiograph (Figure 3-9). Since several fission products are known to migrate to the outer periphery of the fuel during steady-state operation (iodine, cesium, and rubidium) these elements could easily alloy with the molten cladding. The dendritic structure of the metallic glob became evident upon etching as shown in Figure 3-10. This indicates that it was completely molten at one time. The metallic appearing ingots in the central region of the fuel also exhibited a high beta-gamma activity and a marked sensitivity to

the 10% oxalic acid etch (compare Figure 3-11 with Figure 3-8).

Curvature of the longitudinal grain boundaries is noticeable adjacent to the cladding rupture (Figure 3-8). This indicates a gross radial movement of the fuel inward with a corresponding change in the heat flow path causing the change of shape in the grains in that region. The inward movement of the fuel may have been caused by forces resulting from formation of a low density fuel/NaK reaction product at the periphery of the fuel after pin failure.

B3C exhibited extensive damage to the fuel cladding inside diameter surface. Figure 3-12 is a transverse view of a typical region before and after etching. The cladding damage consisted of a uniform layer of general attack which is of a spongy nature. Localized regions of more severe degradation were observed opposite cracks in the fuel periphery. Figure 3-13 is a longitudinal view of one of the more severely attacked regions.

B3C was designed and irradiated in 1965 and had a design maximum cladding surface temperature of 1325°—1350°F, higher than any other specimen in this program. It appears that operation at that temperature resulted in fuel-cladding reactions, severe cladding degradation, and eventual failure. Similar severe cladding attack was observed in the failed specimens B8A-2⁽⁴⁾ (gas-bonded) and B3B.⁽⁵⁾ Specimen E5B⁽⁶⁾ showed general cladding attack comparable in depth to that observed in B3C but without the localized zones of more severe attack while specimen E2D⁽⁵⁾ also showed some general attack of the cladding surface but to a lesser degree than the above-mentioned specimens. A case of minor localized cladding attack was discussed above for B4D. The evidence from these specimens indicates that cladding degradation resulting from fuel-cladding reactions is more affected

by increasing cladding temperature than by the presence of sodium.

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

Preliminary design was completed on a minature loop to study out-of-pile the distribution of dissolved and eroded fuel from defected, irradiated, mixed oxide fuel in dynamic sodium. Final loop drawings and procurement of long lead time items has been initiated. The loop consists of the following components: EM pump, flowmeter, Na sample tubes, sample exposure section (including heaters and sodium purification equipment), cooler assembly (fin tube cooler, cowling, blower, and air damper), and dump tank. Figure 3-14 shows the proposed loop arrangement with components identified. All connecting lines are 1/4 inch outside diameter tubes. Table 3-3 lists the major loop design parameters.

The loop will be operated in a steel containment box inside of one of the radiation locks at Vallecitos. The temperature of the sodium passing the fuel will be 1150°F. The sodium will then pass through the cooler assembly where the sodium temperature is reduced to 750°F. The cooler tube will be packed with sections of screen wire containing a thin matting of steel wool. The steel wool and screen wire will provide additional surface area for deposition within the

cooler, as well as breaking up the sodium flow to help promote deposition. After the loop test run, the cooler will be examined to provide data on the temperature range at which fuel and different major fission products deposit in a sodium system.

The oxygen content of the loop sodium will be controlled by hot trapping with zirconium chips prior to insertion of the fuel sample into the loop sodium.

The sodium sample tubes (Figure 3-14) will be used to sample the loop sodium just prior to the insertion of the fuel into the loop sodium and just prior to the termination of the loop test.

The fuel specimen consists of a 3-inch-long section cut from a medium burnup (approximately 50,000 MWd/T) fuel rod (F2U). A defect slot will be cut in the cladding of the fuel sample. Sodium then flows by the defect slot. Metallographic examination will be used to determine the extent of fuel erosion. With a sodium flow rate of only 0.1 gpm, a sodium velocity representative of regular FCR operation is obtained by directing the sodium through a nozzle aimed at the defect slot. Figure 3-15 shows the fuel sample holder and nozzle arrangement. The sample holder is provided with a twist lock to permit remote loading of the fuel into the sample holder.

TABLE 3-3
LOOP DESIGN PARAMETERS

Sodium Maximum Temperature	1150°F
Sodium Minimum Temperature	750°F
Sodium Flow	0.1 gpm
Sodium Purity	5—10 ppm O ₂
Weight Sodium in Loop	~ 750 gm
Velocity of Sodium Passed Fuel	~ 10—15 ft/sec

**FIGURE 3-6 APPEARANCE OF VENTED FUEL SPECIMEN B4D
AFTER 47,500 MWd/T**

CLADDING →

STRINGER →

← FUEL

FIGURE 3-7. METALLIC STRINGER OBSERVED AT CLADDING
INSIDE DIAMETER IN VENTED FUEL SPECIMEN
B4D AFTER 47,000 MWd/T

FIGURE 3-8 APPEARANCE OF B3C FUEL SPECIMEN
IN FAILED REGION AFTER 45,000 MWd/T

FIGURE 3-9. BETA-GAMMA AUTORADIOGRAPHS OF B3C FUEL CROSS SECTION (SHOWN IN FIGURE 3-8)

10% OXALIC ETCH

250X

**FIGURE 3-10. DENSITIC STRUCTURE OF METALLIC GLOB ADHERING TO
EDGE OF CLADDING OF SPECIMEN B3C (SEE FIGURE 3-8)**

FIGURE 3-11. ETCHED APPEARANCE OF CENTRAL INGOT IN SPECIMEN B3C
(SHOWN IN FIGURE 3-8)

UNETCHED

100X

10% OXALIC ETCH

100X

**FIGURE 3-12. ATTACK OF INSIDE DIAMETER CLADDING SURFACE IN
FAILED FUEL PIN B3C AFTER 45,000 MWd/T**

FIGURE 3-13. APPEARANCE OF SEVERLY ATTACKED CLADDING SURFACE
(INSIDE DIAMETER) OF FAILED FUEL PIN B3C

FIGURE 3-14. MINILoop

FIGURE 3-15. MINILoop SAMPLE HOLDER

SECTION IV

TASK C—SAFETY EXPERIMENTATION

4.1 GENERAL

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

During 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 during which specimen C4A (90.4% smeared density pellet fuel) failed was less severe than the irradiation

of the nearly identical specimen C4B (which also failed during transient irradiation).⁽¹⁾ Specimen C4B (91.2% smeared density pellet fuel) suffered cladding failure just below the fuel-column midplane during a transient irradiation which produced 76 to 80% molten fuel volume.^(7, 8) The failure resulted in a dispersal of ~30% of the fuel outside of the cladding, spread over a 5-inch long region.^(1, 8) The initial fuel column length was ~14 inches.

4.2.1 Specimen C4A Analysis

Specimen C4A also suffered cladding failure during a transient irradiation which produced ~56% to 63% molten fuel volume (as determined by measurement of porosity movement and grain growth during the destructive examination). The failure occurred above the fuel-column midplane, and resulted in only ~5% of the fuel being dispersed outside of the cladding, over a 3-inch region.

Figure 4-1 presents a neutron radiograph of the fuel and blanket region of specimen C4A as it is mounted in its inner capsule (the fuel appears as the dark column, its diameter is ~0.220 inch, and its length is ~14.5 inches). It is possible to see the fuel dispersed outside of the cladding, and to identify nine separate void regions along the fuel column length.

Calculations of the fuel temperature history for the C4A irradiation were made prior to the specimen's destructive examination, using the ARGUS transient heat transfer code.⁽⁹⁾ As shown in a previous report,⁽¹⁰⁾ ARGUS gave excellent sodium annulus temperature history correlations, but predicted a ~75% fuel volume melt and a maximum fuel temperature of $6600^{\circ} \pm 1000^{\circ}\text{F}$. The actual

melt volume determined during the destructive examination, ranges from ~56% to 63%. After the examination, the temperature histories for the C4A transient were recalculated, using the THTD transient-heat transfer code.⁽¹¹⁾ These recalculated results are shown in Figures 4-2 and 4-3. In Figure 4-3, THTD predicts a realistic fuel melt radius of 86.3 mils, which corresponds to a fuel volume melt of ~61.5%. The maximum calculated fuel temperature is $5500^{\circ} \pm 600^{\circ} \text{F}$.

The results of a comparison of the pre- and post-transient profilometer readings for the lower 8-1/4 inches of the C4A fuel column (~95% of the fuel column below the failure zone, and 58% of the overall fuel column length) indicate an average permanent cladding deformation slightly over 1.1 mils (~0.46%). In addition, there is an overlay of an additional 0.2- to 0.6-mil deformation at the fuel pellet interfaces because of pellet ridging. Profilometer readings for the portion of the fuel column above the failure is dominated by a region ~1-1/2 inches long where permanent cladding deformation is on the order of 2% to 3% (to a maximum of 8 to 9 mils with a negative 4 mils in one of four orientations, indicating a complex ovality). This region of maximum cladding deformation was investigated in detail during the destructive examination, with the following result.

At the upper end of the fuel column (at the point of maximum cladding deformation), a separation of the upper two fuel pellets from the remaining portion of the fuel column occurred during the transient irradiation, probably prior to cladding failure. The resultant gap (184 mils wide) was filled by molten fuel, which quenched against the inner surface of the cladding (Figure 4-4). The comparatively large cladding deformation was centered about this molten-fuel-filled gap, and was apparently caused by a short-term, localized overheating of the cladding resulting from its contact with molten fuel.

Detailed examination of the cladding in this gap region and adjacent regions where only solid fuel had contacted the cladding, showed no visually detectable difference in the cladding microstructure, i.e., the cladding was apparently unaffected (except for the larger deformation) by the short-term contact of molten fuel on the cladding's inner surface (see Figure 4-5). In addition, overall examination of the cladding (with the exception of that cladding locally adjacent to the failure zone) indicates no noticeable change from the as-fabricated condition.

Molten fuel also penetrated the solid fuel wall at the very top of the uppermost fuel pellet, forming another molten-fuel-filled gap 26 mils wide (Figure 4-4). This ring of molten fuel also quenched against the cladding, again with no apparent effect on the cladding microstructure. The accompanying cladding deformation looked very similar to the fuel pellet ridging noted in the lower portion of the fuel column.

Two adjacent transverse sections were taken at a location at the top of the cladding failure region. In this area, a streamer of molten fuel (about 1/16-inch in diameter and about 1/2-inch in length) flowed up the inner sodium annulus from the main body of the failure region. (This streamer can be seen at the upper end of the dispersed fuel zone in Figure 4-1). The initial cut was made at the apex of the cladding breach, resulting in the two sections shown in Figure 4-6 (~1/16 inch apart). The section looking toward the failed region contained a breach in the cladding for about 20 to 25 degrees of arc (Figure 4-6a); whereas the section looking away from the failed region contained no cladding breach (Figure 4-6b). Detailed examination of the cladding, in the etched condition, showed a totally molten cladding structure extending for 40 to 45 degrees of arc in both sections. In addition, it was clearly evident that the cladding failure did not occur in the weld

region of this welded tubing (the weld is located about 60 degrees clockwise from the cladding breach region seen in Figure 4-6a). The melted cladding region was directly adjacent to and apparently a direct result of the molten fuel streamer. The fuel directly inside of the melted cladding (i. e., the outer region of the original fuel pellet at this location) showed no evidence of melting (even in the area of the cladding breach). At a transverse section located $\sim 1/4$ inch below the dispersed fuel zone (Figure 4-7), the appearance of the fuel and cladding is quite similar to previous zero-burnup, unfailed transient specimens.

Figure 4-8 presents a longitudinal section of specimen C4A located 6 to 7 inches from the base of the fuel column. The interesting features of this region are the two inverted bottle-shaped voids appearing along the fuel centerline, and the chains of small bubbles near the fuel-melt boundary (which appears very well defined in the beta-gamma autoradiograph presented in Figure 4-8).

Another longitudinal section (located 3-1/2 to 4-1/2 inches from the base of the fuel column) is presented in Figure 4-9. The outstanding feature of this section is the worm-like structure of resolidified fuel which nearly fills the large, nearly straight-walled central void (also note the chains of small bubbles similar in appearance and location to those pointed out in Figure 4-8). This structure and arrangement is very similar in appearance to a structure noted in the lower portion of the failed specimen C4B (Figure 4-10). The fact that these similar structures appeared in the lower (below failure zones) regions of both specimens, could be indicative of axial movement of molten fuel upwards towards the failure zone. The driving force for this fuel movement could be explained by one or more of the following: (1) expansion of the gas contained in the fuel porosity (which undergoes a tenfold temperature increase plus a severe

compression from fuel volume change at melt, resulting in a dispersion of high pressure microbubbles in the molten fuel); (2) fuel vapor pressure; (3) absorbed gas from insufficient pellet off-gassing; or (4) some combination of these forcing functions.

Whatever the cause of this apparent upward fuel movement, the hypothesis of its occurrence is strengthened by the reoccurrence of the phenomenon in both failed specimens C4A and C4B, with a very marked similarity in overall appearance:

- a. Both worm-like structures occurred well below the failure zone, and near the bottom end of the respective fuel columns.
- b. In both cases, below these structures, the once-molten fuel is packed to a very high density, with almost a total absence of void.

The appearance of the structure suggests that once the cladding had failed (providing a vent to a low pressure region, i. e., the inner capsule), the molten fuel moved toward that failure region until the driving force was dissipated. As the transient decayed, the molten fuel started to refill the vacated central void, initially flowing freely (hence, the high density lower region). As the fuel continued to cool, increasing viscosity constrained motion until the fuel ceased to flow freely and moved plastically (similar to paste being squeezed from a tube), before final solidification.

