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AN AEC RESEARCH AND
DEVELOPMENT REPORT

POST-IRRADIATION EXAMINATION OF
HIGH-DENSITY (UPu)C PELLET-FUELED
EBR-II RODS IRRADIATED TO 30,000 Mwd/T

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1. INTRODUCTION

United Nuclear Corporation is developing solid solution uranium-plutonium monocarbide, (UPu)C, as a competitive LMFBR fuel. The major objective of the program is to develop mixed-carbide LMFBR fuel rods that will perform satisfactorily up to 100,000 Mwd/T at power levels of 10 to 40 kw/ft. As the first part of the fast reactor irradiation program, three (UPu)C-fueled rods were irradiated in the EBR-II to a burnup of 10.6×10^{20} fission/cm³ (30,000 Mwd/T).

The objectives of the pilot irradiation were:

1. To obtain fast flux irradiation results on three full-length, helium-bonded, high-density pellet carbide rods in a practical length of time, to the minimum significant burnup, and at a significant power.
 - a. Full-length rod irradiations (as opposed to short rods) give information on the stability of a long column of fuel and the interaction between fuel movement and cladding restraint.
 - b. The fast flux provides the prototypical environment for the fuel rod. This is significantly different from previous thermal flux irradiations.
2. To obtain irradiation experiment design and handling experience in EBR-II.

The design and fabrication of these rods was reported in UNC-5137.¹ This report covers the irradiation at EBR-II, operated by Argonne National Laboratory, and the post-irradiation examination at Battelle Memorial Institute, Columbus, Ohio.

2. SUMMARY AND CONCLUSIONS

The three helium-bonded pilot rods fueled with high-density, hyperstoichiometric structure ($U_{0.8}Pu_{0.2}$)C pellets performed as designed during irradiation at the EBR-II. The rods operated up to a power of 28 kw/ft to a peak burnup of about 10.6×10^{20} fission/cm³ (30,000 Mwd/T) based on Nd^{148} and U-Pu isotopic analyses. This is the highest power at which mixed carbides have been irradiated in a fast reactor.

The fuel and clad performance was excellent. Type 316 clad OD increase was zero within the accuracy of the measurement and a maximum possible of 0.11% per 10,000 Mwd/T. The vanadium-type clad OD increases were a maximum of 0.25% per 10,000 Mwd/T. Fuel structure was essentially unchanged except for fission gas porosity at the grain boundaries in the central region of the highest power pellets. There was no detectable plutonium or fission product migration by alpha or beta-gamma autoradiography. The average fuel density changes, as measured by immersion, were in the range of 1.4 to 1.8% per 10,000 Mwd/T, similar to previous thermal irradiations. Fission gas release was a maximum of 0.5% of theoretical.

There were no reactions between the fuel and the type 316 stainless steel or vanadium clad and a slight reaction with the vanadium-20 w/o titanium. Clad sections from all rods were found to be ductile when bent at room temperature.

The fuel rods performed as predicted in these experiments. The good correlation between predicted and actual clad strain is particularly gratifying. If the fuel continues to behave in this manner at higher burnups, helium-bonded, high-density pellets will be able to achieve 100,000 Mwd/T at 20 to 30 kw/ft.

3. EXPERIMENT DESCRIPTION AND OPERATION

The fuel rods are tubes with welded end plugs, ~23-in. long, containing ~14 in. of $(U_{0.8}Pu_{0.2})C$ pellets. The pellets were fabricated by cold pressing and sintering (UPu)C powder made by carbothermic reduction of the oxides. High densities, 96% of theoretical, were achieved by the use of 0.1 w/o nickel sintering aid. Pellet characterization is given in Table 1, and typical pellets are shown in Fig. 1. The rods were designated as UNC 78, 79, and 80, and were clad with vanadium-20 w/o titanium, type 316 stainless steel, and vanadium, respectively. The design and fabrication of the rods were detailed in a previous topical report.¹ The vanadium and V-20Ti clad rods were designed for operation at a peak power of 985 w/cm while the stainless steel clad rod was designed for a peak power of 655 w/cm. Fuel and rod dimensions are given in Table 2.

The rods were encapsulated in a standard ANL, type 304 stainless steel capsule with a thermal bond of sodium in the annulus between the fuel rod and the capsule. A typical cross-sectional view of a loaded capsule is shown in Fig. 2. After nondestructive tests including radiography for pellet and rod location and eddy current examination for rod-to-capsule sodium bond integrity, the capsules were sent to Argonne National Laboratory, Chicago, for assembly into a 19-rod EBR-II subassembly. At ANL, the rods were placed in Assembly X009 and irradiated for 5355 reactor Mwd or 2850 hr in Row 4, Position 4A2, of the EBR-II. Burnup, power, and temperatures calculated from reactor data are given in Table 2 (and Table 9 of Section 4.2.5), and Figs. 3, 4, and 5.

