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DEVELOPMENT REPORT

FAST REACTOR MIXED-CARBIDE FUEL ELEMENT
DEVELOPMENT PROGRAM

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1. INTRODUCTION

The Fast Reactor Mixed-Carbide Fuel Element Development Program is aimed at the establishment of mixed carbides as a competitive LMFBR fuel. The major objective of the program is to develop fast breeder reactor fuel rods of (UPu)C clad with stainless steel or higher strength alloys that will perform satisfactorily to a minimum of 50,000 Mwd/T and a maximum of 100,000 Mwd/T at power levels of 10 to 45 kw/ft.

The program of active tasks is organized in the following manner:

Task 1200 – High-Density Pellet Evaluation in the EBR-II (10 Rods)

Task 1300 – Annular and Low-Density Pellet Evaluation in the EBR-II
(19 Rods)

Task 1500 – Unrestrained Swelling and Fission Gas Release

Task 1800 – Sodium-Bonded Fuel Evaluation (4 Rods)

Task 1900 – Fabrication Development

1950 – Low-Power Subassembly in the EBR-II (19 Rods)

1960 – Intermediate Power Subassembly in the EBR-II (19 Rods)

Task 2400 – Mixed-Oxide Fabrication

Task 4800 – Evaluation of Thermal Behavior of Uranium-Plutonium Carbide Fuels

Task 4900 – TREAT Evaluation of Mixed Carbides

Task 5100 – High-Power Subassembly in the EBR-II.

Progress during the period April 1, 1970 to June 30, 1970 on the currently active tasks of this program is described in this report.

This is the twelfth in a series of quarterly progress reports written in partial fulfillment of Contract AT(30-1)-3415 between the United States Atomic Energy Commission and the United Nuclear Corporation.

Prior reports include the following:

Topicals

- UNC-5137 Cihi, J., Hurwitz, S., Rosh, D., and Strasser, A.: The Design and Fabrication of Three (UPu)C Pilot Experiments for EBR-II (Oct. 25, 1965).
- UNC-5170 Andersen, J., Cherry, B., Cihi, J., Powers, R., and Stahl, D.: EBR-II Design and Fabrication Reports of Rods 81 through 90 (Mar. 1967).
- UNC-5198 Stahl, D. and Strasser, A.: Post-Irradiation Examination of High-Density (UPu)C Pellet-Fueled EBR-II Rods Irradiated to 30,000 Mwd/T (Jan. 1968).
- UNC-5229 Powers, R. and Strasser, A.: Compatibility of Several Cladding Materials with Uranium-Plutonium Carbides in Sodium (Nov. 1968).

Progress Reports

Fast Reactor Mixed-Carbide Fuel Element Development Program

- UNC-5193 First Quarterly Progress Report, July through September 1967 (Nov. 15, 1967).
- UNC-5201 Second Quarterly Progress Report, October through December 1967 (Feb. 10, 1968).
- UNC-5217 Third Quarterly Progress Report, January through March 1968 (July 10, 1968).
- UNC-5224 Fourth Quarterly Progress Report, April through June 1968 (Sept. 26, 1968).
- UNC-5233 Fifth Quarterly Progress Report, July through September 1968 (Dec. 12, 1968).

- UNC-5238 Sixth Quarterly Progress Report, October through December 1968 (Apr. 1, 1969).
- UNC-5244 Seventh Quarterly Progress Report, January through March 1969 (July 7, 1969).
- UNC-5248 Eighth Quarterly Progress Report, April through June 1969 (Sept. 16, 1969).
- UNC-5251 Ninth Quarterly Progress Report, July through September 1969 (Dec. 1, 1969).
- UNC-5253 Tenth Quarterly Progress Report, October through December 1969 (Mar. 16, 1970).
- UNC-5259 Eleventh Quarterly Progress Report, January through March 1970 (May 1970).

2. HIGHLIGHTS

1. UNC 86 was irradiated to 73,000 Mwd/T in the EBR-II at 20 kw/ft without failure. The fuel rod is clad with Incoloy-800 and contains helium-bonded, high-density MC + 10 v/o M_2C_3 fuel pellets with an initial fuel-to-clad gap of 0.010 in. (Task 1200).
2. UNC 84 and 85, the same designs as UNC 86, but clad with Type 316 SS, both failed at a burnup of 73,000 Mwd/T at 20 kw/ft (Task 1200).
3. The four sodium-bonded fuel rods irradiated to 50,000 Mwd/T at 25 kw/ft were removed from the capsules and all are intact (Task 1800)
4. The fabrication of 2.2 kg 83% dense pellets and 1.5 kg 93% dense pellets for irradiation by ANL was completed successfully (Task 2400).