Analysis is presently under way to correlate fuel transient thermal expansion to the measured permanent deformation of the cladding. As a part of this study, attempts are being made to identify and define semiquantitatively a driving force which could cause such a fuel movement as described above.

4.2.2 Examination of Powder Fuel Specimens

The destructive examination of the two transiently-irradiated zero burnup powder fuel specimens (C4I and C4J) has continued with the deencapsulation of both specimens and the roughcut sectioning of specimen C4I.

4.2.3 Fast Flux Pre-Irradiated Specimens

The final four specimens destined for fast flux, steady-state irradiation prior to their transient irradiation (specimens C4E, C4F, C4K, and C4L) have been delivered to and accepted by the ANL EBR-II facility. They are now presently awaiting core location assignment.

The two specimens already undergoing irradiation in subassembly XO15 at EBR-II, C4G (F7C) and C4H (F7D), have accumulated 28,200 MWd/Te burnup as of March 31, 1968.

4.3 AXIAL FUEL RELOCATION EXPERIMENTS

The hazards analysis for Series 5 irradiations was forwarded to Idaho for review. Revisions and additions suggested by TREAT personnel were incorporated and submitted for approval. Initial transient irradiations of the Series 5 capsules are tentatively scheduled for mid-May.

The reencapsulation and irradiation schedule is contingent upon approval of the new TREAT shipping cask licensing application; approval is expected in early May.

4.4 DEFECTED OR SODIUM-BONDED FUEL EXPERIMENTS

The design parameters for the Series 6 fuel pins have been finalized and are summarized in Table 4-1. Final design drawings for the Series 6 capsule assemblies are nearly completed, and fuel pin fabrication specifications were prepared and issued. Design

drawings for the sodium-fill system to accommodate the longer length Series 6 fuel pins have been completed, and procurement/fabrication of components has been initiated.

4.5 FLOW BLOCKAGE EXPERIMENTS (FORCED CONVECTION CAPSULE)

The design of an EM flowmeter has been completed. This design is intended for use in Mockup 1A, Mockup 2, and the multipin forced convection capsule (FCC). Orders have been placed for the components which are to be purchased, and assembly will begin when these components arrive. Operation in Mockup 1A will provide useful information with regard to the transient flow response of these flowmeters.

Based on consideration of several commercial pressure transducers, a selection has been made and the order placed. These pressure transducers will be tested both for transient response and steady-state operation. The transient response test will be conducted in Mockup 1A to assure that the transducer is capable of functioning under the conditions associated with a sudden flow reduction. These transducers will also be tested for steady-state operation and calibrated in an operating sodium loop. Any design deficiency should be detected at this time and corrective steps taken before operation in the in-pile FCC.

To monitor the pressure and velocity adequately during a flow coastdown, both the flowmeter and pressure transducer are required to detect disturbances with rise times of <1 msec. It is felt that slower responding instruments may not be capable of properly interpreting the consequences of a sudden flow stoppage.

The design of electrically heated "fuel simulators" has been completed. These fuel simulators are expected to provide a surface heat flux of 800,000 Btu/h-ft² at approximately 1200°F (at least for short times during flow

FIGURE 4-1. TASK C SERIES IV SPECIMEN C4A (90.4% SMEARED DENSITY, ZERO-BURNUP, PELLETED, MIXED OXIDE FUEL) POST-TRANSIENT NEUTRON RADIOGRAPH

FIGURE 4-2. SPECIMEN C4A TRANSIENT IRRADIATION RESULTS - SECOND SHOT (TT1093)
TEMPERATURE HISTORIES - INNER AND OUTER SODIUM ANNULI

FIGURE 4-3. SPECIMEN C4A TRANSIENT IRRADIATION RESULTS - SECOND SHOT (TT1093).
CALCULATED RADIAL TEMPERATURE PROFILES.

FIGURE 4-4a. MACROGRAPH FINE GROUND CONDITION

FIGURE 4-4b. MONTAGE FUEL-ETCHED CONDITION
REDUCED 50 X MONTAGE

SECONDARY GAP 0.026-INCH WIDE, ALSO
FILLED WITH ONCE-MOLTEN FUEL

GAP 0.048-INCH WIDE,
FILLED WITH ONCE-
MOLTEN FUEL

FIGURE 4-4. TASK C SERIES IV FUEL SPECIMEN C4A, LONGITUDINAL SECTION AT TOP OF FUEL
COLUMN SHOWING MOLTEN FUEL FILLED GAP BETWEEN FUEL PELLETS

LOCAT ON: MID PORTION OF FUEL
PELLET 185

LOCATION: MID PORTION OF LARGE
MOLTEN-FUEL FILLED GAP

LOCATION: UPPER END OF FUEL
PELLET 186

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FIGURE 4-5. TASK C SERIES IV FUEL SPECIMEN C4A, COMPARISON OF ETCHED CLADDING ADJACENT TO MOLTEN FUEL FILLED PELLET GAP AND ADJACENT TO UNMELTED FUEL, SEE FIGURE 4-4 FOR EXACT LOCATIONS

FIGURE 4-6a. TRANSVERSE SECTION LOOKING TOWARD FAILURE ZONE, ETCHED FUEL, REDUCED 50 X MONTAGE

FIGURE 4-6b. PARTIAL TRANSVERSE SECTION ~ 1/16 INCH HIGHER THAN FIGURE 4-6a, LOOKING AWAY FROM THE FAILURE ZONE, ETCHED FUEL, REDUCED 50 X MONTAGE

FIGURE 4-7. TRANSVERSE SECTION LOCATED ~ 1/4 INCH BELOW DISPERSED FUEL ZONE, FUEL ETCHED, REDUCED 50 X MONTAGE

FIGURE 4-8a. MACROGRAPH OF LONGITUDINAL SECTION LOCATED 6 TO 7 INCHES FROM BASE OF FUEL COLUMN, FINE-GROUND FUEL

FIGURE 4-8b. BETA-GAMMA AUTORADIOGRAPH OF SAME SECTION—NOTE EXCELLENT DEFINITION OF MELT BOUNDARY, AS-POLISHED FUEL

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FIGURE 4-9a. MACROGRAPH FINE-GROUND CONDITION FIGURE 4-9b. ETCHED FUEL REDUCED 50 X MONTAGE

FIGURE 4-9. LONGITUDINAL SECTION, LOCATED 3-1/2 TO 4-1/2 INCHES FROM THE BASE OF THE FUEL COLUMN

FIGURE 4-10a. MACROGRAPH OF LONGITUDINAL SECTION FROM LOWER PORTION OF SPECIMEN C4B FUEL COLUMN SHOWING WORM-LIKE STRUCTURE SIMILAR TO THAT FOUND IN SPECIMEN C4A (FIGURE 4-9)

FIGURE 4-10b. ALPHA AUTORADIOGRAPH OF SAME SECTION

stoppage). Mockup 1A is intended to demonstrate the ability of these heaters to provide the heat necessary to cause sodium voiding during a sudden flow stoppage. Backup heaters of a slightly different design are being considered in the event the initial heaters fail prior to the onset of sodium voiding. Failure of these heaters (fuel simulators) following sodium voiding is expected, and provisions are being made to allow replacement of the heaters without completely disassembling the capsule. In this way, several heater designs and flow reduction transients could be tested without rebuilding Mockup 1A.

The initial steady-state thermal analysis of Mockup 1A has been completed and the detailed design is under way. Difficulties in mounting the flowmeter required some modification of the initial design. These modifications required a second iteration on the thermal analysis to size the binary gas gap. While the second round analysis has not been completed, the operating parameters are not expected to change significantly from these listed below:

Bulk coolant temperature (test section inlet) = 875°F

Bulk coolant temperature (test section exit) = 1210°F

Bulk coolant temperature (pump region) = 975°F

Coolant velocity through test section = 12 ft/sec

Fuel simulator surface heat flux = 800,000 Btu/h-ft²

Total power input (fuel simulators) = 96 kW

The test section will consist of three, 3-foot long, fuel simulators (electrical resistance heaters) surrounded by an insulated chimney. The lower 2 feet of the fuel simulators will provide the power generation while the upper foot serves as a transition between the heated region and the relatively low velocity coolant above the test section.

Steady-state operation of Mockup 1A is planned initially to verify the analytical model, investigate instrumentation response, and study pump behavior, as well as to resolve any other questions within the capabilities of this relatively simple design. After the steady-state phase has been completed, a sudden flow reduction test (100% to ~0 flow in <1 sec) will be initiated. The information obtained from Mockup 1A will be used in the design of the more sophisticated Mockup 2 and the in-pile FCC.

TABLE 4-1

DESIGN PARAMETERS FOR SERIES 6 FUEL PINS

	Sodium-Bonded		
	Gas-Bonded C6A-3, C6B-3	Pellets C6A-1, C6B-1	Powder C6A-2, C6B-2
Composition	UO ₂		
Enrichment	10% (U-235)		
O/M	2.00		
Length	24 inches (active fuel)		
Fuel/Cladding Gap	0.008 inch (diametral)		---
Density			
Smear	80% TD		
Pellet	86% TD		---
Fuel Diameter	0.212 inch		---
Blanket (upper)	Natural UO ₂		
Length	15 inches		
Density	93% TD Pellets		
Blanket (lower)	Natural UO ₂		
Length	2 inches		
Density	93% TD Pellets		65%—70% TD Powder

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 E1E and E1F (Smeared Thermal Conductivity at High Burnup)

Results of the post-irradiation and metallographic examinations of the E1E and E1F fuel pins have been reported previously.^(1, 6) Calculations of the integral conductivity from the fuel surface temperature (1020°C) to melting were completed and a value of 42.8 ± 6.4 watts/cm was obtained. Data on the $\int k d\theta$ to melting of $(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_{2.00}$ irradiated to low burnup indicated $\int k d\theta$ from 1020°C to melting was 36.5 ± 5.5 watts/cm. Thus, the difference between the two values is about equal to the experimental error.

5.2.2 Capsule Series E1H (Effect of Void Deployment on Fuel Performance)

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

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

Capsule E1AA contains two W-Re in-fuel thermocouples and is designed to measure the in-pile conductivity of mixed oxide fuel continuously to burnups of 100,000 MWd/Te. During this quarter fabrication of the capsule was completed, and reactor insertion is scheduled for early June.

5.2.4 Capsule Series E2J (Effect of Restructuring on Smeared Thermal Conductivity)

This capsule series is designed to measure in-pile thermal conductivity to melting of mixed oxide fuel under conditions where sintering is allowed to occur prior to peak power operation. Preliminary design drawings for capsule series E1J have been initiated.

5.3 HIGH BURNUP IRRADIATIONS

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

Capsule E2J designed to compare the high burnup performance of solid and annular pellet fuel, is being destructively examined. The capsule operated successfully to a burnup of 125,000 MWd/Te at a power of 22 ± 1 kW/ft.

Neutron radiographs and gamma scans of the fuel pins indicated no unusual fuel movement or gamma activity in either of the fuel pins. Radiographs and gamma scans are shown in Figure 5-1. Elements identified at different positions along the fuel pins are listed in Table 5-1. The appearance of the fuel pins after removal from the capsule are shown in Figure 5-2. Destructive examination of the fuel pins is continuing.

TABLE 5-1

INCREMENTAL GAMMA SCANNING OF FCR E2J FUEL PINS NUMBER 1 AND 2

Position	Zr-95	Nb-95	La-140	Cs-134	Cs-137	Pr-144	Ru-103	Pu-106	Co-60	Other Observed Isotopes
1A	85	250							30	Cr-51
1B	75	325		40					225	Cr-51
1C	240	22,000	25	1,250	8,000				55	
1D	1,100	7,350	80	800	5,600				50	
1E	3,600	10,700	35		300	20	5,500	5,000	80	Ce-141; Ce-144
1F	4,000	12,700	30		400	30	4,000	1,500	40	Ce-141; Ce-144
1G	5,400	19,700	40		700	30	7,000	3,000	130	Ce-141; Ce-144
1H	350	1,200		100	450		250	50	50	
2A	175	500							125	Cr-51; Fe-59
2B	130	500		20	50				400	Cr-51; Fe-59
2C	150	10,700	25	3,800	18,200				80	
2D	2,000	10,900	20	500	2,300				60	
2E	1,900	20,800	50	300	1,800	10			80	
2F	1,100	7,100	25	325	1,700	5	800		70	
2G	1,200	9,900	30	1,200	5,600	5	1,000		70	
2H	4,600	12,800	30	250	1,300	25	6,000	2,000	70	Ce-141; Ce-144
2I	4,600	12,800	30	250	1,300	25	6,000	2,000	70	Ce-141; Ce-144
2J	4,000	12,800	30		400	25	14,600	4,500	40	Ce-141; Ce-144
2K	6,500	19,600	40	200	1,300	40	7,000	3,000	60	Ce-141; Ce-144
2L	350	1,200		100	450		300		40

FIGURE 5-1. NEUTRON RADIOGRAPH AND GAMMA
SCAN OF CAPSULE E2J AFTER
125,000 MWd/Te BURNUP

FIGURE 5-2. FUEL PINS E2J-1 AND E2J-2 AFTER 125,000 MWd/Te BURNUP

5.3.2 Capsules E2P and E2Q (Swelling Under Axial Restraint)

Fabrication of capsules E2P and E2Q was completed and their irradiation initiated on March 29, 1968. All instrumentation functioned well. However, the thermocouples indicated that the design power could not be obtained in either capsule. Meetings with GETR physics personnel indicated the flux in the E2P and E2Q reactor position could not be more than 20% lower than predicted. Capsule instrumentation indicated that capsule powers were low by more than 45%. Reasons for the difference in the calculated and observed capsule powers have not been resolved. The capsules were removed from the reactor on April 10, 1968 and neutron radiographed to determine the condition of the fuel and placement of the thermocouples in the capsules. The location of the thermocouples in each capsule appeared normal in the neutron radiographs. Formation of a central void and penetration of fuel into the blanket of fuel pin E2P was observed. The formation of a central void indicates that E2P operated at a higher power than E2Q (thermocouple data indicated that E2Q operated at 25% higher power than E2P). The capsule is being stored in the RML until the reason(s) for the anomalous thermocouple readings are determined.