TABLE 1 — CHARACTERIZATION OF (U_{0.8}Pu_{0.2})C PELLETS

Rod	Diameter, in.	Sintered Density		C, w/o	Pu, w/o	N, ppm	O, ppm	Ni, ppm	Equivalent C, w/o	X-Ray Analysis (Diffractometer)
		By Dimensions and Weight, g/cm ³	By CCl ₄ Immersion, g/cm ³							
78	0.2512 ± 0.0005	13.23 ± 0.03	13.31 ± 0.02	4.81	18.9	300	3800	1400	4.92	Major (UPu)C type a ₀ = 4.9676 ± 0.0011 Å Faint/weak (UPu) ₂ C ₃ type Faint indication of (UPu)O ₂ type
79	0.2520 ± 0.0008	13.25 ± 0.05	13.52 ± 0.02	4.73	19.1	1155	4600	1400	5.18	Major (UPu)C type a ₀ = 4.9660 ± 0.0005 Å Weak (UPu) ₂ C ₃ type
80	0.2509 ± 0.0008	13.18 ± 0.10	13.01 ± 0.02	4.78	19.2	300	3900	1400	5.10	Major (UPu)C type a ₀ = 4.967 ± 0.0004 Å Faint (UPu) ₂ C ₃ type

TABLE 2 — SUMMARY OF IRRADIATION CONDITIONS

	<u>UNC 78</u>	<u>UNC 79</u>	<u>UNC 80</u>
Clad	V-20 Ti	Type 316	Vanadium
Fuel	(U _{0.8} Pu _{0.2})C	(U _{0.8} Pu _{0.2})C	(U _{0.8} Pu _{0.2})C
Smear Density, % T.D.	94	93	93
Fuel OD, in.	0.251	0.252	0.251
Fuel-Clad Gap Cold at Startup, diametral in.	0.005	0.007	0.006
Clad Thickness, in.	0.021	0.023	0.022
Calculated Maximum Fuel Temperature at Startup, °C	1505	1465	1595
Calculated Clad Surface Temperature, °C	630	600	630
Design Maximum Clad Strain, %	0.4	0.0	0.4
Maximum Linear Power, w/cm	920	625	920
Maximum Clad Surface Heat Flux, w/cm ²	385	255	385
Maximum Fuel Power Density:			
kw/cm ³	2.90	1.95	2.90
w/g (U + Pu)	230	155	230
Max $\int_{T_s}^{T_c} k d\theta$, w/cm	73.0	49.5	73.0
Time at Power, hr	2850	2850	2850
Clad Exposure, nvt	1.7×10^{22}	1.7×10^{22}	1.7×10^{22}
Burnup Peak:			
fiss/cm ³	10.2×10^{20}	7.4×10^{20}	10.6×10^{20}
(Mwd/T)	(29,000)	(21,000)	(30,000)

Some of the pellets used for UNC 79

Fig. 1 — Typical $(U_{0.8}Pu_{0.2})C$ Fuel Pellets

Fig. 2 — Cross-Sectional View of Typical EBR-II Irradiation Capsule Assembly

Fig. 3 — UNC 78 Performance Summary

Fig. 4 — UNC 79 Performance Summary

Fig. 5 — UNC 80 Performance Summary

The relative positions of the UNC rods in the subassembly are shown in Fig. 6. Vibratory compacted and low-density pellet fuel was irradiated by ANL in positions 8, 12, and 16 of this subassembly in cladding from the same lot of tubing as the UNC rods. The results will offer a good performance comparison between the different types of fuel.

<u>Position</u>	<u>Capsule Type</u>	<u>Ident. No.</u>	<u>Comments</u>
1	ANL	AS-27	Structural material
2	GE	L 4 C	Structural material
3	ANL	SP-13	Cermet fuel
4	BNWL	A1	Structural material
5	ANL	SOV-6	Oxide fuel
6	UNC	UNC-80	Carbide fuel
7	UNC	UNC-79	Carbide fuel
8	ANL	SMV-1	Carbide fuel
9	BNWL	A5	Structural material
10	ANL	AS-14	Structural material
11	ANL	5P-14	Cermet fuel
12	ANL	VMV-1	Carbide fuel
13	ANL	SOV-5	Oxide fuel
14	BNWL	A2	Structural material
15	ANL	AS-15	Structural material
16	ANL	SMP-1	Carbide fuel
17	GE	L4D	Structural material
18	BNWL	A6	Structural material
19	UNC	UNC-78	Carbide fuel

As Loaded

Subassembly No.	X009
Internal orifice diameter	0.750 in.
For core location	4A2

View looking down
on T-bar grid

Fig. 6 — Positions of UNC Rods in EBR-II Subassembly

4. POST-IRRADIATION EXAMINATION

The fuel rods were shipped from EBR-II after about three months of cooling time to Battelle Memorial Institute, Columbus, Ohio, for post-irradiation examination. The examination included the determination of:

- Fuel and clad dimensional stability
- Fuel location by gamma scanning
- Fission gas release
- Metallography of fuel and clad structure
- Burnup determination
- Plutonium and fission product distribution
- Fuel oxygen content
- Clad hardness and ductility.