3. PAST RESULTS AND CURRENT PROGRESS

FAST REACTOR IRRADIATIONS

3.1 TASK 1200 – HIGH-DENSITY PELLET EVALUATION IN THE EBR-II

Introduction and Past Results

The objective of the task is to evaluate the performance of high-density (96% of theoretical, minimum) uranium-plutonium carbide pellets with varying size helium annuli between fuel and clad to provide void space for fuel swelling. The experiments are designed to determine the fuel-clad gap and clad strength combination to give satisfactory performance.

Ten doubly encapsulated fuel rods (Rods UNC 81 through 90) were designed, and nine were inserted for irradiation in EBR-II at powers of 20 to 30 kw/ft to burnups of 50,000 to 100,000 Mwd/T. The fuel is high-density, helium-bonded (UPu)C which was prepared by carbothermic reduction of the mixed dioxides followed by cold pressing and sintering of pellets. The cladding is Type 316 stainless steel and Incoloy-800. Eight of the capsules contain full-length prototypic fuel rods, while two capsules contain four segmented experiments each. A detailed description of these experiments is given in Table 1 and UNC-5170.

Nine of the 10 rods prepared under this task have completed irradiation in the EBR-II. The peak burnups and powers achieved were 45,400 Mwd/T at 20 kw/ft

TABLE 1 — SUMMARY OF CURRENT UNITED NUCLEAR MIXED-CARBIDE IRRADIATION EXPERIMENTS

Task No.	UNC Experiment No.	Description of Experiment	Design Parameters								Operating Conditions				Status
			Pellet Diameter, in.	Fuel Length, in.	Density, % T.D.		Pu, a/o	Cladding Material	Maximum Linear Power (Design), kw/ft	Target Burnup, Mwd/T $\times 10^{-3}$	Facility	Operating Maximum Linear Power kw/ft	Accumulated Burnup, Mwd/T		
					Pellet (Type)	Smeared									
1200	81-83	EBR-II rods, helium-bonded (8 rods)	0.248 to 0.252	14	98 (solid)	81 to 86	20	316 SS and Inc-800	20 to 30	50 to 100	EBR-II	19 to 31	45,500	Irradiation complete. UNC 81-87, 89, 90 post-irradiation examination in progress.	
	84-86					73,000									
	89-90	EBR-II multiple specimen rods, helium-bonded (2 rods)	0.249 to 0.252	4	98 (solid)	82 to 90	20	316 SS and Inc-800	24 to 30	40	EBR-II	22 to 28	32,000		
1300	91-114	EBR-II rods, helium-bonded (24 rods)	0.232 to 0.269	14	75 to 84 (solid), 99 (cored)	70 to 85	15	316 SS and Inc-800	20	50 to 100	EBR-II	17 to 23	53,000	UNC 101 irradiation complete.	
1500													74,000	18-rods continuing irradiation.	
	115-119	Unrestrained swelling specimens	0.06	0.5	75 to 90 (solid)	—	15	—	Low	20	Hanford	—	14,000	Irradiation complete.	
	120-124	Fission gas release specimens	0.375 OD, 0.255 ID	0.500	75 to 98 (annular)	—	15	—	Low	20	Hanford	—	20,000		
1800	125-128	Sodium-bonded rods (4 rods)	0.240	14	93 to 98 (solid)	83 to 87	15	304 SS and Inc-800	25	50	GETR	25	50,000	Irradiation complete.	
1950	129-147	19-rod low-power subassembly, helium and sodium bonds	0.240 to 0.246	14	75 to 98	70 to 87	15	316 SS, 304 SS, and Inc-800	15	100	EBR-II	15	34,000	Irradiation in progress.	
1930	185-190	Intermediate-power rods (6 rods)	0.249 to 0.253	14	84 to 96 (solid)	80 to 88	15	316 SS and Inc-800	30	50 to 100	EBR-II	—	—	UNC 185, 186 will not be irradiated. UNC 187-190 inserted with Task 1960.	
1960	191-209	19-rod intermediate-power subassembly, helium and sodium bonds	—	14	84 to 96	—	15	316 SS, 304 SS, and Inc-800	30	30 to 100	EBR-II	—	11,000	UNC 202-205 will not be irradiated; 15 rods + 4 Task 1930 rods inserted for irradiation.	
4800	210-221	“Rabbit” irradiations. Thermal behavior of (UPu)C fuels.	0.300	3.9	98	—	15	316 SS, 304 SS	45	—	GETR	45	—	Design and fabrication complete.	
4900	222-225	TREAT evaluation of mixed carbides	0.240 to 0.246	14	97 (solid)	—	15	316 SS, 304 SS	—	—	TREAT	—	—	Design in progress.	

for UNC 81 through 83, 73,000 Mwd/T at 20 kw/ft for UNC 84 through 86, and 32,100 Mwd/T at 30 kw/ft for UNC 87, 89, and 90. The irradiation conditions for these experiments are given in Table 2.