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, Pu-106, and Pu-239 was completed. Each isotope was found to increase in concentration at the fuel outside diameter. Cladding samples from each pin were obtained and are being analyzed to determine the fission product ratios at the fuel-cladding interface.

5.4.2 Lattice Parameter Measurements of Irradiated Fuel

Lattice parameter measurements on irradiated mixed oxide fuel samples were completed. Six samples of mixed oxide fuel, ranging in burnup from 14,000 to 255,000 MWd/Te were analyzed. The samples had stoichiometries of 1.96, 1.98, and 2.00. The irradiation parameters for each sample were reported earlier in the Twenty-Fifth Quarterly Report.⁽⁶⁾ The following conclusions have been reached based on the analysis:

- a. Some crystallinity was retained in all samples, whether irradiated in fast or thermal flux.
- b. Lattice disruption was more extensive in samples irradiated in a fast flux to 58,000 MWd/Te than in samples irradiated in a thermal flux after 255,000 MWd/Te burnup. This was indicated by the broadening of the 220, 311, and 222 x-ray peaks shown in Figure 5-3. The upper curve is the x-ray diffraction pattern obtained from fuel irradiated in a fast flux to 58,000 MWd/Te burnup.
- c. Initially substoichiometric fuel remained substoichiometric to burnups of 127,000 MWd/Te. This is shown in Figure 5-4. Fuel from pins F2E and E2D-2 which were originally substoichiometric were found to have a lattice parameter greater than stoichiometric material after irradiation, indicating they remained substoichiometric.
- d. The lattice parameter of all samples decreased with irradiation. This is shown in Figure 5-4 and Table 5-2 which list the pre- and post-irradiation lattice parameter of the samples. The composition of the irradiated samples is shifted toward

higher UO_2 concentrations in Figure 5-4 because the plutonium-to-uranium ratio decreases with burnup, because of the higher absorption cross section of plutonium. As shown by the solid line for unirradiated stoichiometric (0.2 Pu-0.8 U) $\text{O}_{2.00}$ in Figure 5-4 the lattice parameter would normally increase as the Pu-to-U ratio decreases. The post-irradiation lattice parameters indicate that a net contraction of the lattice occurred. This could be caused by an increase in oxygen-to-metal ratio or by the inclusion within the lattice of soluble fission products such as cerium and zirconium. The latter is probably the dominant effect. This is indicated by the significant decrease in lattice parameter measured from samples E2D-1 and E2H which were initially stoichiometric. Based on thermodynamic data, these samples should not become hyperstoichiometric since they were irradiated in stainless steel cladding. In addition, sample E2H was found to have a greater decrease in lattice parameter than E2D-1 which had about one-half the burnup. The lattice parameter decreases measured, taking into account the precision of measurements, are in good agreement with theoretical calculations if all the soluble fission products were homogeneously distributed in solid solution. (13)

Substoichiometric samples E2D-2 and F2E also show unit cell contraction; they may be caused by the increase in stoichiometry as well as the inclusion of fission products within the lattice. In summary, the inclusion of solid fission products either as elements or cations are believed responsible for the lattice parameter reduction.

A comparison between samples of two different stoichiometries which failed at

14,000 MWd/Te showed that the unit cell sizes were equal, and slightly smaller than the pre-irradiated values. Sodium reacted with the samples after the failure occurred.

5.5 PHYSICAL PROPERTY STUDIES

5.5.1 Mechanical Properties of Ceramic Fuels

Initial creep and plasticity data has been obtained on 95% dense, 12–18 micron grain size UO_2 with an O/U ratio of 2.008 ± 0.005 . These data are plotted in Figure 5-5 and can be described by an equation of the form:

$$\dot{\epsilon} = A \sigma^m e^{(-Q/RT)} \quad (1)$$

where

- $\dot{\epsilon}$ = strain rate (in./in./h)
- A = constant
- M = constant
- σ = applied stress
- e = constant
- Q = activation energy (kcal/mole)
- R = gas constant
- T = absolute temperature

An activation energy of 97.4 kcal/mole was calculated from these preliminary results. This value is in good agreement with the values of 90 to 100 kcal/mole reported by other authors. (14, 15)

5.5.2 Melting Point of Irradiated Fuel

The melting point of (0.25 Pu-0.75 U) $\text{O}_{2.00}$ irradiated to 250,000 MWd/Te burnup was observed to decrease 130°C to ~2640°C. Previous measurements on fuel ranging in burnup from 55,000 to 93,000 MWd/Te indicated a decrease of 50° to 100°C in the melting point. (3)

FIGURE 5-3. EFFECT OF IRRADIATION AND NEUTRON ENERGY ON MIXED OXIDE DIFFRACTOGRAMS

FIGURE 5-4. EFFECT OF IRRADIATION ON LATTICE PARAMETERS OF $\text{UO}_2\text{-PuO}_2$ COMPOUNDS

TABLE 5-2

RESULTS OF X-RAY DIFFRACTION ANALYSES OF IRRADIATED FUEL SAMPLES

Sample	Pre-Irradiation				Post-Irradiation					Remarks
	Weight/Percent PuO ₂	Oxygen-to-Metal	Lattice Parameter	Site	Weight Percent PuO ₂ ⁽²⁾	Burnup $\times 10^3$ MWd/Te	Lattice Parameter Å	Lattice Parameter of Unirradiated Stoichiometric Compound of Post-Irradiation Composition Å	Lattice Parameter of Unirradiated Stoichiometric Compound of Post-Irradiation Composition Reduced 0.005 Å/100,000 MWd/Te ⁽⁴⁾ Å	
E2D-1	24.9	2.00	5.454 $\pm$ 0.0005	HAPO	20.5	127	5.453 $\pm$ 0.002	5.4555	5.4492	0.046% reduction of parameter excluding composition differences; within estimate of contraction from inclusion of Zr ⁽⁴⁾ and rare earth's in unit cell.
E2D-2	24.9	1.96	5.467 $\pm$ 0.0005	HAPO	20.5	127	5.467 $\pm$ 0.002	---	---	Unchanged from pre-irradiation a_0 , but smaller than unirradiation for this composition.
E2H	22.1	2.00	5.4545 ⁽³⁾		12	255	5.450 $\pm$ 0.002	5.4620	5.4495	0.22% reduction in parameter excluding compositional change. Greater contraction than E2D-1 believed caused by greater concentration fission products; within calculated estimate of contraction. ⁽⁴⁾
F2E	19.7	1.98	5.4676 $\pm$ 0.0005	HAPO	18.9	58	5.462 $\pm$ 0.002	---	---	0.1% reduction parameter; extensive fine broadening.
B8B-1 ⁽¹⁾	24.7	2.00	5.4535	HAPO	18.4	14	5.452 $\pm$ 0.002	5.4570	5.4560	Exposed to sodium; slight cell reduction.
B8B-2 ⁽¹⁾	24.7	1.94	5.4730 ⁽³⁾		18.4	14	5.452 $\pm$ 0.002	---	---	Exposed to sodium; drastic cell size reduction.

- NOTES: 1. Exposed to sodium after failure.
 2. Calculated from burnup analyses.
 3. Estimated from VNC data.
 4. Estimate per F. Anselin, W. E. Baily, Trans. ANS, 10, 1, 1967.

FIGURE 5-5. CREEP RATES OF $\text{UO}_{2.008}$ AS A FUNCTION OF TEMPERATURE

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

Final assembly drawings for the furnace to measure thermal conductivity of mixed oxides have been received and approved. Construction of the furnace is proceeding satisfactorily, and delivery is scheduled for the middle of May.

5.5.4 Thermal Gradient Tests

Analysis of three thermal gradient capsules containing 1.935, 1.978, and 1.998 O/M fuel was completed.

The results from the 1.978 and 1.998 O/M capsules and an early test⁽¹²⁾ on 1.935 O/M material are shown in Table 5-3. In each case the heating period was approximately 4 hours at a maximum temperature of 2000°C. Post-test O/M measurements on individual pellets from each capsule showed that oxygen migrated from the hot to the colder regions of the fuel. In the capsule containing 1.935 O/M fuel an oxygen balance showed that the average O/M of the fuel was within 0.0005 units of the initial O/M; an indication that no oxygen was lost (or gained) during the experiment. The 1.978 and 1.998 O/M capsules indicated a net loss of oxygen from the capsule. The largest oxygen loss was found in the capsule with the highest O/M ratio. Molybdenum metal (from the capsule body) was found in the hottest pellets in each capsule. The oxygen loss is most likely due to the high internal oxygen partial pressure of the high O/M fuels which, in turn, increased the oxygen permeation through the molybdenum capsule wall. The molybdenum found on the pellets probably resulted from the high oxygen partial pressure. If the oxygen partial pressure was high enough, MoO₃ gas would be produced. After oxygen was lost, however, the oxygen pressure decrease would result in a reduction of the MoO₃ and deposition of molybdenum on the pellet surfaces.

A fourth test was run using 1.935 O/M fuel. In this test, the heating period was extended to 15 hours at a maximum temperature of 2000°C. Compared to the earlier short time test, the O/M ratio of the pellets at the cold end of the fuel increased markedly.

Results of each test were plotted in terms of the Arrhenius equation. The data plotted in Figures 5-6 through 5-9 indicate a reasonably linear relationship corresponding to a heat of transport of about -10 kcal/mole.

These tests indicate that: (1) O/M profiles in mixed oxide fuel can be predicted under steady-state operation if the temperature profile of the fuel is known and (2) steady-state O/M profiles are probably reached in times very short compared to the lifetime of the fuel.

5.6 INTEGRAL THERMAL FLUX IRRADIATIONS

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

The destructive examination of the two E5B fuel pins was completed. A summary of the results of the examination is given below.

- a. Fission gas release from the powder and pellet fuel was 81% and 91% respectively.
- b. Cladding attack of 0.002 to 0.003 inch occurred.
- c. The peak power region of each fuel pin ran at a power of 23.5 kW/ft which resulted in incipient center fuel melting.
- d. Central void size and columnar grain growth radii indicate similar fuel performance in sections from each pin operating at the same power.

FIGURE 5-6. RESULTS OF THERMAL GRADIENT TEST ON $(0.2 \text{ Pu}-0.8 \text{ U})\text{O}_{1.935}$

FIGURE 5-7. RESULTS OF THERMAL GRADIENT TEST ON $(0.2 \text{ Pu}-0.8 \text{ U})\text{O}_{1.998}$

FIGURE 5-8. RESULTS OF THERMAL GRADIENT TEST ON $(0.2 \text{ Pu}-0.8 \text{ U})\text{O}_{1.978}$

FIGURE 5-9. RESULTS OF THERMAL GRADIENT TEST ON $(0.2 \text{ Pu}-0.8 \text{ U})\text{O}_{1.935}$

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

Capsules E5C, E5D, and E5F continue to operate at linear powers of 19.5 kW/ft. Burnup as of May 1, 1968 was calculated to be 53,000 MWd/Te.

5.6.3 Capsule E5F (Linear Fuel Movement)

Fuel pin E5F continues to be irradiated in GETR, and, as of the end of

cycle 94, has successfully attained a peak burnup of approximately 12,400 MWd/Te. Cycle 95 irradiation will be completed April 28, 1968. Because of transducer chart calibration difficulties, data analysis was not completed. The difficulties were resolved, however, and data analysis is continuing. In addition, more power traverses will be run to note any significant ΔL versus linear power changes with burnup.

TABLE 5-3
OXYGEN-TO-METAL RATIOS AS A
FUNCTION OF POSITION IN A THERMAL GRADIENT CAPSULE

Position	Pin No. 1 Initial O/M—1.935	Pin No. 2 Initial O/M—1.998	Pin No. 3 Initial O/M—1.978
1	1.934	Lost	1.979
2	1.936	1.997	1.983
3	1.944	1.987	1.987
4	1.944	1.975	1.974
5	1.936	1.968	1.966
6	1.922	1.963	1.954
7	1.921	1.960	1.949
8	1.910	1.963	1.945
9	1.929	1.973	1.949
10	1.971	1.994	1.957
11	1.936	2.000	1.994
12	None	2.000	1.974
Average change in O/M per Pellet	+ 0.0005	-0.0183	-0.0108

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 re-

lease, 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 IRRADIATION

6.2.1 Group 2

The irradiation of Group 2, consisting of nine capsules in subassembly XG05, was terminated after receiving a peak burnup calculated to be 74,000 MWd/Te. This subassembly was removed from the core as a suspected leaker, however, fission gas activity was subsequently detected in the reactor cover gas with S/A XG05 out of the core. The capsules were removed from the subassembly at the EBR-II site and neutron radiographed using the TREAT reactor for a neutron source. The results indicate that eight fuel pins are intact and that one, F-20, failed. The Group 2 test parameters and operating conditions are included in Table 6-1.

6.2.2 The isotopic distribution of the fission gas collected from the Group 2 fuel pins in S/A XG06 has been measured. The results of the mass spectrographic analysis of the gas samples are presented in Table 6-2.