The results of the examination are given in detail in the following sections and are summarized in Table 3.

Additional examinations will be carried out by other AEC-sponsored programs on samples supplied by United Nuclear from these fuel rods:

Hot burst tests on the clad gas plenum section	BMI, Columbus
Electron microscopy and gas analysis of the clad	ORNL
Electron microscopy and microprobe analysis of the fuel	ANL
Reprocessing studies on the fuel	ORNL.

TABLE 3 — SUMMARY OF POST-IRRADIATION EXAMINATION RESULTS

	<u>UNC 78</u>	<u>UNC 79</u>	<u>UNC 80</u>
Clad Material	V-20 Ti	Type 316	Vanadium
Clad Appearance	Excellent	Excellent	Excellent
Clad Integrity	Intact except one thin crack near weld	Intact	Intact
Fission Gas Release, % of theoretical	0.20	0.36	0.50
Condition of Pellets	Most pellets recovered whole, some sodium oxide on surface	Surface excel- lent, all pel- lets cracked	Surface excel- lent, all pel- lets cracked
Clad OD Increase:			
Average, %	0.4	0.1	0.4
%/10,000 Mwd/T	0.15	0.07	0.15
Pellet OD Increase:			
Average, %	2.5	—	—
%/10,000 Mwd/T	0.9	—	—
Fuel Density Decrease by Immersion:			
Average, %	4.1	3.3	4.0
%/10,000 Mwd/T	1.6	1.8	1.4
Fuel Metallography	As before, but with fission gas bubbles at pellet centers		
Clad Metallography	Reaction op- posite fueled region	No fuel-clad reaction	No fuel-clad reaction
Clad Bend Test	Ductile	Ductile	Ductile
Plutonium and Fission Product Distribution	Uniform	Uniform	Uniform

4.1 FUEL ROD PERFORMANCE

4.1.1 Visual Examination

The fuel rods were removed from the capsules and examined. The surfaces of all the rods were excellent. The rods were intact except for a thin crack near the weld of the bottom end plug in UNC 78, the V-20 Ti clad rod (Fig. 7). The crack was sufficiently small to prevent significant fission gas leakage to the capsule. All the rods were rolled on a granite block to obtain a rough measurement of bowing which could possibly be due to the thermal gradient. UNC 78 was fairly straight while UNC 79 and UNC 80 were slightly bowed. With handling, the bow of the stainless clad rod, UNC 79, increased to a value of about 0.015 to 0.030 in.

4.1.2 Clad Measurement

The outside diameters of the rods were measured every inch in two planes, the second 90 degrees from the first. The rods were placed in a V-shaped tray and readings were taken with a micrometer. The diameter values obtained are given in Tables 4, 5, and 6 together with the calculated percent change in diameter. The average percent changes or strains are plotted with the calculated design strains in Fig. 8. The experimental values are in agreement with the design values of clad strain within the uncertainty of the measurements.

The percent change in diameter was plotted as a function of length in the fuel region and compared to the burnup-power profile in Figs. 3, 4, and 5. The shape of the curves for UNC 78 and 80 (V clad, 28 kw/ft) follows the shape of the burnup, power, and gamma scan curves indicating greater clad strain at the higher performance levels. The maximum clad strain was 0.8% or 0.25%/10,000 Mwd/T.

The data for UNC 79 (type 316, 19 kw/ft) give a flat curve indicating nearly zero change within the accuracy of the measurements. The maximum possible clad strain was 0.2% or 0.11%/10,000 Mwd/T.

TABLE 4 — POST-IRRADIATION CLAD DIAMETER MEASUREMENTS OF
UNC 78* (V-20 Ti CLAD)
(Pre-Irradiation Diameter — 0.2990 in.)