UNC 88 was never placed in the reactor because of the lack of suitable available space. A decision not to irradiate it was made recently, when post-irradiation examination revealed failures of fuel rods with a similar design.

Fuel rods UNC 87, 90, and three segments of UNC 89 were intact. One of the four segments of UNC 89, with a 3% hot void, had several brittle cracks in the Type 316 SS clad. Fuel rods UNC 81 through 83 all had brittle cracking of the Type 316 SS and Incoloy-800 clads.

The examination of the fuel microstructure of UNC 81 through 83 and 87, 89, 90 showed that it is similar to the previously examined material. The sesquicarbide phase is depleted from the high-burnup, high-temperature portion of the fuel and is distributed normally in the low-temperature, low-burnup regions. There is no central void, but porosity due to fission gas agglomeration is evident in the central region of the fuel.

Microprobe analysis of the Type 316 SS clad from UNC 89 irradiated to 40,000 Mwd/T showed that the metallographically observed dark etching zone at the clad ID contained no fuel constituents or fission products. No elements other than the constituents of stainless steel were found.

Microprobe analysis of the Incoloy-800 clad showed that some diffusion of the clad constituents into the fuel occurred at the fuel-clad interface.

Current Results

Fuel rods UNC 84, 85, and 86, containing high-density, helium-bonded MC + 10 v/o M₂C₃ pellets and clad with Type 316 SS or Incoloy-800, were removed

TABLE 2 — SUMMARY OF IRRADIATION CONDITIONS FOR TASK 1200 EXPERIMENTS

	UNC Rod Numbers															
								89				90				
	81	82	83	84	85	86	87	A	B	C	D	A	B	C	D	
Fuel Material	$(\text{UPu})\text{C} + 10\% (\text{UPu})_2\text{C}_3$															
Clad Material	316 SS	316 SS	Inc-800	316 SS	316 SS	Inc-800	Inc-800	316 SS	316 SS	316 SS	316 SS	Inc-800	Inc-800	Inc-800	Inc-800	
Fuel-to-Clad Gap, diametral, in. ‡	0.008	0.005	0.006	0.009	0.010	0.010	0.006	0.006	0.002	0.006	0.009	0.007	0.003	0.006	0.009	
Hot Void, % ‡	6.8	4.8	4.7	7.6	7.8	7.5	4.8	4.9	2.7	4.5	6.4	4.8	2.6	4.4	6.3	
Burnup, Mwd/T $\times 10^{-3}$	45.4*	42.8*	44.3*	73†	73†	73†	29.9*	32.0*	32.1*	31.1*	27.8*	31.3*	31.6*	30.7*	27.4*	
Max Fuel Temp, °C ‡	1640	1495	1520	1670	1690	1700	1820	1870	1905	1800	1855	1850	1900	1815	1840	
Max Clad OD Temp, °C	595	595	600	590	590	590	625	550	570	550	500	540	540	535	500	
Max Specimen Heat Generation Rate, w/cm	720	730	730	655	655	655	930	880	1010	930	730	910	1080	960	705	
Max Clad Surface Heat Flux, w/cm ²	295	300	305	219	219	219	400	360	415	380	300	390	160	410	300	
$\text{Max} \int_{T_s}^{T_c} K d\theta$, w/cm	57.3	58.1	58.1	52.1	52.1	52.1	76.4	70.0	80.4	74.0	58.1	72.5	86.0	76.4	56.1	
Peak Clad Fluence, nvt $\times 10^{-22}$	3.0	2.9	3.0	4.9	4.9	4.9	0.8	0.9	0.9	0.9	0.8	0.9	0.9	0.8	0.8	

*Burnup determined by Nd¹⁴⁸ analysis.

†Calculated burnup.

‡At startup.

from the capsules at Los Alamos Scientific Laboratory. The fuel rods were irradiated to a calculated peak burnup of 73,000 Mwd/T in the EBR-II at 20 kw/ft. The irradiation conditions for the fuel rods are given in Table 2.