6.3 LOW POWER FUEL SWELLING IRRADIATION

The irradiation of Groups 0, 6, and 8A was temporarily discontinued when the EBR-II released fission gas periodically during Run No. 27. Attempts to isolate the source of fission gas resulted in the removal of the subassemblies containing these experiments from the EBR-II core. They are currently located in reactor storage. Their accumulated exposures to date are presented in Table 6-3.

The nineteen Group 8B capsules were delivered to the EBR-II site, passed the pre-irradiation inspection, and are ready to be incorporated into a subassembly for insertion into the EBR-II core.

6.4 HIGH POWER INTERMEDIATE BURNUP IRRADIATION

The seven remaining capsules from Group 4 will be incorporated into a new subassembly for reinsertion into the EBR-II core for Run No. 29. They have accumulated a calculated burnup of 35,300 MWd/Te. The target burnup is 50,000 MWd/Te.

6.5 ELEVATED CLADDING TEMPERATURE TESTS

Fuel pin fabrication is continuing, with 14 pins having been completed and assembly of the remaining two in process. Encapsulation of fuel pins is continuing. ANL has responded to the request for a flow test of an EBR-II Mark A subassembly incorporating cold sodium bypass flow. Flow testing has been initiated on the coolant mixing promoters based on the staggered tube design.

6.6 UNENCAPSULATED PIN IRRADIATIONS

6.6.1 Group 3A (High Power)

A revised hazards analysis has been submitted to EBR-II to cover the insertion of these 16 pins in a 37-pin subassembly shared with ANL. The 0.290-inch outside diameter pins are presently at the EBR-II site ready for incorporation into a subassembly.

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

Fabrication of the unencapsulated fuel pins (0.250-inch outside diameter) for the 37-pin Group 9A is continuing. Fabrication of Group 9B has been delayed until completion of Groups 9C and 9D. The modified subassembly EBR-II Mark 37-F has been accepted for these tests. Fabrication of the hardware for these subassemblies was initiated by ANL.

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

The design has been completed and fabrication initiated on the 0.250-inch outside diameter fuel pins for these two 37-pin subassemblies. The fabrication parameters for 9C and 9D are shown in Table 6-3. The design parameters for these two subassemblies were initially determined with PNL-FFTF.

6.7 FAILED FUEL EXAMINATION

6.7.1 Examination of Failed Rod F2F (58,000 MWd/Te)

Metallographic examination of this rod has continued and the fuel-sodium reaction phase within the fuel-cladding gap has appeared in all specimens along the fuel region. This alpha active phase however, was not observed within the gap of the

slightly enriched UO_2 blanket directly atop the fuel. This is shown in Figure 6-1. Its absence in the blanket region may be because of the surface temperatures being below a critical reaction temperature or the migration of oxygen from the hyperstoichiometric $\text{UO}_{2.02}$ to the hotter, substoichiometric mixed oxide fuel.

A gas release of 33% was measured which is lower than that observed in the unfailed pins (50%—60%) operating at comparable power and burnup levels. Examination of the bottom end cap in the region of the weld in the transverse plane has shown several radial microcracks in the cladding during the course of grinding at intervals of 15, 22, and 50 mils. None of these are through cracks. Remounting the sample and grinding to obtain a longitudinal view revealed a crack extending through the thickness of the cladding to the interface between the cladding and the bottom end cap. These views are illustrated in Figure 6-2. It is also clear from this figure that these cracks are considerably below and not related to the lowest location of the cracking observed over the fuel column. Although it is not certain that any of the observed cracks were passage ways for the release of fission gas and/or the entrance of sodium, their existence does indicate a possible location and initiating mode of pin failure.

6.7.2 Examination of Failed Rod F4D (34,000 MWd/Te)

This Incoloy 800 clad pin was removed from testing to determine the amount of strain that an unfailed pin with this cladding alloy could sustain.

The detection of fission gas within both the capsule and pin (a total of 51%, with about 9% in the capsule) inferred a fuel pin leak during its lifetime although no other indications of rupture were evident. Profilometer measurements indicated an average diametral increase of about 0.7 mil.

6.7.3 Examination of Failed Rod F4F (34,000 MWd/Te)

The fuel pin was removed from its capsule and examination showed that only about 6 inches of the 14.2-inch fuel column were damaged; the lower 5.5 inches of the pin were unaffected and essentially no dimetral increases were measured. A striking feature of the damage regions was the localized zones of melted cladding which took the form of elongated craters extending over a circumferential arc of about 60 degrees. A 35-mil-high ridge of solidified metal, as shown in Figure 6-2, surrounded the unrestrained fuel which was exposed to the sodium coolant.

The ridges display the dendritic structure of solidified melt, while the fuel directly adjacent showed no localized microstructural change. A condition of drastic overheating, confined to the cladding, seems to have occurred, possibly as the result of a gas blanketing condition after initial pin failure. No significant fuel-sodium reaction phase was noted.

6.8 EBR-II POWER COEFFICIENT CALCULATIONS

The effect of fully-enriched oxide fuel on the EBR-II power coefficient of reactivity was calculated using the two-dimensional synthesis code BISYN. The normal metallic fuel of EBR-II was replaced by oxide in fuel rows 1, 2, and 3. The power produced per oxide subassembly approximated that of the metallic EBR-II fuel in the same position. Power coefficients were also calculated for the all-metal core and used as a methods check by comparing the results with those published by ANL 7382, "Reactor Development Progress Report," pages 15 to 17, September 1967.

Results of the calculations suggest that even with cladding controlled axial expansion of the oxide fuel, comparable power reactivity coefficients were obtained.

6.9 HYDRAULIC FLOW TESTS

Flow testing has been initiated. Tests at flow velocities from 4 to 14 ft/sec (actual water velocities) have been completed for a P/D of 1.3 and 1.1. Some skewing of the velocity profile has been observed and attempts to define the problem are now under way.

TABLE 6-1
GROUP 2 TEST PARAMETERS AND OPERATING CONDITIONS

Capsule No.	Fuel Type	ρ_s (% TD)	Oxygen-to-Metal Ratio	Cladding Material	Peak Heat Rating (kW/ft)	Calculated (BU max)	Diameter Gap (mils)
*F-2C	Pellet	93.1	2.00	Incoloy	16.5	71	2-5
*F-2D	Pellet	94.9	2.00	347 SS	17.0	73	2-6
*F-2G	Pellet	92.9	1.98	347 SS	16.6	71	2-6
*F-2H	Pellet	94.9	1.98	316 SS	17.2	74	2-6
F-20 (failed)	Pellet	94.0	2.03	347 SS	16.5	71	1-5
*F-2R	Pellet	95.2	2.00	347 SS	16.5	71	1-2
F-2T	Pellet	94.6	2.00	347 SS	17.3	74	1-6
F-2V	Pellet	86.8	1.99	316 SS	16.3	70	2-6
F-2X	VCP	83.0	1.99	316 SS	13.7	59	

* Axially restrained fuel column

20 wt % plutonium 80 wt % uranium Nominal
14.2 inches Fuel Column Nominal

FIGURE 6-1. GAP REACTION PHASE AT TOP OF FAILED PIN F2F

FIGURE 6-2. INTERGRANULAR CRACKING OF CLADDING AT BOTTOM END CAP OF FAILED PIN F2F

TABLE 6-2
ISOTOPIC DISTRIBUTION OF FISSION GAS IN GROUP 2 FUEL PINS

<u>Fuel Pin</u>	<u>F2A</u>	<u>F2B</u>	<u>F2E*</u>	<u>F2E*</u>	<u>F2N</u>	<u>F2P</u>	<u>F2Q</u>	<u>F2S</u>	<u>F2U</u>	<u>F2W</u>	<u>F2Y</u>	<u>F2Z*</u>	<u>F2Z*</u>
<u>Isotopic Abundances of Krypton and Xenon</u>													
Kr-83	15.68	15.86	16.25	15.98	16.06	15.84	15.46	15.67	15.77	15.71	15.72	15.93	15.90
84	27.47	27.43	26.39	27.39	27.78	27.52	27.64	27.66	27.55	28.01	27.43	27.52	27.54
85	7.08	6.94	7.69	7.18	6.81	6.74	6.89	7.01	7.20	7.05	6.73	7.30	7.29
86	49.77	49.77	49.66	49.46	49.35	49.90	50.01	49.67	49.48	49.24	50.08	49.25	49.27
Xe-131	15.37	15.28	15.52	15.62	15.15	15.70	15.14	15.01	15.29	15.36	15.02	15.81	15.65
132	21.90	21.84	22.37	22.14	22.16	22.00	21.83	21.85	22.21	22.33	22.21	22.44	22.21
134	34.17	34.29	33.75	33.78	34.31	33.39	34.17	34.31	34.11	34.01	34.02	33.73	33.88
136	28.58	28.58	28.36	28.46	28.38	28.91	28.86	28.84	28.39	28.30	28.75	28.01	28.26

* Separate analysis performed on aliquot samples.

TABLE 6-3
FABRICATION PARAMETERS FOR 9C AND 9D

Number of Fuel Pins	37 total
Fuel Preparation	Mechanically blended Powder (20) Coprecipitated powder (17)
Fuel Geometry	Solid pellet (33) Vi-Pack pellet (4)
Stoichiometry (O/M)	1.99 ± 0.01 (37)
Composition wt % PuO_2	25 ± 1.5 (37)
U-235 Enrichment	91% minimum (37)
Cladding Material	Type 316 stainless steel (37)
Fuel Pellet Diameter	0.2150 ± 0.0020 (33)
Fuel Pellet Density	$92\% \pm 2\%$ TD (25) $89\% \pm 2\%$ TD (8)

* Number of parenthesis indicates number of pins having stated parameter.

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 de-

sign 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

Modified cross sections for iron, which show more detail in the total and elastic scattering data than those presently on ENDF/B, have been sent to Brookhaven National Laboratory at the request of S. Pearlstein. These modified values are those used to obtain the more highly self-shielded effective cross sections reported in the Twenty-Fifth Quarterly Report.⁽⁶⁾ It is expected that BNL will distribute these modified data to members of the CSEWG as in interim set of improved ENDF/B iron data, for use until the reevaluated iron data, requested by the CSEWG, become available.

7.2.2 Analysis of Critical Experiments

A comparison of the corrected neutron flux spectrum measured in the center of ZPR-III, Assembly 48⁽¹⁷⁾ and the spectra computed earlier show that the corrected measurements agree substantially better with calculations. A more careful comparison will be made when Argonne National Laboratory provides the corrected measured spectrum in detail.

7.2.3 Cross Section Files

The up-dated cross sections for U-238 and Pu-239, as described in the Twenty-Fifth Quarterly Report,⁽⁶⁾ have been incorporated into a modified version of the CROF code, called CROF03. CROF03 provides the capability of obtaining effective cross sections for the fuel isotopes which correspond to a particular reactor composition and fuel temperature, and which are based on higher values for σ_f and α of Pu-239, and lower values of the U-238 capture cross section. Updated cross sections for the other materials will be included in the 60-group, BAS03 file, which is nearing completion. It was decided that ENDRUN

results, using the ENDF/B file, would be used for BAS03 for all reaction types of iron and Boron-10, capture and fission of U-235, and fission, capture, and $\bar{\nu}$ of Pu-241.

New data measurements, corrections to existing experimental results, and theoretical interpretations of nuclear data were presented and discussed at the Conference on Neutron Cross Sections and Technology, Washington, D.C., March 4-7, 1968. The more important results indicate a small downward correction to the high Pu-239 alpha values of Schomberg,⁽¹⁸⁾ particularly in the energy range above 10 keV, and lower values of σ_c for U-238 between 5 and 100 keV. The visit to the RPI linac facility gave the impression that meaningful results will be forthcoming from these measurements of the Pu-239 alpha value.

7.2.4 Multigroup Cross Section Adjusting

The ENDRUN routines for averaging smooth data and for generating matrices for inelastic scattering and (n, 2n) reactions have been checked out in the linked form. This now provides the capability of generating files for the PNL FCC-IV code, including infinitely dilute cross sections, self-shielding factors, a scattering matrix and the group dependent quantities $\bar{\nu}$, $\bar{\mu}$, $\bar{\xi}$, and χ from ENDF/B materials which do not have resonance parameter data.

Checkout of the resolved and unresolved resonance averaging routines had indicated that greater accuracy was needed in the numerical integration procedures for certain cases. The unresolved resonance integration procedure has now been improved and the first tests with Pu-239 resonance parameters indicate that it provides sufficient accuracy. Improvements in the resolved resonance procedure have been coded, but checkout has not begun. These improved integration techniques

should make both resonance averaging routines accurate enough for virtually all applications of ENDRUN without greatly increasing the computation time. (The major computation time for ENDRUN is expected to be in the resonance averaging calculations for fuel materials.) Checkout of the final stage of ENDRUN, that of combining resolved resonance, unresolved resonance, and smooth, self-shielded cross sections, has begun.

7.2.5 Fast Reactor Sensitivity to Nuclear-Data Uncertainties

The study of the sensitivity of fuel costs to uncertainties in nuclear data, the results of which were reported at the meeting of the American Nuclear Society in November, 1967, ⁽⁶⁾ has been continued by considering the effects these uncertainties may have on fast reactor designs. This extension of the study was reported in a paper presented at the Conference on Neutron Cross Sections and Technology, in Washington, D.C., on March 4—7, 1968.