	Axial Position †	Diameter at Punch Mark, in.	Diameter 90° from Mark, in.	Average Diameter, in.	Change (Strain), in.	Change (Strain), %
Gas Plenum	0	0.2984	0.2986	0.2985	0	0
	1	0.2987	0.2985	0.2986	0	0
	2	0.2986	0.2986	0.2986	0	0
	3	0.2988	0.2989	0.2989	0	0
	4	0.2984	0.2987	0.2986	0	0
	5	0.2985	0.2987	0.2986	0	0
	6	0.2985	0.2988	0.2987	0	0
	7	0.2987	0.2990	0.2989	0	0
	8	0.2988	0.2994	0.2991	0	0
Fueled Portion	9	0.2994	0.2998	0.2996	0.0006	0.2
	10	0.2995	0.2998	0.2997	0.0007	0.2
	11	0.2997	0.3000	0.2999	0.0009	0.3
	12	0.3000	0.3002	0.3001	0.0011	0.4
	13	0.3002	0.3004	0.3003	0.0013	0.4
	14	0.3008	0.3005	0.3007	0.0017	0.6
	15	0.3007	0.3006	0.3007	0.0017	0.6
	16	0.3005	0.3004	0.3005	0.0015	0.5
	17	0.3006	0.3002	0.3004	0.0014	0.5
	18	0.2998	0.2998	0.2998	0.0008	0.3
	19	0.2996	0.2997	0.2997	0.0007	0.2
	20	0.2993	0.2995	0.2994	0.0004	0.1

*Post-irradiation diameters are accurate to within ± 0.0005 in.

†Axial positions start at the top of the fuel rod and are approximately 1-in. apart.

TABLE 5 — POST-IRRADIATION CLAD DIAMETER MEASUREMENTS OF
UNC 79* (TYPE 316 CLAD)
(Pre-Irradiation Diameter — 0.3060 in.)

	Axial Position †	Diameter at Punch Mark, in.	Diameter 90° from Mark, in.	Average Diameter, in.	Change (Strain), in.	Change (Strain), %
Gas Plenum	0	0.3069	0.3063	0.3066	0.0006	0.2
	1	0.3060	0.3059	0.3060	0	0
	2	0.3061	0.3063	0.3062	0	0
	3	0.3060	0.3060	0.3060	0	0
	4	0.3060	0.3060	0.3060	0	0
	5	0.3062	0.3064	0.3063	0	0
	6	0.3059	0.3060	0.3060	0	0
	7	0.3061	0.3061	0.3061	0	0
Fueled Portion	8	0.3059	0.3062	0.3061	0	0
	9	0.3063	0.3059	0.3061	0	0
	10	0.3062	0.3059	0.3061	0	0
	11	0.3064	0.3064	0.3064	0.0004	0.1
	12	0.3066	0.3060	0.3063	0	0
	13	0.3063	0.3063	0.3063	0	0
	14	0.3065	0.3063	0.3064	0.0004	0.1
	15	0.3065	0.3061	0.3063	0	0
	16	0.3063	0.3061	0.3061	0	0
	17	0.3060	0.3062	0.3061	0	0
	18	0.3067	0.3064	0.3066	0.0006	0.2
	19	0.3069	0.3064	0.3067	0.0007	0.2
	20	0.3068	0.3066	0.3067	0.0007	0.2
	21	0.3065	0.3069	0.3067	0.0007	0.2
	22	0.3068	0.3064	0.3066	0.0006	0.2

*Post-irradiation diameters are accurate to within ± 0.0005 in.

†Axial positions start at the top of the fuel rod and are approximately 1-in. apart.

TABLE 6 — POST-IRRADIATION CLAD DIAMETER MEASUREMENTS OF
UNC 80* (V CLAD)
(Pre-Irradiation Diameter — 0.3007 in.)

	Axial Position †	Diameter at Punch Mark, in.	Diameter 90° from Mark, in.	Average Diameter, in.	Change (Strain), in.	Change (Strain), %
Gas Plenum	0	0.3003	0.3001	0.3002	0	0
	1	0.3004	0.3003	0.3004	0	0
	2	0.3005	0.3007	0.3006	0	0
	3	0.3007	0.3008	0.3008	0	0
	4	0.3007	0.3009	0.3008	0	0
	5	0.3009	0.3009	0.3009	0	0
	6	0.3010	0.3008	0.3009	0	0
	7	0.3009	0.3010	0.3010	0	0
Fueled Portion	8	0.3008	0.3006	0.3007	0	0
	9	0.3014	0.3008	0.3011	0.0004	0.1
	10	0.3011	0.3010	0.3011	0.0004	0.1
	11	0.3020	0.3013	0.3017	0.0010	0.3
	12	0.3022	0.3020	0.3021	0.0014	0.5
	13	0.3020	0.3020	0.3020	0.0013	0.4
	14	0.3027	0.3025	0.3026	0.0019	0.6
	15	0.3028	0.3024	0.3026	0.0019	0.6
	16	0.3032	0.3028	0.3030	0.0023	0.8
	17	0.3027	0.3025	0.3026	0.0019	0.6
	18	0.3021	0.3030	0.3026	0.0019	0.6
	19	0.3011	0.3020	0.3016	0.0009	0.3
	20	0.3008	0.3012	0.3010	0	0
	21	0.3009	0.3009	0.3009	0	0

*Post-irradiation diameters are accurate to within ± 0.0005 in.