Fuel rods UNC 84 and 85 were clad with Type 316 SS and had a cold fuel-to-clad diametral gap of 0.009 and 0.010 in., respectively. Both fuel rods failed with a large number of circumferential and longitudinal clad cracks. In comparison, UNC 86 clad with Incoloy-800 and a cold fuel-to-clad gap of 0.010 in. did not fail. Macro-photographs of the fuel rods are shown in Fig. 1.

The fuel structure of the three fuel rods is similar to that observed in the previous irradiation experiments. There is depletion of the sesquicarbide in the center of the fuel and a normal distribution toward the edge of the fuel (Fig. 2). The amount of porosity in the center of the fuel appears to be greater than at lower burnups. A platelet structure was observed in all the fuel, but is very similar to a platelet structure observed in the unirradiated archive samples (Fig. 3).

The dark etching zone at the Type 316 clad ID observed in the previously examined fuel rods at lower burnups is not nearly as pronounced in UNC 84 and 85. In many areas of the fuel clad interface this zone does not appear at all (Fig. 4). The zone at the ID of the Incoloy-800 clad of UNC 86 was similar in appearance to the zones noted at lower burnups (Fig. 5). Previous microprobe examination identified this as an area of interdiffusion of clad components containing no fuel or fission products.

A summary of post-irradiation examination for UNC 81 through 87, 89, and 90 is shown in Table 3.

UNC 84 – Type 316 SS

UNC 85 – Type 316 SS

UNC 86 – Incoloy-800

Fig. 1 — Fuel Rods UNC 84, 85, and 86
73,000 Mwd/T 20 kw/ft EBR-II
97% Dense Pellets MC + 10 v/o M_2C_3

500× Fuel Center Etchant — 1:1:1 H₂O, acetic, nitric acid

500× Fuel Edge Etchant — 1:1:1 H₂O, acetic, nitric acid

Fig. 2 — UNC 86 — Irradiated Fuel Microstructure
73,000 Mwd/T 20 kw/ft EBR-II
97% Dense Pellets MC + 10 v/o M₂C₃

Etchant:
1:1:1 H₂O,
Acetic,
Nitric Acid

500× Irradiated Fuel Microstructure
73,000 Mwd/T 20 kw/ft EBR-II
97% Dense Pellets MC + 10 v/o M₂C₃

Etchant:
1:1:1 H₂O,
Acetic,
Nitric Acid

500× Unirradiated Fuel Microstructure
97% Dense Pellets MC + 10 v/o M₂C₃

Fig. 3 — UNC 84 — Fuel Microstructure with Platelets

Dark etching
zone

75× Type 316 SS Dark Etching Zone As-polished

100× Type 316 SS As-polished

Fig. 4 — UNC 85 — Type 316 SS
 73,000 Mwd/T 20 kw/ft EBR-II
 97% Dense Pellets MC + 10 v/o M_2C_3

500× Fuel-Clad Interface As-polished

Fig. 5 — UNC 86 — Incoloy-800 Fuel Clad Interface
73,000 Mwd/T 20 kw/ft EBR-II
97% Dense Pellets MC + 10 v/o M_2C_3

TABLE 3 — SUMMARY OF POST-IRRADIATION EXAMINATION FOR UNC 81 THROUGH 87, 89, AND 90
IRRADIATED IN THE EBR-II

Fuel Rod No. UNC	Clad Material	Clad Integrity	Fission Gas Release, % of Theoretical	Max Clad OD Increase, in.	Max Restrained Fuel % $\Delta V/V$ per 10,000 Mwd/T		Fuel Density Decrease, %
					Based on ΔD_{clad}	Based on ΔV_{fuel}	
81	316 SS	Cracked	*	0.0053	2.3	1.9	8.7
82	316 SS	Cracked	*	0.0044	1.8	2.0	8.7
83	Inc-800	Intact	*	0.0104	2.8	2.2	8.4
84	316 SS	Cracked	*	0.0122	3.1	†	13.4
85	316 SS	Cracked	*	0.0074	3.0	†	14.7
86	Inc-800	Intact	‡	0.0033	2.3	†	10.2
87	Inc-800	Intact	1.30	0.0032	2.5	2.1	6.1
89A	316 SS	Intact	‡	0.0001	2.0	§	§
B	316 SS	Cracked	*	0.0035	2.0	§	§
C	316 SS	Intact	4.05	0.0018	2.5	3.2	9.5
D	316 SS	Intact	1.17	0.0004	2.3	§	§
90A	Inc-800	Intact	0.70	0.0000	2.5	2.3	6.9
B	Inc-800	Intact	0.23	0.0030	2.5	§	§
C	Inc-800	Intact	2.47	0.0029	2.3	2.8	8.3
D	Inc-800	Intact	‡	0.0012	2.3	§	§

*Sample lost due to clad failure.