If nuclear data are appreciably on the adverse side of the uncertainty ranges,

present design considerations and associated development programs may not achieve the desired performance targets for LMFBR's. An example of design direction changes which could be necessary in order to achieve, for example, a target of 10 years in doubling time are shown in Table 7-1. The base design is an advanced LMFBR which will achieve a 10-year doubling time if the present best estimates of nuclear data are correct. The other two cases are modifications to the base design which would give 10-year doubling times even if the Pu-239 α value below 15 keV is at the upper limit of the uncertainty range. To achieve this condition, the fuel was sodium-bonded to the cladding in one case, and the fuel rod diameter reduced from 0.25 to 0.225 inch in the other. The fuel rod pitch-to-diameter ratio was reduced for both alternates, permitting a slightly higher fuel volume fraction with lower sodium content. Fissile loadings, breeding ratios, doubling times, and fuel costs given in Table 7-1 have been computed with the high Pu-239 α values, thus increasing the doubling time of the base design to 13.5 years and the fuel costs from 0.50 to 0.62 mills/kWh.

TABLE 7-1

POSSIBLE CORE DESIGN CHANGES TO COUNTERACT HIGH Pu-239 ALPHAS

	<u>Base Design</u>	<u>Sodium Bonded</u>	<u>Small Rod Diameter</u>
Fuel Rod Diameter (inch)	0.25	0.25	0.225
Sodium-Bonded	No	Yes	No
Rod Pitch- to -Diameter	1.25	1.20	1.20
Fuel/Sodium/Steel (vol %)	42/39/18	44/37/18	44/37/18
Core Diameter (ft)	8.0	7.15	7.15
Pressure Drop (psi)	80	190	210
In-Pile Fissile Pu (kg)	2430	2020	2020
Breeding Ratio	1.28	1.31	1.31
Doubling Time (year)	13.5	10	10
Fuel Cost (mill/kWh)	0.62	0.51	0.54

Sodium-bonding or the smaller fuel rod diameter each permit an increase in fuel specific power, leading to a reduction in the fissile inventory by about 20%. The higher fuel volume fraction and lower sodium fraction increase the breeding ratio from 1.28 to 1.31, but the tighter lattice and higher heat transfer from fuel to coolant require about a 35 ft/sec sodium flow velocity for the alternate designs, which increases the coolant pressure drop across the core from 80 to about 200 psi.

7.3 CORE ANALYSIS

7.3.1 Multigroup Cross Section Condensation Methods

The two-dimensional, synthesis, diffusion-theory code BISYN is presently restricted to a maximum of 16 energy groups, and a maximum of eight down-scattering groups. The two-dimensional diffusion-theory codes CRAM and EXTERMINATOR-2 are also normally limited in the number of groups because of excessive computer times required. Since the base cross section libraries used at GE contain cross sections in a 60-group format, these libraries must be condensed to ≤ 16 groups for use in these two-dimensional diffusion codes. A study of the sensitivity of the results of neutronics calculations to the condensation methods used, the number of condensed groups, and the condensed group structure has been undertaken.

In a typical condensation, region-integrated fluxes from one-dimensional, 60-group calculations are used to weight the cross sections. Axial and radial models of Reference Core 24C (see the Twenty-Fifth Quarterly Report⁽⁶⁾) were used for this sensitivity study. The axial model of this Reference Core contained three core regions, four blanket regions and an axial reflector. The radial model consisted of three core zones of different "enrichment," two blanket regions and a reflector. Radial

Core Zone 1 was identical to the core in the axial model.

The procedure used in the sensitivity study consisted of 60-group, one-dimensional calculations of the axial and radial models of the Reference Core, regionwise condensation of the cross sections and bucklings to several different 16-group structures, and then recalculation of the identical problems with the condensed cross sections. In addition to regionwise condensation, a homogenization routine was used to combine the cross sections and fluxes of the three axial core regions into one representative region, and the four blanket regions into another, prior to condensation to one of the 16-group structures. Differences obtained between the 16- and 60-group calculations can be attributed to the condensation procedures and to the 16-group structure chosen.

The three different 16-group structures listed in Table 7-2 were considered. The first of these is the one currently used for most BISYN calculations at GE. The second group structure was obtained by applying a recipe which requires that the spatial peak flux (considering both the sodium-in and out cases) in each condensed group be approximately the same. The third group structure was obtained by requiring that the spatial peak of the product of the flux and adjoint-flux (again for both the sodium-in and out cases) in each condensed group be approximately the same. The BISYN restriction to eight downscatter groups in conjunction with the 34 downscatter groups used in the 60-group libraries limited the number of coarse groups between lethargies 0 and 7.5 to 9.

Partial results of the calculations are presented in Table 7-3. The set of values denoted "Axial 2 Reg." were obtained by condensing only in the second axial core region and the second axial blanket region and using these cross sections in the core and blanket, respectively.

TABLE 7-2
16-GROUP STRUCTURES

Group	Lethargy* Width		
	Structure 1	Structure 2	Structure 3
1	1.5	1.5	1.5
2	0.5	1.0	1.0
3	0.5	1.0	0.5
4	0.5	0.5	0.5
5	1.0	0.5	0.5
6	1.0	0.5	0.5
7	1.0	0.5	0.5
8	1.0	1.0	1.0
9	0.5	1.0	1.5
10	0.5	1.0	1.0
11	0.5	0.7	0.7
12	0.7	0.7	0.7
13	0.7	0.7	0.7
14	0.7	1.0	1.0
15	3.0	2.0	2.0
16	∞	∞	∞

* Lethargy (10 MeV) = 0.

TABLE 7-3
PARTIAL RESULTS OF CONDENSATION STUDY

	Base Case 60-Groups	Percent Difference Using 16-Group Cross Sections			
		Structure 1	Structure 2	Structure 3	Structure 3
<u>k_{eff}</u>					
Sodium-In - Axial	0.99445	-0.31	+0.007	-0.15	-
- Axial 2 Reg.	-	-0.35	-0.008	-0.16	+0.17
- Radial	0.99589	+0.005	+0.008	+0.01	-
Sodium-Out - Axial	1.00803	-0.41	-0.003	-0.20	-
- Axial 2 Reg.	-	-0.45	-0.014	-0.20	+0.16
- Radial	1.00376	+0.002	+0.003	+0.01	-
<u>Doppler (T dk/dT)</u>					
Sodium-In - Axial	0.005797	-3.0	-3.3	-3.5	-
- Axial 2 Reg.	-	-2.9	-3.1	-3.4	-3.7
- Radial	0.005433	-2.5	-3.0	-3.3	-
Sodium Out - Axial	0.003625	-1.9	-3.1	-3.3	-
- Axial 2 Reg.	-	-1.7	-3.0	-3.2	-3.0
- Radial	0.003477	-1.5	-3.0	-3.0	-
<u>(Δk_{Na}) total loss</u>					
Axial	-0.01366	-7.7	-0.77	-3.4	-
Axial 2 Reg.	-	-7.4	-0.41	-3.0	-0.75
Radial	-0.00790	-0.5	-0.57	-0.6	-

The following conclusions can be drawn from the results in Table 7-3:

- a. The error in k_{eff} , using the current group structure (Structure 1) with region-wise condensation, is significant in the axial calculations.
- b. The results for k_{eff} and sodium loss reactivity are significantly improved with Structure 2, which is based on equal peak fluxes in the coarse groups.
- c. The error in all three quantities introduced by using only typical core and blanket spectra, as indicated by the values denoted "Axial 2 Reg.", is small. For typical LMFBR problems, condensation in every region does not appear to be necessary.
- d. The homogenization procedure does not improve the calculations.
- e. Except for Structure 2, the errors are much larger in the axial model than in the radial model. This may be explained by the fact that the axial leakage from the core is four to five times larger than the radial leakage, and the results are most sensitive to the condensation of the "transport" cross sections.

In addition to the results shown in Table 7-3, the fission spectrum, power distribution, breeding ratios, and Pu-239 reaction rates were examined. The agreement was found to be generally good, except in the case of homogenization.

7.3.2 Anisotropic Scattering

The transport and diffusion theory calculations presented in Reference (19) have been extended, for two fast reactor models, to include anisotropic scattering. The results indicate that the errors in diffusion theory calculations due to inadequate treatment of flux anisotropy and scattering anisotropy are nearly independent.

Both errors can be significant, but much of the error from anisotropic scattering can be compensated for by the use of a transport cross section which accounts for the anisotropic scattering.

Three transport cross section approximations were considered in the study. These are, for group g ,

$$(1) \quad (\sigma_{\text{tr}})_g = (\sigma_a)_g + (\sigma_s)_g (1 - \bar{\mu})$$

$$(2) \quad (\sigma_{\text{tr}})_g = (\sigma_a)_g + (\sigma_s)_g - \sum_{g'} \sigma_{1, g' \rightarrow g} \frac{(\Delta u)_{g'}}{(\Delta u)_g}$$

$$(3) \quad (\sigma_{\text{tr}})_g = (\sigma_a)_g + (\sigma_s)_g - \sum_{g'} \sigma_{1, g \rightarrow g'}$$

where $\bar{\mu}$ is the average cosine of the scattering angle for isotropic scattering in the center of mass system, $\sigma_{1, g' \rightarrow g}$ is the P_1 elastic scattering matrix, and $(\Delta u)_g$ is the lethargy width of group g . Approximation (1) is the usual transport cross section definition. Approximation (2) is obtained from the P_1 equations by assuming the net current varies as $1/E$ in the anisotropic scattering source term. Approximation (3) is given in Reference (18) and is there called the "extended" transport approximation, and is essentially the same as Approximation (1) in which $\bar{\mu}$ is the average cosine for anisotropic scattering.

The code ANISN was used both for the transport and the "diffusion" calculations. Simplified, spherical reactor models were considered, in which hydrogen was taken to be the primary scattering material, since it would exaggerate the anisotropic effect. The results obtained for the two reactors considered are presented in Table 7-4, along with S_n results and "consistent P_1 " diffusion theory results. The results

labeled "Diff-TR#1", "Diff-TR#2", and "Diff-TR#3" refer to diffusion theory results using, respectively, the transport cross section approximations (1), (2) and (3), respectively.

7.3.3 Core Model Studies

Refinement of the value used for radial blanket power and assumed reprocessing costs has resulted in a somewhat greater calculated economic incentive than was previously reported, ⁽⁷⁾ for using (23.75 kW/ft) sodium-bonded fuel in a 1000 MWe LMFBR. In addition, an estimate was made of the economic implication of using 30 kW/ft sodium-bonded fuel in an LMFBR.

The results of several RAPFU⁽²¹⁾ cases were used to refine the initial estimates of radial blanket power splits for 1000 MWe cores of varying specific power. All calculations were initially done using the methods described in previous quarterlies. As might be expected in constant pressure drop cores, higher specific powers result in greater sodium volume fractions and somewhat smaller core diameters. Both of these effects tend to increase the radial blanket power more than was previously assumed, resulting in improved blanket and core fuel cycle economics. A lower, more realistic recovery cost was also assumed. As a result, the previously reported ≈ 0.02 mill/kWh fuel cycle cost incentive for using 23.75 kW/ft sodium-bonded fuel appears to actually be ≈ 0.05 mill/kWh, assuming no additional cost for fabricating or reprocessing sodium-bonded fuel (see Table 7-5).

Because the radial blanket residence time built into RAPFU is twice that of the core, this parameter could not be varied. As a result, a somewhat lower radial blanket burnup caused slightly higher fabrication and recovery costs in the higher

specific power cases. A reoptimization of the refueling interval might decrease or eliminate this cost penalty.

Two other cores were considered which assumed higher specific power (30 kW/ft) sodium-bonded fuel. A peak linear power of 30 kW/ft was chosen as a reasonable operating target if, in addition to increasing the fuel-cladding gap conductivity, the sodium was found to increase the effective fuel thermal conductivity. The power splits, burnups, and enrichments had to be extrapolated from the power linear power cases and, therefore, are of questionable accuracy. In one core the design pressure drop was maintained at 100 psi, while in the second a ΔP of 200 psi was allowed to reduce the sodium volume fraction. As can be seen in Table 7-5, the reduction in fuel cycle cost appears to be an additional ≈ 0.03 mill/kWh below the 23.75 kW/ft case if the pressure drop remains at 100 psi, and ≈ 0.08 mill/kWh if it is raised to 200 psi. In the latter 200 psi case, the cost of producing power may not be decreased significantly, however, since the cost of pumping power increases approximately 0.06 mill/kWh per 100 psi.

Some of the major assumptions used in this work were as follows:

Core Design Parameters

- Core height—24 inches
- Pin diameter—0.25 inch
- Spiral wire spacers
- Fission gas plenum
- Coolant $\Delta T = 300^\circ F$
- Average core burnup = 110,000 MWd/Te
- Local pressure drop = 15% of total
- Non-freestanding channels
- Fuel pellet density = 9.65 gms/cc (core), 10.00 gms/cc (blanket)
- Net plant efficiency = 40%
- Plant load factor = 80%
- Radial power shape constant
- Axial peaking factor varied with composition

Fuel Cycle Cost Parameters

Fuel cycle costs calculated using method similar to GEAP-3824⁽²²⁾
 Plutonium value = \$10/gm
 Inventory charge rate = 9%/year
 Fabrication plant designed for 10,000 MWe
 Fabrication time = 0.33 year
 Plutonium in-house time = 0.166 year
 Recovery and shipping costs (\$60/kg metal) core + axial blanket (\$50/kg metal) radial blanket
 Recovery and cooldown = 0.5 year
 Half of first core is charged off over the life of the plant at \$14%/year

7.4 TRANSIENT AND SAFETY ANALYSIS

7.4.1 FORE System

The restart problems encountered with the code have been found and corrected. A new tape with corrected restart, a new fuel center temperature model and improved time step procedure has been prepared and checked out, and will be entered in the program library as the production version of the code.