†Axial positions start at the top of the fuel rod and are approximately 1-in. apart.

12×

Fig. 7 — Thin Crack near UNC 78 Bottom End
Plug Weld

Fig. 8 — Comparison of Calculated and Actual Clad Strain vs Burnup

The percent change per 10,000 Mwd/T for peak and average values of burnup is summarized in Table 7.

4.1.3 Gamma Scanning

Gamma scans of all three rods were similar in appearance. Gamma plots are shown in Figs. 3, 4, and 5. The gamma scans for the capsules were similar to the scans for the associated rods. Gamma activity increased from the top of the fuel column to the fuel midplane and then decreased slowly on approaching the bottom of the fuel column. The sharp localized dips in gamma activity are probably due to small pellet separations. These separations are probably caused by mechanical movement during transfer operations since no pellet-pellet interactions were found.

Changes in fuel column length were determined from the gamma scan since the scan is accurate to within ± 0.01 in. The results of these measurements, which are given below, show no appreciable length change.

<u>UNC Rod</u>	<u>Fuel Column Length, in.</u>		<u>Fuel Column Length Increase, %</u>
	<u>Pre-Irradiation</u>	<u>Post-Irradiation</u>	
78	14.090	14.088	0.0
79	14.130	14.163	+0.2
80	14.310	14.288	-0.2

4.1.4 Fission Gas Release

Fission gases were removed from all rods and analyzed by mass spectrometer. Based on average burnup values, the fission gas release was found to be as follows:

TABLE 7 — SUMMARY OF CLAD DIAMETER CHANGES

<u>Clad</u>	<u>Value</u>	Pre- Irradiation, <u>in.</u>	Post- Irradiation, <u>in.</u>	Change, <u>in.</u>	<u>% Change</u>	<u>% Change per 10,000 Mwd/T</u>
V-20 Ti	Average	0.2990	0.3001	0.0011	0.37	0.15
	Peak	0.2990	0.3007	0.0017	0.57	0.20
Type 316	Average	0.3060	0.3064	0.0004	0.13	0.07
	Peak	0.3060	0.3067	0.0007	0.23	0.11
Vanadium	Average	0.3007	0.3019	0.0012	0.40	0.15
	Peak	0.3007	0.3030	0.0023	0.76	0.25

<u>Fuel Rod</u>	<u>Fission Gas Release, %</u>
78	0.20
79	0.36
80	0.50

A small quantity of fission gas (<0.01% of that produced based on gamma spectrometric analysis for Kr⁸⁵, the minimum detectable by mass spectrometer) was found in the outer capsule shell of UNC 78. This gas probably leaked through the crack in the bottom end plug weld of the fuel rod. No fission gases were found in the capsule shells of UNC 79 and 80.

4.2 FUEL PERFORMANCE

4.2.1 Visual Examination

The clad was slit longitudinally in three planes approximately 120° apart, and the clad was peeled back as shown in Fig. 9. In all cases, the cuts extended into the fuel, as shown in Fig. 10, and some damage to the pellets resulted. Whole pellets were obtained only from UNC 78. A typical group is shown in Fig. 11.

The appearance of the fuel pieces from UNC 79 and 80 was excellent, as shown in Fig. 12. Pellets from these rods contained one or two longitudinal "S-shaped" cracks prior to slitting as evidenced by replicas on the internal clad surfaces. Similar cracks were noted in the pellets from UNC 78 but these pellets did not fall apart during slitting. The surfaces of pellets from UNC 78 were covered with sodium oxide from sodium which leaked through the crack in the bottom end plug weld.

4.2.2 Dimensional Changes

The physical dimensions and weights were determined for the 30 whole pellets taken from UNC 78. The post-irradiation weights were within 1% of the pre-

Fig. 9 — UNC 79 during Decladding (Note Type 316 Clad Ductility)

3×

Fig. 10 — UNC 80 Partially Declad [(UPu)C Pellets in Vanadium Clad]

Fig. 11 — Whole Pellets from UNC 78 (V-20 Ti Claã)

4x

Fig. 12 — Typical Broken Fuel Pellet from UNC 79
(Type 316 Clad)

irradiation weights, with most values (85%) lying within 0.5% of the pre-irradiation weights. Pre-irradiation weights are accurate to $\pm 0.4\%$ while post-irradiation weights are accurate to $\pm 0.1\%$, so that the changes were within experimental error. Diameter values varied randomly and bore no relation to flux profile. The diameter increases were greater than would be expected from clad measurements. This discrepancy is due to a widening of the cracks in the pellets either from relief of the clad restraint or by the reaction of sodium in the cracks with the atmosphere.