†Analysis in progress.

‡Sample lost due to equipment failure.

§Fuel density not measured.

3.2 TASK 1300 – ANNULAR AND LOW-DENSITY PELLET EVALUATION IN THE EBR-II

Introduction and Past Results

The objective of the task is to evaluate various methods of void deployment in helium-bonded uranium-plutonium carbide pellets to provide void space for fuel swelling. Specifically, high-density annular (97% of theoretical) and low-density pellets (75 and 84% of theoretical) with various smear densities are being evaluated to determine the best fuel rod design.

The most commonly used solution to the fuel swelling problem is a controlled helium gap between the fuel and clad as in Task 1200. As an alternative approach, accommodation for swelling can be provided within the fuel. This can be in the form of a central void (cored pellets) or as uniformly distributed porosity.

Rods UNC 91 through 114 were designed for powers of 20 kw/ft to burnups of 50,000 to 100,000 Mwd/T. Design variables include power, burnup, clad material and thickness, fuel structure, and fuel smear density. Design details are shown in Table 1.

The 19-rod subassembly was discharged from the EBR-II with an average burnup of 53,000 Mwd/T. Ten fuel rods were to be removed, but approval was granted to irradiate nine of the 10 rods to 73,000 Mwd/T. The tenth rod, UNC 101, was removed for examination and shipped to LASL.

Neutron radiography of the 19 fuel rods showed no gross failures.

Post-irradiation examination of UNC 101 revealed a 3-in. crack ~3 in. above the bottom end plug. The Incoloy-800 clad fuel rod contained 84% of theoretical density pellets with a 0.001-in. fuel-to-clad gap and a hot void of 17%.

The fuel structure is the same as that observed in unirradiated 84% of the theoretical density fuel. The porosity does not appear to close and accommodate the fuel swelling.

Current Results

The subassembly X-075 has accumulated a burnup of 74,000 Mwd/T and is expected to be discharged early in the next quarter.

3.3 TASK 1500 – UNRESTRAINED SWELLING AND FISSION GAS RELEASE

Introduction and Past Results

The objective of the experiments is to investigate the unrestrained swelling and fission gas release characteristics of several types of (UPu)C fuel as a function of irradiation exposure and temperature. Since temperature measurement is essential to obtain meaningful data, the experiments have to be carried out in instrumented capsules; this is feasible in thermal reactors only. The problem of flux depression and resulting ΔT 's, which are not representative of fast reactors, is resolved by the use of small cross-section specimens operating at essentially isothermal conditions.

A description of the capsule trains was given in the third quarterly report, UNC-5217. The capsule trains were irradiated at design conditions. The experiments were operated in the Douglas United Nuclear Corporation special irradiation facilities with fuel temperatures between 1000 and 1600°C to 20,000 Mwd/T. A list of the experiments is given in Table 4.

Both capsule trains are at Argonne National Laboratory for post-irradiation examination.

Current Results

The post-irradiation examination of the experiments was initiated.

The fuel specimen assemblies UNC 120B, 121A, and 121B were removed intact, while the tantalum container of UNC 120A apparently oxidized and only fragments of the container remained. The contents have not been examined. Disassembly of the remaining capsules and fuel specimen assemblies is in process.

TABLE 4 — SUMMARY OF CAPSULE FUEL MATERIALS AND
OPERATING DATA

Unrestrained Swelling Capsule Train						
Capsule	Fuel*	% T.D.	Fuel Temperature, °C		Burnup, × 10 ⁻³ Mwd/T	
			Design	Operating		
UNC 116A†	MC + M ₂ C ₃ ‡	98	1300	1289	14	
116B	MC	84	1300	1340	14	
117A	MC	93	1300	1325	14	
117B	M(OC)	93	1300	1360	14	
118A	MO ₂	>90	1600	1366	14	
118B	MO ₂	>90	1600	1578	14	
119A	MC + M ₂ C ₃	98	1600	§	14	
119B	MC	84	1600	1304¶	14	
Fission Gas Release Capsule Train						
UNC 120A	MC + M ₂ C ₃	98	1000	1003	20	
120B	MC	84	1000	978	20	
121A	MC + M ₂ C ₃	98	1300	1275	20	
121B	MC	84	1300	1373	20	
122A	MC	93	1300	1308	20	
122B	MC	75	1300	1375	20	
123A	M ₂ C ₃	93	1300	1288	20	
123B	M(OC)	93	1300	1398	20	
124A	MC + M ₂ C ₃	98	1600	1298¶	20	
124B	MC	84	1600	1320¶	20	

*M = (U_{0.85}Pu_{0.15}), U = 93% U²³⁵.