TABLE 7-4

EFFECT ON k_{eff} OF ANISOTROPIC SCATTERING

	50 cm Sphere				100 cm Sphere			
	Reactor 1		Reactor 2		Reactor 1		Reactor 2	
	k_{eff}	Error*	k_{eff}	Error	k_{eff}	Error	k_{eff}	Error
S ₄ - P ₃	0.81329	-	0.99827	-	-	-	-	-
S ₄ - P ₁	0.81326	0.00004	0.99808	0.00019	1.16955	-	1.21446	-
S ₄ - P ₀	0.81830	-0.00491	1.02539	-0.02712	-	-	-	-
Diff - P ₁	0.80725	0.00605	0.99406	0.00421	1.16877	0.00077	1.21402	0.00045
Diff - P ₀	0.81223	0.00107	1.02168	-0.02341	1.17125	-0.00171	1.22384	-0.00938
Diff-TR#1	0.80664	0.00665	0.98984	0.00843	1.16831	0.00124	1.21177	0.00269
Diff-TR#2	0.80823	0.00507	1.00032	-0.00204	1.16947	0.00007	1.21885	-0.00439
Diff-TR#3	0.80666	0.00663	0.99014	0.00814	1.16837	0.00118	1.21222	0.00224

* Error = $k_{eff}(S_4 - P_3) - k_{eff}$

TABLE 7-5

FUEL CYCLE COST SUMMARY (mills/kWh)

	Gas-Bonded Q _L = 19.3 kW/ft	Sodium-Bonded Q _L = 23.75 kW/ft	Sodium-Bonded (Extrapolated) Q _L = 30 kW/ft	Sodium-Bonded (Extrapolated) Q _L = 30 kW/ft ΔP = 200 psi
Inventory	0.3897	0.3442	0.3050	0.2757
Depletion	-0.4022	-0.4121	-0.4166	-0.4318
Fabrication*	0.2121	0.2169	0.2263	0.2221
Recovery*	0.1407	0.1462	0.1568	0.1585
First Core Capitalization*	≈0.0364	≈0.0301	≈0.0243	0.0233
				0.2478
				≈0.06
Total	0.3767	0.3253	0.2958	0.3078
Cost Differentials		0.0514	0.0295	
		0.0809		

* Possible extra cost of sodium-bonding is not considered.

A condensed kinetics model, ATD, has been written and debugged. The code employs an explicit heat transfer scheme, as does FORE II, so that it can be used as a check on FORE II. It also has the capability of dividing one channel into many axial sections (25) for examining numerical problems inherent in low flow accidents.

7.4.2 Fuel Failure Propagation Study

A complex heat transfer model has been designed for the nodal computer program THTD. The initial complexity is caused by the need for a flexible model to handle mechanically blocked channels, vapor blocked channels, cladding temperature distribution, radial flux distribution, flow reduction, and a combination of such effects. The initial effort in designing this model has resulted in an input which is quickly and inexpensively modified to correspond to many situations associated with a small group of fuel pins. A diagram of the nodal structure is shown in Figure 7-1. Typical blockage area is the space between four adjacent pins, by symmetry only the space EFGH needs to be considered. Five layers of nodes in the cladding will give a comprehensive temperature gradient which can be used in thermal stress or thermal bowing studies.

The model described has been set up and run under steady-state conditions. Output has been checked against the VELVET program for slug flow conditions. Comparison of outputs confirm that both programs are consistent.

The THTD program has a comprehensive restart capability, and it is hoped to use this model to trace the probable chain of events associated with failure propagation.

7.5 THERMAL HYDRAULICS AND FUEL ANALYSIS

7.5.1 Spacer Analysis

An examination of the thermal effects of spacers on the pins on the outer edge of a bundle has been initiated. The analysis employs a three-dimensional model with slug sodium flow, and is being run using the THTD code. A parametric study of various spacer sizes is being performed to attempt to define an optimum spacer thickness that will minimize thermal bowing gradients in the pin.

7.5.2 Fuel Swelling Model Development

A 2-year program of work was submitted to AEC-RDT which involves development of a theoretical model to predict the swelling of fast reactor fuel by fission product accumulation, correlation of the model with experimental data, and development of a computer code to predict fuel expansion.

A model was developed to predict fission gas bubble size as a function of burnup, temperature, and the surface tension and viscous properties of the fuel. The fraction of fuel most susceptible to swelling was also determined as a function of power level assuming a constant thermal conductivity. This work has been previously reported in the Twenty-Fourth Quarterly Report.⁽⁷⁾

Work is now under way to predict how the bubbles change in size and affect fuel growth once they break away from their anchor sites and begin to move to the fuel center. Evaluation is also under way to assess the effect of cladding constraint on the bubble growth process.

FIGURE 7-1 MODEL FOR TEMPERATURE PROFILES IN SODIUM ONLY

The data from the F2 irradiation capsules are being analyzed. The cladding profile and diameter after irradiation are being compared to the pre-irradiation condition to see the effect of local pellet density, fuel-cladding gap, and local power. The effect cladding material and thickness is also being investigated.

A computer program has been developed to predict the temperature of mixed oxide fuel as a function of linear power. Three regions of different density are assumed. The program has the option of changing the density of the three regions with burnup, and thereby obtaining a temperature profile that is burnup dependent.

SECTION VIII

TASK L—CLADDING DEVELOPMENT

8.1 GENERAL

The technical and economic feasibility of ceramic-fueled fast reactors depends on the attainment of high levels of performance of the fuel system, especially the survival of the cladding at temperatures and specific powers of practical interest. The present status of testing and analysis of potential cladding material with regard to alloys and environmental variables indicates good prospect that adequate initial performance of cladding for fast reactor fuel elements can be obtained by using carefully specified materials within the range of compositions of commercial processing methods. 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 April 31, 1968, the approximate total fluences, based on EBR-II core physics, are 4.6, 2.0, 1.3, and 0.7×10^{22} n/cm², respectively. Complete descriptions of these series and their experimental objectives have been given previously. (4, 12)

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

A description of these capsules has been presented earlier with results regarding the post-irradiation testing of tubular and flat tensile samples. (6, 7)

Further testing of tensile samples with the carbide agglomerated treatment was completed. These specimens were annealed at 1800°F/15 minutes and air quenched followed by a heat treatment at 1650°F for 24 hours. Tensile tests were performed in a manner similar to that previously reported. (6)

The results are presented in Table 8-1 for the five candidate cladding alloys. Tensile tests were generally performed in duplicate.

The effects of neutron fluence ($E_n > 1$ MeV) upon the uniform tensile elongation, and yield strength, are shown in Figure 8-1.

Figure 8-2 shows how the total elongation and yield strength respond to decreasing strain rates at constant fluence at 1300°F;

these materials received $2.4 - 3.0 \times 10^{21}$ n/cm² ($E_n > 1$ MeV).

The response of carbide-agglomerated alloys to tensile evaluation is more complex than that of mill-annealed structures. It is apparent from Figure 8-1 that carbide agglomerated Type 304 results show an anomalous strain behavior with increasing fast fluence. At approximately 0.8×10^{21} n/cm² ($E_n > 1$ MeV), the uniform strain begins to increase with increasing fast fluence to 13% at 3×10^{21} n/cm² ($E_n > 1$ MeV). Carbide agglomerated Types 321 and 347 both have larger uniform elongations, at comparable fluences, than the mill-annealed structures. (6) However, both carbide agglomerated Type 316 and Incoloy 800 have lower uniform elongations than comparably-exposed mill-annealed structures.

The effect of fast fluence on yield strengths of carbide-agglomerated alloys (Figure 8-2) is similar to that on mill-annealed material; however, the degree of radiation strengthening is generally less. When an alloy showed increased radiation strengthening with fluence, the degree of embrittlement also increased. When an inflection occurred in the yield stress versus fluence curve, generally the degree of embrittlement was reduced, e.g., Types 304 and 347.

Carbide-agglomerated Type 316 and Incoloy 800 data revealed an increase in total elongation with decreasing strain rate (Figure 8-3), while data for mill-annealed structures showed the reverse. Yield stress of Type 316 was observed to increase with slower strain rates, contrary to mill-annealed structures. The effect of strain rate on the yield stress of Incoloy 800, however, followed similar trends in both structural conditions.

* 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 OF
AUSTENITIC STAINLESS STEEL WITH CARBIDE AGGLOMERATES STRUCTURE⁽¹⁾

Alloys	Neutron Exposure ($E_n > 1$ MeV) $10^{21} \text{ n/cm}^2 \times 10^{21}$	Test Temperature (°F)	Strain Rate (min ⁻¹)	Uniform ⁽²⁾ Strain (percent)	Total ⁽²⁾ Strain (percent)	Yield Stress 0.2% (ksi)	Ultimate Stress (ksi)
Type 304	0.22	1300	0.02	7.6—9.0	15.1—15.6	25.3—27.8	32.6—33.8
	0.76	1300	0.02	7.7—8.0	10.5—11.0	26.4—28.7	35.6—37.9
	2.4	1300	0.02	10.7—12.4	12.1—14.2	16.4—16.7	26.8—28.6
	3.0	1300	0.02	13.4	18.0	16.2	26.4
	2.5	1100	0.02	22.2	23.0	15.3	39.5
	3.0	1100	0.02	15.5	16.8	17.5	39.2
Type 316	0.30	1300	0.02	6.1—6.3	8.9—11.2	21.4—27.3	35.5—36.3
	0.96	1300	0.02	5.5—6.7	9.9—11.0	38.1—38.7	43.1—43.4
	3.10	1300	0.01	1.9—3.7	2.1—3.9	30.7—32.7	36.7—37.3
	3.30	1300	0.02	5.7	7.1	19.0	31.2
	3.90	1300	0.01	1.6	2.1	43.8	49.0
	3.90	1300	0.02	1.7	2.1	29.8	41.1
	3.30	1300	0.0002	0.4—1.0	12.5—14.3	20.6—24.2	20.9—24.2
	3.10	1100	0.02	11.6—20.9	15.4—22.1	15.9—51.3	43.0—62.7
	3.30	1100	0.02	11.3	15.0	18.5	52.7
Type 321	0.76	1300	0.02	6.6—6.9	14.8—15.2	23.8—25.2	28.9—30.4
	3.0	1300	0.02	2.2—2.7	5.3—5.8	27.4—31.9	29.3—34.6
	2.5	1100	0.02	1.7—1.7	2.6—3.1	57.4—57.7	59.8—60.3
	2.4	1500	0.02	2.0—2.4	9.8—10.1	14.1—14.4	15.2—15.6
Type 347	0.22	1300	0.02	11.2—11.4	26.6—32.2	22.4—23.1	31.6—32.2
	0.76	1300	0.02	2.0—8.2	5.0—10.0	19.9—31.1	29.9—32.2
	3.0	1300	0.02	8.1—8.6	10.3—11.8	21.4—21.9	29.4—29.8
	2.5	1100	0.02	8.1—8.3	9.4—10.5	39.6—41.3	47.6—48.1
	2.4	1500	0.02	3.8—4.1	8.9—10.2	13.0—13.2	15.4—15.7
Incoloy 800	0.76	1300	0.02	2.6—3.2	5.3—8.9	35.3—38.6	37.8—41.0
	3.0	1300	0.02	1.6—2.8	3.6—4.4	35.9—36.1	37.5—42.8
	2.4	1300	0.0002	0.4—0.9	6.3—7.9	19.6—34.9	19.6—36.9
	3.0	1300	0.0002	0.5	8.2	31.0	31.3
	3.0	900	0.02	9.2	10.7	53.1	74.5
	2.5	1100	0.02	2.4—2.5	2.7—3.1	60.4—63.1	67.5—70.2
	2.4	1500	0.02	1.0—1.6	9.6—10.5	15.4—16.5	15.8—17.0

(1) Annealed at 1800°F for 15 minutes; air quenched, and heat treated at 1650°F for 24 hours, air quenched.

(2) Original gage length—1 inch.

FIGURE 8-1. THE EFFECT OF FAST NEUTRON FLUENCE ON 1300°F DUCTILITY OF CARBIDE AGGLOMERATED AUSTENITIC STAINLESS STEELS AND INCOLOY 800 IRRADIATED AT 1000°F - 1200°F

FIGURE 8-2. THE EFFECT OF FAST NEUTRON FLUENCE ON 1300°F STRENGTH OF CARBIDE AGGLOMERATED ANSTENITIC STAINLESS STEELS AND INCOLOY 800 IRRADIATED AT 1000°-1200°F

FIGURE 8-3. THE EFFECT OF STRAIN RATE ON THE 1300°F TENSILE PROPERTIES OF TYPE 316 AND INCOLOY 800 IRRADIATED AT 1000°-1200°F (M.A.—MILL ANNEALED; FLUENCE $-6.2-6.7 \times 10^{21} E_n > 1 \text{ MeV}$; C.A. — CARBIDE AGGLOMERATED; FLUENCE $-3.0-3.3 \times 10^{21} E_n > 1 \text{ MeV}$)

Optical metallography and transmission electron microscopy are presently being used to correlate microstructure with the observed tensile properties for the alloys tested. Creep tests will also be performed to clarify the effect of strain rate on mechanical properties.

8.2.3 Stress-Strain History of Irradiated Materials Capsule L2Q

An effort was undertaken to define the stress-strain history, during irradiation in EBR-II, of the L2Q (Type 304) materials capsule. The complete irradiation history for this capsule was reported earlier.⁽⁷⁾

Through consideration of the gamma-heat generation of all components and the pressure rise caused by temperature-volume changes, this capsule was designed to reach a temperature of 1200°F and a hoop stress of 8800 psi in Row-4 EBR-II.