For these reasons, the densities measured by immersion technique are probably a better assessment of swelling than are the dimension and weight measurements. The values are given in Table 8 along with relative volume change with burnup, $\Delta V/V$ per 10,000 Mwd/T. The maximum specific volume change for fuel from both UNC 78 and 80, the higher power (28 kw/ft) rods, was 2.0% per 10,000 Mwd/T with an average of 1.6 and 1.4%, respectively. The maximum and average fuel volume changes for UNC 79, the lower power (19 kw/ft) rod, was 2.2 and 1.8% per 10,000 Mwd/T, respectively.

4.2.3 Structure

Fuel pellets or portions of pellets were taken from the high and low power positions of all rods, as shown in Figs. 3, 4, and 5, and mounted for metallographic examination. All specimens were examined in the unetched as well as etched condition.

The outer portions of the pellets were as before irradiation. Burnup did not increase the porosity. There was no central void formation; however, porosity due to fission gas agglomeration was found in the central portions of the pellets. The amount of porosity increased with temperature. Fig. 13 shows the typical distribution of voids at the edges and centers of pellets taken from various power positions in the rods. The pores tend to agglomerate at grain boundaries, as previously noted in the thermal irradiations.²

TABLE 8 — IMMERSION DENSITIES OF PELLETS

UNC Rod	Pellet No.	Density, g/cm ³		Density Change, %	% $\Delta V/V$ per 10,000 Mwd/T *
		Pre- Irradiation	Post- Irradiation		
78	7	13.10	12.73	-2.8	1.2
	19	13.26	12.58	-5.1	1.8
	33	13.26	12.49	-5.8	2.0
	45	13.22	12.69	-4.0	1.6
	53	13.20	12.80	-3.0	1.5
79	3	13.26	12.96	-2.3	1.6
	12	13.22	12.90	-2.4	1.4
	20	13.27	12.76	-3.8	1.9
	30	13.27	12.64	-4.8	2.2
	45	13.25	12.75	-3.8	2.2
	52	13.25	12.85	-3.0	2.2
80	6	13.05	12.69	-2.8	1.2
	25†	13.11	12.31	-6.1	2.0
	33	13.12	12.37	-5.7	1.9
	45	13.14	12.70	-3.4	1.3
	53	13.14	12.85	-2.2	1.0

*Corrected for burnup profile in the rod.

†Probable pellet number from pellet group.

Edge

Center

Pellet 7, UNC 78: 720 w/cm, 22,800 Mwd/T

Pellet 35, UNC 79: 610 w/cm, 20,600 Mwd/T

Pellet 27, UNC 80: 920 w/cm, 30,000 Mwd/T

Fig. 13 — Typical Void Distribution in Fuel Pellets
(Fuel Unetched, 500×)

The fuel specimens were etched with the standard nitric-acetic-water mixture. No grain growth was present in any of the samples. The structure in the cooler outer two-thirds of the fuel pellets was unchanged from the pre-irradiation (UPu)C + 15 v/o (UPu)₂C₃ structure. At the center of the pellets, which operated at 720 w/cm and below, the structure was unchanged. At 760 w/cm and above, the sesquicarbide phase disappeared. Typical structures are shown in Figs. 14, 15, and 16.

The mechanism for sesquicarbide disappearance is not known. In no case was there evidence of carbon migration to the outer pellet regions, fuel migration, or sesquicarbide decomposition to monocarbide plus dicarbide. The sesquicarbide disappearance is probably connected with a relatively small stoichiometry change induced by nickel or oxygen migration. It also may be connected with a higher fissioning rate in the sesquicarbide, based on microprobe investigator's reports of a higher plutonium concentration in the M₂C₃ than the MC phase.

Oxygen analysis by vacuum fusion on a high burnup pellet gave inconclusive results. The hot cell vacuum fusion rig gave a value of ~400 ppm for the irradiated pellet compared to 550 ppm for the unirradiated control sample. Since (UPu)C pellets with less than 2000 ppm oxygen had never been made at the time this lot of fuel was fabricated, the analyses are considered unreliable.

4.2.4 Plutonium and Fission Product Distribution

The specimens mounted for metallographic examination were used for the determination of plutonium and fission product distribution by alpha and beta autoradiography. Included were the samples from high and low power positions of all rods.