†“A” specimen was the control specimen and operated at ±25°C of the design temperature.

‡90 v/o MC + 10 v/o M₂C₃.

§Thermocouple failed. Control was on “B” specimen.

¶Operating temperature was below design due to control rod shadowing.

3.4 TASK 1800 – SODIUM-BONDED FUEL EVALUATION

Introduction and Past Results

The objective of the irradiations is to evaluate the performance of sodium-bonded, uranium-plutonium carbide rods. The test parameters for the capsules are given in Table 1.

Capsules UNC 125, 126, 127, and 128 had been discharged from the GETR after achieving a peak goal burnup of 50,000 Mwd/T. All experiments operated at design power.

Neutron radiography of the four capsules indicated that approximately two-thirds of the original 0.030-in. diametral gap was utilized to accommodate fuel swelling.

Current Results

Post-irradiation examination of the capsules was initiated at Los Alamos Scientific Laboratory.

Examination of the betatron radiographs of the capsules has shown that the fuel is generally cracked and further confirms the previous measurements that approximately one-third of the original 0.030-in. diametral fuel-to-clad gap remains.

The fuel rods have been removed from the capsules and all the fuel rods are intact.

3.5 TASK 1950 – LOW-POWER SUBASSEMBLY IN THE EBR-II (19 RODS)

Introduction and Past Results

The objective of the experiment is to obtain performance data on uranium-plutonium carbide fuel rods at low powers. Fuel swelling data are of particular interest, because a large portion of an LMFBR will be operating at less than the 20 to 30 kw/ft peak power range at which the carbides are expected to operate. The test variables are given in Table 1.

Fuel rods and capsules for the subassembly were completed successfully and inserted in the EBR-II. This is the second all-carbide subassembly to be irradiated.

Current Results

The subassembly X-055, composed of UNC 129 through 147, operating in the EBR-II at 10 to 15 kw/ft, has accumulated an exposure of 34,000 Mwd/T.

3.6 TASK 1960 – INTERMEDIATE-POWER SUBASSEMBLY IN THE EBR-II (19 RODS)

Introduction and Past Results

The objective of the 19-rod experiments is to extend the current irradiation study to higher powers (30 to 35 kw/ft).

High linear and specific powers at high burnup are necessary in fast reactor fuels to reduce fuel cycle costs significantly. The most critical problem of high-power, high-burnup fuels is the accommodation of fuel swelling without rupturing the clad. One approach to this problem is the use of a sodium fuel-to-clad bond. The sodium provides expansion space, eliminates clad strain, and reduces fuel temperatures. Additional approaches utilizing a helium fuel-to-clad bond are finely dispersed voids in low-density pellets and high-density annular pellets.

Re-evaluation of the power output for the subassembly, based on the latest fission rates, indicated that it will operate between 26 and 29 kw/ft rather than 30 to 35 kw/ft. The reduced power will be caused by an enlarged EBR-II core which has depressed the flux near the core center.

The capsules have been fabricated, shipped, and accepted for irradiation by the EBR-II project.

Based on the results of the Task 1200 fuel rods, four high-probability-of-failure fuel rods, UNC 202 through 205, will not be irradiated but are replaced in the subassembly by UNC 187 through 190 from Task 1930.

Current Results

The subassembly X-079 has accumulated a burnup of 11,000 Mwd/T in the EBR-II.

3.7 TASK 2400 – MIXED-OXIDE PELLET FABRICATION

Past and Current Results

The objective of the task was to fabricate UO_2 -25 w/o PuO_2 pellets for the Argonne National Laboratory EBR-II Irradiation Program.

Approximately 2.2 kg of 83% and 1.5 kg of 93% of theoretical density pellets were prepared and shipped. All fuel met specifications as shown in Tables 5 and 6.

The fuel pellets were prepared by the conventional cold pressing and sintering method. Fully enriched UO_2 and PuO_2 powders were pre-blended in the twin shell blender with water and binder to provide 1/4 w/o polyvinyl alcohol. The agglomerates were vacuum dried. The dried and sized agglomerates were lubricated with 0.3 w/o dry zinc stearate powder in the twin shell blender to form uniform lots of free-flowing press feed. The pellets were cold pressed in an automatic hydraulic press, and sintered in the temperature range 1560 to 1725°C in a He-6% H_2 atmosphere. The sintered pellets were dry centerless ground to meet the specified diametral tolerance.