Post-irradiation examination indicated that the irradiation temperature, (T), as monitored by fusible metal thermometers, was 900° to 1070°F and that nonuniform diametral deformation (as much as 12%) resulted.⁽⁷⁾ Since design temperature was not achieved, the in-reactor stress level was lower because the internal pressure is related to the temperature. Moreover, the localized deformation that resulted would produce a condition in which the in-reactor stress would be a function of the existing deformation.

It was determined, by using the thermometry data, that the initial hoop stress was only 6,900 psi, 22% lower than design conditions. With the post-irradiation profilometry curves and the stress analysis method described earlier,⁽⁷⁾ the hoop stress at the end of the reactor exposure was calculated and the result are shown in Figure 8-4.

It can be seen that the observed localized deformation caused the stress to decrease with time in regions of the test section where the diametral deformation was <8%. Where

the creep deformation was >8%, the stress increased with time. The total stress variation along the capsule was about 20%.

The irradiation exposure was about 5000 hours and assuming a linear creep rate throughout the exposure (possibly a high estimate), the creep rate can be estimated to be $2.4 \times 10^{-3}\%$ /h. Since the published creep rate on unirradiated Type 304 at 1000°F and 7500 psi (the maximum stress located at the peak deformation region) is about $1.5 \times 10^{-6}\%$ /h, it appears that the in-reactor creep rate was 10^3 times greater.

There are four applicable parameters that could have altered the creep rate: (1) stress, (2) temperature, (3) neutron flux, and (4) stress cycling. At 1000°F and 7000 psi, a 50% increase in stress would increase the creep rate only six times. It is, therefore, unlikely that any small variation in stress could account for the higher creep rates observed. Temperature, however, can dramatically alter creep rates. A 100°F increase in temperature, at constant stress, can increase the creep rate approximately tenfold. Thus, a 300°F rise in temperature alone could produce the necessary creep rate.

It also may be possible that neutron flux can increase the creep rate. The unirradiated creep rate in austenitic stainless steel is controlled by the rate of dislocation climb, a diffusion process by which atoms or vacancies are supplied to the dislocation. (Time-dependent deformation at $\geq 1/2 T_m$ is a thermally-activated process; the activation energy approximately that of self diffusion). Since fast-neutron bombardment with the lattice produces a vacancy concentration orders of magnitude greater than that found from the thermal equilibrium concentration, dislocation climb (thus creep rate) could be enhanced. Based on available data,^(23, 24, 25) the effect of neutron radiation on creep properties is still undefined. Bloom⁽²³⁾ has found that irradiation in ORR at 650°C increased the in-reactor creep rates, threefold at

15,000 psi, for Type 304 when compared to unirradiated controls. These differences increased with lower stresses. However, the British⁽²⁴⁾ found that for Type 316 irradiated in the DMTR (thermal), in-pile creep rates decreased tenfold at 1200°F and 18,000 psi hoop stress. They also observed no change in post-irradiation creep rates at 1200°F for the similar Type 316 irradiated in the DFR.

In-pile creep tests⁽²⁵⁾ were also performed in the ETR on Type 304 stainless steel where it was observed that minimum creep rates were essentially the same as those for unirradiated Type 304. These tests were conducted over a wide range of temperature (923° to 1382°F) and stress (10,000 to 30,000 psi). The effect of stopping a creep test and reloading after a no-load rest (lowering the temperature during irradiation as in a reactor shutdown reduces the hoop stress on the test section) is to produce, upon retesting, a higher creep rate.⁽²⁶⁾ Since L2Q experienced 18 power cycles in EBR-II with a 400°F change in temperature/cycle, stress cycling may be a factor in accounting for the anomalously high in-pile creep rate.

Some question has been raised regarding the maximum temperature attained in the L-2' series. Therefore, selected irradiation temperature sentinels will be reexamined for meltdown.

8.3 MICROSTRUCTURAL STUDIES

8.3.1 Electron Microprobe Analytical Facility

The development of an analytical facility capable of handling and analyzing oxide fuel is complete. The facility consists of two electron microprobe x-ray analyzers: a standard commercial instrument to handle low activity or routine specimens and a shielded instrument for highly radioactive specimens. Both instruments are currently installed at Vallecitos. The former has been operating for several months; the latter is undergoing the final stages of checkout.

Utilization of the shielded microprobe as an out-of-cell facility depends in large measure on the ability to prepare alpha-clean specimens of irradiated mixed-oxide fuel with gamma activities commensurate with the shielding capability of the instrument. As established by design, this level corresponds to a gamma source of not more than 10 curies and represents a 62-mil thick disc of mixed oxide fuel irradiated to 100,000 MWd/Te and cooled for 60 days. A technique for preparation of thin discs of irradiated fuel has been developed and was demonstrated in the recent metallographic preparation of a radial cross section from fuel pin F2F.

A massive section of F2F was first potted in hysol to retain the fuel during subsequent handling. A transverse cut near the base of the section provided a 5/32-inch-high disc which was repotted to further assure fuel retention. The disc was then mounted in an aluminum specimen holder which contained depth-gaging holes as shown in Figure 8-5. (This procedure is similar to that discussed, but not demonstrated, in a previous report.)⁽²⁷⁾ The specimen was ground in stages to a final disc height of 50 to 70 mils, as determined by the depth indicators. A height of 65 mils was subsequently established by micrometer measurements. Activity survey of the specimen after final grinding yielded a dose rate of 100 mR/h at 6 feet, a level well within the safe handling capability of the instrument.

An activity survey along the center surfaces of the shielded microprobe, with a Cobalt-60 source in the specimen chamber, was performed to assess the shielding capability of the instrument. The Co-60 source provided a dose rate of 2 R/h at 6 feet in air, a somewhat higher activity level than the shielding was designed to contain. The survey showed a maximum outer surface dose rate of 60 mR/h at an isolated point on the electron column and a 6 mR/h rate at the eyepiece of the optical microscope. The shielding was therefore judged to be acceptable.

FIGURE 8-4. THE EFFECT OF DEFORMATION ON THE IN-REACTOR HOOP STRESS OF MATERIALS CAPSULE L-2Q

MATERIAL: ALUMINUM

FIGURE 8-5. MICROPROBE SPECIMEN MOUNT

8.4 FUEL-CLADDING ANALYTICAL STUDY

The nonstoichiometric behavior of urania-plutonia solid solutions makes difficult to define fuel compatibility with surrounding materials. Since the oxygen activity of the fuel varies with the composition and temperature, the experimental situation must be carefully analyzed. This analytical study discusses the tendency of oxide fuels to react with stainless steel cladding and sodium. These materials are strong absorbers of oxygen and, if in sufficient supply, could govern the overall oxygen activity level in the fuel element. The objective of this study was to establish the types of chemical interactions one might encounter in mixed oxide fuel systems based on the application of physical-chemical principles to available thermodynamic data.

8.4.1 Compatibility Under Isothermal Conditions

The reaction tendency between urania-plutonia fuel and sodium or the cladding alloy is almost exclusively a function of the oxygen activity. In a fuel pin, the source of oxygen is predominantly the fuel phase. The oxygen sinks may be the chromium in the alloy and sodium present for bonding purposes or introduced during cladding failure. They will function as sinks if the chemical potential of oxygen in the fuel phase exceeds (is more positive than) that for the oxidation of chromium or sodium. Markin and McIver⁽²⁸⁾ have published results of the partial molar-free energy of oxygen, $\Delta\bar{G}_{O_2}$, for urania-plutonia oxides at various oxygen-to-metal ratios. Weeks⁽²⁹⁾ has estimated $\Delta\bar{G}_{O_2}$ for the sodium-oxygen system. Figure 8-6 shows a comparison of the oxygen potential for each of the two materials. The results on the right are combined for UO_{2+x} (hyperstoichiometric compositions) and $U_{0.8}Pu_{0.2}O_{2-x}$ (hypostoichiometric compositions). It is evident that hypostoichiometric fuel has oxygen activities which are too low for reduction by sodium unless the oxygen content in the sodium is

below about 1000 ppm. On the other hand, the hypersotichiometric fuel is easily reduced by sodium.

The oxygen potentials at 700°C for some of the alloying constituents in stainless steel are shown in Table 8-2. It is seen that Cr_2O_3 has the largest oxygen affinity which is similar to the oxygen affinity for sodium to form Na_2O . Hyperstoichiometric fuel also listed in the table will readily oxidize not only the chromium, but also the iron and nickel. As the composition of the fuel passes from hyper- to hypostoichiometric, its oxygen potential plunges rapidly to a value below that for the formation of chromium oxide. These reaction conditions have been verified experimentally at 700°C in sealed capsules containing fuel at oxygen-to-metal ratios of 2.04, 2.00, and 1.97. In long-term heat treatments at this temperature, it was observed that only the 2.04 composition results in oxidation of the alloy.⁽³⁰⁾

8.4.2 Compatibility Under Nonisothermal Conditions

It has been established that, in a thermal gradient, fuel O/M increases at the colder region. The relationship between the composition of MO_{2-x} and temperature at steady-state follows the general form.

$$\ln x = \frac{Q^*}{RT} \quad (2)$$

Fuel pins which are producing heat uniformly have large temperature gradients. The temperature distribution for a solid cylinder is described as follows:

$$T_r - T_b = \frac{Q}{4\pi k} \left(1 - \frac{r^2}{b^2} \right) \quad (3)$$

where T_r and T_b represent the temperatures at radius, r , and the outer radius, b . Q is the heat generation rate and k is the thermal conductivity. By combining Equations 2 and

3, we can relate the composition to the position, r , and by integration over the radius, determine the average O/M of the fuel. The calculation assumes that the boundary conditions at $r = b$, namely x_b and T_b , are known. T_b is independent of the heat removal conditions external to the fuel column and is frequently established by overall design requirements. x_b in general may be any quantity; however, in the presence of an oxygen sink, x_b is fixed by the oxygen affinity of the sink. For temperatures of 9000°C, in the presence of chromium, it is seen that x_b will be very small, i.e., ~ 0.005 .

The centerline temperature, T_o , which is equal to $\frac{Q}{4\pi k}$ is a parameter that may be adjusted for various heat generation rates. The volume fraction of fuel between b and r is expressed by

$$V = \frac{\int_b^r 2\pi r dr}{\int_b^o 2\pi r dr} \quad (4)$$

which becomes simply

$$V = \frac{T_r - T_b}{T_o - T_b} \quad (5)$$

for the case of uniform heat generation in a solid cylinder. The mean value for the O/M deficiency, x , may be calculated as follows:

$$\langle x \rangle = \frac{\int_{T_b}^{T_o} x(T_r) \frac{dV}{dT_r} dT_r}{\int_{T_b}^{T_o} \left(\frac{dV}{dT_r} \right) dT_r} \quad (6)$$

Substitution for dV (the differentiation of Equation 5) and incorporation of Equation 2 yields:

$$\langle x \rangle = \frac{x_b}{T_o - T_b} \exp - \left(\frac{Q^*}{RT_b} \right) \int_{T_b}^{T_o} \exp \left(\frac{Q^*}{RT_r} \right) dT_r \quad (7)$$

This integral, which must be calculated by numerical methods, provides the basis for evaluating the amount of oxygen available for reaction with an oxygen sink over a wide range of conditions in a fuel element. $\langle x \rangle$ represents the average oxygen deficiency in the fuel and is the cross-over value for reaction of the

TABLE 8-2
OXYGEN POTENTIAL FOR FORMATION OF
VARIOUS OXIDES IN STAINLESS STEEL SCALES AT 700°C

Reaction	ΔG kcal/mole of O_2 at 700°C	O moles/cm ² /mil
Fe_3O_4/Fe_2O_3	-50	4.26×10^{-4}
$U_{0.8}Pu_{0.2}O_{2.04}$	-55.7	---
Ni/NiO	-74	2.84×10^{-5}
0.1 wt % C in Fe/CO (1 atm)	~ -87	1.75×10^{-6}
FeO/Fe_3O_4	-90	3.78×10^{-4}
0.1 wt % C in Fe/CO ₂ (1 atm)	~ -91	3.5×10^{-6}
$Ni + Cr_2O_3/NiO \cdot Cr_2O_3$	~ -93	7.1×10^{-5}
Fe/FeO	-97	2.84×10^{-4}
$Fe + Cr_2O_3/FeO \cdot Cr_2O_3$	~ -116	1.44×10^{-4}
Na/Na_2O	-134	---
Cr/Cr_2O_3	-137	1.1×10^{-4}
Na/100 ppm O in Na	~ -147	---
$U_{0.8}Pu_{0.2}O_{1.98}$	-153	---

fuel with an oxygen sink. If the initial oxygen deficiency of the lattice is above $\langle x \rangle$, then no reaction would be expected. In this way, it is possible to determine the appropriate starting oxygen-to-metal ratio for fuel operated at a given set of reactor power conditions to ensure that it will be compatible with stainless steel or sodium.

By evaluating the integral of Equation 7 at various center-line temperatures, and a fixed boundary temperature of 900°C, one can determine the maximum fuel O/M that will be compatible with a given oxygen sink. Some representative values, based on a Q^* of -10 kcal/mole (obtained from thermal gradient tests), are shown in Table 8-3 for reaction with (1) chromium, (2) sodium at saturation, and (3) sodium at 100 ppm of oxygen.