The intensity of all films was uniform. Therefore, no detectable segregation of plutonium or fission products occurred during the irradiation. Typical

Pellet Edge

500×

Pellet Center

500×

(UPu)C is gray matrix, $(UPu)_2C_3$ is white phase

Fig. 14 — Typical Fuel Structure (UNC 78, Pellet 7)
after Irradiation to: 720 w/cm (22 kw/ft)
170 w/g
22,800 Mwd/T

Pellet Edge

500×

Pellet Center

500×

(UPu)C is gray, $(UPu)_2C_3$ is white phase, voids are black

Fig. 15 — Typical Fuel Structure (UNC 80, Pellet 9)
after Irradiation to: 760 w/cm (23 kw/ft)
180 w/g
24,200 Mwd/T

Pellet Edge

500×

Pellet Center

500×

Pellet Center

1000×

(UPu)C is gray matrix, (UPu)₂C₃ is white phase,
voids are black

Fig. 16 — Typical Fuel Structure (UNC 80, Pellet 27)
after Irradiation to: 920 w/cm (28 kw/ft)
225 w/g
30,000 Mwd/T

results are shown in Fig. 17 along with the photomacrograph of the specimen. The lack of plutonium or fission product redistribution confirms previous thermal irradiation results.

4.2.5 Burnup Analysis

The (UPu)C burnup analyses were based on the comparison of Nd^{148} and the U and Pu isotopes to unirradiated controls. General Electric's Vallecitos Laboratories made the analyses on solutions of entire pellets prepared by Battelle. The results are given below.

Rod No.	Pellet No.	Atom % Fissioned		10^{20} fiss/cm ³ *	
		Nd^{148}	U,Pu Isotopic	Nd^{148}	U,Pu Isotopic
78	35	2.82	3.44	9.0	11.0
79	36	2.00	2.60	6.4	8.3
80	32	2.98	3.46	9.5	11.1

The axial variation of burnup was calculated for each rod, based on the flux profile reported in Reference 3. The accuracy of the Nd^{148} analysis is reported to be $\pm 10\%$ while that of the isotopic analysis is reported to be $\pm 15\%$. Using these error bands, the most probable burnup was calculated and plotted in Figs. 3, 4, and 5. The peak burnups are a factor of 1.15 higher than the average for each rod and both are given in Table 9. The peak burnup of the highest power rod was 10.6×10^{20} fiss/cm³ (30,000 Mwd/T) while that of the lower power rod was 7.4×10^{20} fiss/cm³ (22,000 Mwd/T).

4.2.6 Temperature Analysis

The temperature of fuel in helium-bonded rods with large fuel-clad gaps changes continuously. At startup the fuel operates at the highest temperature of its life cycle. As the fuel swells and fills up the helium gap, its temperature drops rapidly until it contacts the clad.

*cm³ of (UPu)C.

TABLE 9 — SUMMARY OF BURNUP ANALYSES

	<u>UNC 78</u>	<u>UNC 79</u>	<u>UNC 80</u>
<u>Average Fuel Burnup:</u>			
Reactor data			
fiss/cm ³	8.7×10^{20}	5.9×10^{20}	8.7×10^{20}
(Mwd/T)	(25,000)	(17,000)	(25,000)
Nd analysis			
fiss/cm ³	8.2×10^{20}	5.8×10^{20}	8.4×10^{20}
(Mwd/T)	(24,000)	(17,000)	(25,000)
U,Pu analysis			
fiss/cm ³	9.8×10^{20}	7.5×10^{20}	9.7×10^{20}
(Mwd/T)	(29,000)	(22,000)	(28,000)
Probable burnup			
fiss/cm ³	8.8×10^{20}	6.4×10^{20}	9.2×10^{20}
(Mwd/T)	(26,000)	(19,000)	(26,000)
<u>Peak Fuel Burnup:</u>			
Probable burnup			
fiss/cm ³	10.2×10^{20}	7.4×10^{20}	10.6×10^{20}
(Mwd/T)	(29,000)	(21,000)	(30,000)

Metallographic Section* (3×)

Alpha Autoradiograph

Beta-Gamma Autoradiograph

*Grooves in clad from post-irradiation slitting.

Fig. 17 — Typical Plutonium and Fission Product Distribution
(UNC 78, Pellet 2) after Irradiation to: 620 w/cm (19 kw/ft)
150 w/g
19,500 Mwd/T

After clad contact is made, temperature drop is only slight and interface pressure dependent.

Two things could happen to prevent the temperature from dropping: the gas gap conductivity could decrease by fission gas release, and the fuel conductivity could decrease with burnup. Fission gas release during the initial burnup period is low, less than 1/2%, and does not affect gas gap conductivity. The effect of burnup on fuel conductivity is not significant up to ~8000 Mwd/T and is likely to be insignificant above that, compared to the increase in the gas gap conductivity.

The fuel temperature history therefore was calculated assuming constant fuel conductivity and a gas gap containing helium only. A simplified assumption of a linear swelling rate of 1.8% $\Delta V/V$ was made.

The fuel temperature was calculated using an iterative technique which takes into account the temperature dependence of the helium gap conductivity and the thermal expansion of both fuel and clad.⁴ Pre-irradiation thermal expansion and conductivity were used since reliable post-irradiation values were not available.