TABLE 5 — CHARACTERIZATION OF $\text{UO}_2\text{-PuO}_2$ FOR THE ANL IRRADIATION PROGRAM

	Batch No.				
	OX-36	OX-36	OX-37	OX-38	
Production Run No.	1	2	1	1	
Pellet Quantity, g	670	849	1451	774	
<u>Chemical Analysis</u>					<u>Specification</u>
O/M	1.96	1.95	1.96	1.96	1.97 ± 0.02
Pu, w/o	21.99	21.88	22.00	21.88	None
U, w/o	66.12	66.18	66.08	65.96	None
Pu/(U +Pu)×100, w/o	24.96	24.85	24.98	24.91	25.00 ± 0.5
Carbon, ppm	43	40	15	26	<200
Fluorine, ppm	11	3.4	9.5	8	<50*
Chlorine, ppm	<1	6.5	<1	5	<200
Nitrogen, ppm	31	20	12	28	<200
<u>X-Ray Analysis</u>					
Second phase, w/o	<15	<15	<15	<15	<15
<u>Alpha Autoradiography</u>					
Pu-containing particle size:					
Maximum, μ	43	46	65	56	<100
Average, μ	35	38	40	38	None
<u>Physical Analysis</u>					
Density, g/cm ³	{ —	—	10.43 ±0.02	—	10.36 ±0.17
	9.25 ±0.06	9.20 ±0.06	—	9.22 ±0.07	9.21 ±0.17
Diameter, in.	All pellets met specification				0.214 ± 0.001

*Sum of F + Cl.

TABLE 6 — SPECTROCHEMICAL ANALYSIS OF UO_2 - PuO_2 PELLETS FOR
THE ANL IRRADIATION PROGRAM

		Batch No.			
		<u>OX-36</u>	<u>OX-36</u>	<u>OX-37</u>	<u>OX-38</u>
Production Run No.:		1	2	1	1
<u>Element</u>	<u>Specification</u>	<u>ppm</u>	<u>ppm</u>	<u>ppm</u>	<u>ppm</u>
Al	500	41	53	19	13
B	20	< 0.5	< 0.5	< 0.5	< 0.5
Ca	250	15	15	13	10
Cd	20	< 0.5	< 0.5	< 0.5	< 0.5
Cr	250	115	104	87	51
Fe	500	175	190	127	217
Mg	50	22	25	19	25
Na	500	4.0	4.7	3.0	3.3
Ni	500	91	100	47	61
V	500	< 5	< 5	< 5	< 5
Total of Cu, Zn, Si, Ti	800	132	133	93	79
Total of Ag, Mn, Pb, Mo, Sn	200	< 10	< 11	< 11	< 17

3.8 TASK 4800 – EVALUATION OF THERMAL BEHAVIOR OF URANIUM-PLUTONIUM CARBIDE FUELS

Introduction and Past Results

The objective of this task is to study the short-time behavior of (UPu)C fuel, with both helium and sodium thermal bonds at very high heat fluxes and linear power, to determine whether any heat transfer limitations exist with typical fuel rod designs. Both good and defective fuel-to-clad bonds will be tested.

Twelve experiments with both helium and good and defective sodium fuel-to-clad bonds have been designed to operate at 45 kw/ft with a surface heat flux of 660 w/cm².

Post-irradiation examination will include neutron radiography and capsule disassembly to examine the thermal bonds and general condition of the fuel and clad.

The metal component fabrication and the fuel fabrication were completed.

Current Results

The 12 experiments were assembled and sent to GETR for irradiation. The irradiation, however, was delayed from June 1970 until July 1970 because of additional analyses required by GE for the hazards summary report.

3.9 TASK 4900 – TREAT EVALUATION OF MIXED CARBIDES

Introduction and Past Results

The objective of the task is to evaluate mixed carbides under transient conditions.

In order to define test parameters for mixed-carbide fuels, a review of transient testing on LMFBR-type fuels was made.

Two fuel rods will be removed from the Task 1950 subassembly for the irradiated TREAT experiments. One is a sodium-bonded, 304 SS clad fuel rod with high-density, two-phase fuel with a hot void of 22%. The second is a helium-bonded 316 SS clad fuel rod with high-density, two-phase fuel with a hot void of 8%. Two identical unirradiated fuel rods will be tested for comparison.

Current Results

A parametric study was completed to define a series of transients for the experiments. The transients selected are given in Table 7. These transients yield incipient fuel melting for the sodium-bonded fuel rod and 50% fuel melting in the helium-bonded fuel rod.