These calculations require a precise determination of x_b , the composition in equilibrium with the oxygen sink. From the isothermal measurements of Markin and McIver, at temperatures between 700° to 1000°C, x_b is <0.005 . Attempts are being made to evaluate this term more precisely by experimental methods before carrying out the calculation further. Qualitative examination of the above derivations indicates that $\langle x \rangle$ will be greater than x_b for a negative value of Q^* ; hence, the compatibility condition requires that the initial composition of the fuel should be more hypostoichiometric in the nonisothermal case than in the isothermal one.

Experimental work on stoichiometric urania-plutonia fuel heated in a reactor (nonisothermal) has shown evidence of an oxide layer (which is likely Cr_2O_3) adjacent to the cladding. Under isothermal conditions, where the cladding was at the same temperature as in the reactor experiment, no oxide scale was evident which supports the general validity of the above calculation.

8.5 CLADDING OPTIMIZATION STUDIES

8.5.1 Ring Ductility Test

Evaluation of the barreling-mandrel-ring ductility test has been completed. This test was described earlier⁽³¹⁾ and, with slight modifications from the original dimensions of the barrel and ring, successful tests were completed. The data obtained were then reduced to meaningful engineering strength and ductility parameters.

8.5.1.1 Description of Test

The concept of a barreled cylinder mandrel deformed in compression is shown in Figure 8-7. Circumferential strain is imparted to the ring by compressing the mandrel at a constant rate. Homogeneous strains in the rings are possible for diameter increases as large as 10%. By providing proper lubrication between the mandrel and the bearing block, mandrel barreling can be kept to a minimum over this strain range.

The mandrels used were made of Type 304 stainless steel (0.219-inch in diameter by 0.250-inch high). The rings tested were 0.250-inch outside diameter by 0.015-inch wall by 0.075-inch high. A compression cage was used to compress the mandrels, thus minimizing the heat loss from the furnace since only 1/2-inch diameter bars are necessary to support the load for the test. The cage, whose members are in tension, also aids in maintaining the load parallel to the axis of the mandrel, thus preventing buckling. The load was applied and monitored by an Instron Testing Machine.

The ring strain was obtained from calibration curves which relate decreases in mandrel height to increases in mandrel diameter. The calibration curves for the mandrel were not measurably affected by the ring, i.e., the increases in mandrel diameter with decreasing mandrel height were the same with and without the ring.

FIGURE 8-6. COMBINED PLOT OF ΔG_{O_2} VERSUS OXYGEN COMPOSITION IN SODIUM, UO_{2+x} AND $U_{0.8}Pu_{0.2}O_{2-x}$

To obtain both strength and ductility data, two similar tests were performed: one test with just the mandrel and the other with the mandrel plus a ring. The resulting load versus deflection curves, Figure 8-8, differ and this difference can be interpreted as a measure of the tensile properties of the ring. Fracture of the ring is indicated by a sudden drop in load as shown for notched ring.*

* It was found that a notch was necessary to induce fracture so that the technique could be checked out using a ductile material.

The total elongation of the ring was measured from the point where deformation of the ring begins as observed by a load difference to the point where a drop in load occurs. The stress-strain curve (Figure 8-9) for the ring can be estimated by an energy difference method based on the assumption that the energy applied to the mandrel during the compression test (equal to the area under the compression load-deflection curve for the mandrel) is equal to the absorbed strain energy (both thermal

TABLE 8-3

MAXIMUM O/M OF $U_{0.8}Pu_{0.2}O_{2-x}$ TO BE
COMPATIBLE WITH VARIOUS OXYGEN SINKS IN NONISOTHERMAL STATE

Centerline Temperature (°C)	Maximum Fuel O/M		
	Chromium	Sodium at Saturation	Sodium at 100 ppm Oxygen
900	1.995	1.995	1.988
1500	1.988	1.988	1.972
2000	1.980	1.980	1.953
2500	1.970	1.970	1.934

FIGURE 8-7. BARRELING CYLINDER DUCTILITY TEST CONCEPT

FIGURE 8-8. TYPICAL LOAD-DEFLECTION CURVE FOR RING SPECIMENS, MANDREL MATERIAL - TYPE 304 STAINLESS STEEL, RING MATERIAL - 30% COLD WORKED INCOLOY 800

FIGURE 8-9. STRESS-STRAIN CURVES FOR 30% COLD-WORKED INCOLOY 800 RINGS

and residual) in the mandrel plus the friction energy at the loading surfaces which causes barreling. The additional energy required for compression of the mandrel with the ring is then the strain energy absorbed by the ring since the friction and absorbed energy for the mandrel is the same for both tests.

The Instron load chart actually plots load (P) versus crosshead motion (H) since both chart and crosshead move at constant rates. Therefore, if P_i is the load without the ring and P'_i the load with the ring, the energy absorbed by the ring (E_R) is then

$$E_R = \frac{1}{V_R} \sum_{i=0}^n (P'_i - P_i) \Delta h_i \quad (8)$$

where V_R = ring volume.

By computing the ring energy as a function of diametral ring strain (e_R), an energy versus strain curve can be obtained for the entire ring test. The stress in the ring (S_R), then, is equal to the slope of this curve since by definition,

$$dE = S de. \quad (9)$$

Figure 8-8 is the stress-strain curve for the load versus deflection curve shown in Figure 8-7.

The results from ring tests described above are quite similar to the results from an uniaxial tensile test and qualitatively represents the transverse tensile properties in uniaxial tension. Tests are presently under way to quantitatively compare the transverse and longitudinal properties of tubes using the tensile, burst, and ring tests.

The total strains obtainable from the ring test will differ substantially from strains obtainable from tensile tests since surface shear loads will be developed from friction between the mandrel and ring. This friction will support necking instabilities. This

phenomenon became readily apparent when excessively large strains, >55% were obtained in the ring test for 30% cold worked Incoloy 800. Total elongation for standard tensile tests were 5 - 6%. Therefore, it is expected that the ring test will result in a total strain value as determined by $(\Delta D/D)$ for the ring equal to the fracture strain value as determined by reduction in area from the tensile test. The tests of irradiated specimens may not show significant differences in these values since it has been reported⁽³²⁾ that due to the more brittle nature of irradiated material fracture strain approaches total elongation values.

A significant amount of testing has been conducted on tubing using internal gas pressure. With proper considerations of their biaxial nature, these tests correspond to tensile tests in the transverse direction since no supporting surface shear can be developed. In reactor operation, fuel-cladding interaction will provide a certain degree of friction shear support for instabilities in the absence of fuel cracks but perhaps not to the extent that the solid mandrel does. The addition of the ring mandrel test, therefore, completes the bound on loading character. At one boundary is the pressure tube or tensile specimen with no instability support; on the other end the ring mandrel test with supporting friction. In between is actual reactor service conditions.

The possible existence of cracks in fuel at the surface may be expected to produce high local strain regions in the cladding. Proper analysis of such regions will provide an additional requirement for experimental values of fracture strain such as those obtained in the ring mandrel tests.

8.5.2 Characterization Program of EBR-II Materials Capsule Alloys

This program is directed toward adequately characterizing the mechanical properties and microstructure of FCR alloys presently used in active irradiation experi-

ments and those planned for future experiments. Characterization includes tensile testing as-received materials and materials after selected thermal exposure. Two basic kinds of materials are being irradiated:

- a. Commercially available material in the form of tubing.
- b. Specially prepared commercial and experimental materials in the form of sheet.

The scope of work is presently , separately for each type of material.

8.5.2.1 Commercial Available Tubing

This category of materials includes all five FCR cladding candidate alloys which were included as flat tensile samples in the Series L-2, L-2', L-4, L-4', L-6, L-10, and L-15 materials capsules. The alloys and their compositions are shown in Table 8-4. The starting material for all alloys was 0.250-inch outside diameter by 0.015-inch wall tubing. The tubing was slit longitudinally in half and flattened into strips. Tensile samples were punched from these strips and then given heat treatments to produce two microstructures: (1) as-received mill-annealed fine grain size, and (2) fine grain size with carbides agglomerated at grain boundaries. The fabrication of these samples is shown diagrammatically in Figure 8-10. These materials will be tested in the as-fabricated condition and after 1000-, 2000-, 4000-, and 8000-hour exposures at 1000° and 1200°F.

Results from 1300°F tests for as-received mill-annealed material, Table 8-5, show that the Type 304 tested is slightly weaker than the other four alloys. Total elongation values were above 30% for all five alloys tested.

8.5.2.2 Commercial Sheet Material

This category of materials includes the three alloys in the form of commercial sheet

stock, which were included in the Series L-14 and L-15 materials capsules. The alloys and their compositions are shown in Table 8-6. The starting material in all cases was 0.062-inch-thick sheet. Strips were cut parallel to the rolling direction and cold reduced to 0.015 inch thick in two ways, resulting in 25% and 75% cold-reduced fine grain strip.

Tensile samples were punched from these strips before any final heat treatments were performed. Selected annealing conditions were predetermined⁽¹²⁾ for the various cold-worked materials to produce three different microstructures: (1) a fully-annealed fine grain structure (ASTM-10), (2) a fully-annealed coarse grain structure (ASTM-6), and (3) a fine substructure pinned by intra-granular carbides (Garafalo) heat treatment). The mechanical thermal treatments given these samples are shown in Figure 8-11.

These materials will be tested in the as-fabricated condition and also after an extensive thermal exposure of 1200°F/4000 hours (similar to samples irradiated in EBR-II) to define the effect of thermal exposure alone on the tensile properties and microstructure. The thermal exposure was initiated on March 25, 1968.

8.6 MATERIALS MANAGEMENT

The selection of austenitic stainless steel as the prime candidate cladding for the liquid metal fast breeder reactor (LMFBR) has prompted a broad experimental study to evaluate the effect of metallurgical and dimensional variables on the elevated temperature mechanical properties of this class of materials. This comprehensive effort is expected to serve two basic objectives:

- a. Define the highest quality commercial grade material that can be consistently and economically produced in large quantities, and
- b. Establish realistic tolerances on dimensional and surface imperfections.

TABLE 8-4
CHEMICAL COMPOSITION OF TUBE MATERIALS

Alloy	Heat No.	Ni	Cr	Fe	C	Mn	Si	Cu	Mo	P	S	B*	N*	Other
Type 304	125653	9.15	18.42	Balance	0.06	1.55	0.41	0.27	0.50	0.025	0.010	16	540	
Type-316	850145	13.16	17.63	Balance	0.04	1.90	0.52	0.15	2.81	0.021	0.006	7.8	630	
Type 316	155882	12.78	17.18	Balance	0.05	1.74	0.52	N.D.	2.56	0.018	0.019	N.D.**	N.D.	
Type 316-L	133310	13.31	17.39	Balance	0.02	1.80	0.66	0.07	2.20	0.017	0.008	9.5	500	
Type 321	31053	10.62	18.83	Balance	0.04	1.06	0.57	N.D.	N.D.	0.016	0.013	N.D.	N.D.	0.46 Ti
Type 321	800738	9.53	17.13	Balance	0.05	1.35	0.64	0.15	0.29	0.019	0.011	N.D.	N.D.	0.53 Ti
Type 347	99973	9.32	18.20	Balance	0.05	1.16	0.72	0.20	0.14	0.023	0.009	12.0	560	0.91 Cb + Ta
Incoloy 800	4167	32.00	20.44	46.09	0.05	0.99	0.25	0.33	N.D.	N.D.	0.007	12	930	0.51 Ti + 0.40 Al

* Impurities in ppm.

** N.D. refers to not determined.

TABLE 8-5

TENSILE TEST RESULTS ON MILL-ANNEALED AS-RECEIVED SPECIMENS

Material Identity	Test Temperature (°F)	Strain Rate (min ⁻¹)	0.2% Offset Yield Stress (ksi)	Ultimate Stress (ksi)	Total Elongation (%)
304-MA	1300	0.02	21.8—22.7	31.3—31.8	35.4—42.1
316-MA*	1300	↑ ↓	21.2—26.6	35.9—37.4	44.0—50.8
312-MA	1300		26.5—31.0	32.2—32.2	31.0—31.8
347-MA	1300		29.7—32.9	36.0—36.1	32.9—37.9
Incoloy 800-MA	1300	0.02	28.6—29.6	37.3—37.8	40.2—43.3

* Type 316 tested was from Heat No. 155882.

TABLE 8-6

CHEMICAL COMPOSITION OF SHEET MATERIAL

Alloy	Number	Ti	Ni	Cr	Fe	C	Mn	Si	Cu	Mo	P	S	Co
Type 316	820467	N.D.	13.12	17.32	Balance	0.049	1.83	0.67	0.13	2.17	0.015	0.007	0.13
Type 321	156289	0.52	9.70	17.78	Balance	0.06	1.40	0.54	0.30	0.26	0.027	0.018	0.12
Incoloy 800	HHA4534-A	N.D.	30.53	20.64	46.93	0.05	0.98	0.40	0.44	N.D.*	N.D.	0.010	N.D.

* N.D.—Not Determined.

FIGURE 8-10. FLOW DIAGRAM FOR FABRICATION OF TENSILE SAMPLES FROM COMMERCIAL TUBE

FIGURE 8-11. FLOW DIAGRAM FOR FABRICATION OF TENSILE SAMPLE FROM COMMERCIAL SHEET.

Since the adoption of an acceptable tubing specification for reactor application depends in large part on the establishment of allowable dimensional tolerances; the latter program has been given immediate priority. This phase of work will attempt to:

- a. Establish a statistical relationship for the size and type of tubewall surface defects and dimensional irregularities which are common in commercial tubing purchased to rigid specifications,
- b. Determine the effect of various levels of defects and dimensional variance on high-temperature mechanical properties and
- c. Establish the maximum allowable variation from perfection in dimensions and surface defects that can be tolerated in as-received material, during fuel element handling and processing, shipping, and reactor loading.

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