The fuel rods for the unirradiated transient experiments and the replacement rods for the Task 1950 subassembly have been fabricated. Characterization of the fuel is given in Table 8 and shows that the fuel met UNC specification.

TABLE 7 — TRANSIENTS FOR TREAT EVALUATION OF URANIUM-
PLUTONIUM CARBIDE

<u>Fuel Rod Type</u>	<u>Transient Type</u>	<u>Excess Reactivity Input, %</u>	<u>Time, sec</u>	<u>Fuel Melt, %</u>
Sodium bonded	Temperature limited	1.1	3.0	Incipient
Helium bonded	Temperature limited	0.94	3.5	50

TABLE 8 — CHARACTERIZATION OF (UPu)C * PELLETS FOR TRANSIENT
EXPERIMENTS AND REPLACEMENT RODS

Physical Characterization					
UNC Fuel Rods	Intended		Sintered		
			Density		
	Diameter, in.	Density, % T.D.	Diameter, in.	g/cm ³	% T.D.
146 } 146A }	0.240 ± 0.001	97 ± 2	0.2396 ± 0.0002	13.17 ± 0.02	98.3
138 } 138A }	0.246 ± 0.001	97 ± 2	0.2456 ± 0.0002	13.14 ± 0.04	98.1
Chemical Characterization					
UNC Fuel Rods	C, w/o	Pu, w/o	O ₂ , ppm	N ₂ , ppm	
146 } 146A }	4.71 ± 0.07	13.96 ± 0.06	3370 ± 310	145 ± 9	
138 } 138A }	4.74 ± 0.05	14.01 ± 0.06	3580 ± 900	140 ± 5	
Metallographic and X-Ray Characterization					
UNC Fuel Rods	Metallography †				
	% MC	% M ₂ C ₃	Lattice Parameters, Å		
146 } 146A }	90	10	MC: 4.9662 ± 0.0004 M ₂ C ₃ : Not determined		
138 } 138A }	89	11	MC: 4.9652 ± 0.0001 M ₂ C ₃ : Not determined		

*M = U_{0.85}Pu_{0.15}.

†Metallographic examination of the fuel pellets revealed a 10-micron layer of M₂C₃ on the pellet surface. The layer was removed by surface grinding.

3.10 TASK 5100 – HIGH-POWER SUBASSEMBLY IN THE EBR-II

Introduction and Past Results

The task objective is to conduct irradiation experiments at linear powers of up to 45 kw/ft (1500 w/cm) in the EBR-II.

A feasibility study has shown that a linear power of 45 kw/ft with a 0.030-in. diametral sodium annulus can be achieved in either an E-19 or B-19 singly clad type subassembly with a clad OD of 0.022 in. greater than the reference capsule. However, prior to 45 kw/ft operation, EBR-II prefers to operate the first singly clad carbide fuel rod at a previously demonstrated power, such as 30 kw/ft.

A peak power of 43 kw/ft could be achieved by a doubly clad experiment in Row 3.

Current Results

The design for a singly clad subassembly to operate between 30 and 35 kw/ft was started. The variables for the subassembly are shown in Table 9.

Subassembly hardware was designed by ANL based on a clad OD of 0.310 in. Analysis of the preliminary design showed that adequate coolant flow rates can be achieved for a nominal fuel rod power of 33 kw/ft.

Based on the successful irradiation of UNC 86, a helium-bonded, Incoloy-800 clad rod with a 10-mil diametral gap and several high-power, helium-bonded rods were considered with 12 to 15-mil gaps. All would have operated with center melting, a condition not considered desirable for singly clad rods at this time. The designs will be considered for doubly clad experiments on Task 1960.

TABLE 9 — DESIGN VARIABLES FOR SINGLY CLAD SUBASSEMBLY

UNC Rod No.	Fuel*	% T.D.	Fuel-Clad Bond	Clad	Burnup, Mwd/T $\times 10^{-3}$
222	MC	93	Sodium	304 SS	50
223		93		304 SS	100
224		93		316 SS	50
225		93		316 SS	100
226		93		316 SS	100
227		93		Inc-800	50
228		93		Inc-800	50
229		93		Inc-800	100
230		93		Inc-800	100
231	MC + 10 v/o M ₂ C ₃	97		304 SS	50
232		97		304 SS	100
233		97		316 SS	50
234		97		316 SS	50
235		97		316 SS	100
236		97		316 SS	100
237		97		Inc-800	50
238		97		Inc-800	50
239		97		Inc-800	100
240		97		Inc-800	100

*M = (U_{0.85} + Pu_{0.15})

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