The change in the fuel centerline temperature with burnup is shown in Fig. 18. The rate of temperature change for UNC 79 is less since the initial gap is larger. The change in temperature distribution with burnup was also calculated for each pellet used as a metallographic specimen and plotted as a function of fractional radius. Plots for UNC 80 given in Figs. 19 and 20 correspond to the pellets experiencing minimum and maximum temperatures of the rod, respectively. The solid lines in the figures are the beginning and end-of-life temperature distributions. The area in between these two curves represents the distribution at any intermediate time and can be imagined to be made up of a series of curves of the same shape.

Fig. 18 — Fuel Centerline Temperature vs Burnup (UNC 78, 79, 80)

Fig. 19 — Change in Radial Fuel Temperature Distribution with Time (UNC 80, Pellet 9, 760 w/cm)

Fig. 20 — Change in Helium-Bonded Carbide, Radial Temperature Distribution with Time (UNC 80, Pellet 27, 850 w/cm)

The curves show that although startup temperatures are high, they drop rapidly to equilibrium values of about 1000°C, independent of power.

4.3 CLAD PERFORMANCE

4.3.1 Structure and Compatibility

Metallographic examination of the clad was made on samples taken from the region opposite to the high burnup portion of the fuel rod and from the gas plenum. Longitudinal and radial sections were examined. Figs. 3, 4, and 5 show the sample locations.

The type 316 stainless steel from Rod 79 had typical stainless steel microstructure, as shown in Fig. 21. There was no evidence of fuel-clad reactions. The inside, and to a lesser extent the outside, surface of the sample taken opposite the fuel had more rapidly etching grain boundaries than the sample from the gas plenum. This is probably due to sensitization caused by operation at temperatures of 550° to 600°C for 2850 hr.

The vanadium samples did not show any evidence of fuel-clad reactions. The sample from the fueled and gas plenum regions were the same as the unirradiated control (Fig. 22).

The fueled region of the vanadium-20 w/o titanium clad, UNC 78, interacted with the fuel, while the plenum region was unaffected (Fig. 23). The interaction of 1 to 2 mils was uniform and did not extend along grain boundaries toward the outside surface. This interaction was unlike that found in out-of-pile tests which produced intergranular carbide precipitation. The in-pile reaction is probably related to the presence of sodium in the annulus. The fuel showed no depletion of sesquicarbide from the surface.

The vanadium and V-20 w/o Ti spacer discs from the ends of the fuel stacks

Unirradiated

100×

Irradiated

150×

Irradiated

500×

Fig. 21 — Type 316 Clad Adjacent to Fuel after
Irradiation (1.7×10^{22} nvt, 650°C at ID, 610°C
at OD, 2850 hr)

Fig. 22 — Vanadium Clad Adjacent to Fuel after Irradiation (1.7×10^{22} nvt, 650°C at ID, 590°C at OD, 2850 hr)

Unirradiated

500×

Irradiated — from gas plenum

500×

Irradiated — adjacent to fuel

500×

Fig. 23 — Vanadium-20 w/o Titanium Clad after
Irradiation (1.7×10^{22} nvt, 650°C at ID, 590°C
at OD, 2850 hr)

were examined metallographically. Little or no interaction was found. In this case, the interaction was in the form of carbides in the grain boundaries at the hottest region opposite the fuel.

Further investigations of the interactions are not planned, since V-20 w/o Ti is an undesirable clad for carbides, based on the out-of-pile incompatibility alone.

4.3.2 Microhardness

Microhardness measurements were made on the clads from each rod in the gas plenum area and the fueled area. These values are presented in Table 10 along with hardness data for unirradiated material.

In general, the irradiated clad was harder than the unirradiated material. Also, the V-20 w/o Ti clad and the type 316 stainless steel clad were harder in the fueled region than in the gas plenum region, while the vanadium clad was fairly uniform in hardness. The reaction zone on the inside edge of the fueled region of V-20 w/o Ti alloy clad was about 70 KHN harder than the unreacted clad.

4.3.3 Bend Tests

Relative ductilities of the clads from all fuel rods were determined by bending at room temperature a longitudinal section of clad 180° around a 0.5-in. diameter bar so that the clad inner surface became the convex side of the bend. All of the clad samples survived the bend tests without cracking. As evidenced by these bend tests and the slitting operations, the stainless steel was very ductile, the vanadium moderately ductile, and the V-20 w/o Ti less ductile.

TABLE 10 — KNOOP MICROHARDNESS OF CLADS
(100-g Load)

<u>Clad Type</u>	<u>Sample Location</u>	<u>Inside Edge</u>	<u>Inside Fourth</u>	<u>Middle</u>	<u>Outside Fourth</u>	<u>Outside Edge</u>	<u>Unirradiated</u>
V-20 w/o Ti	Fuel region	320*	243	243	249	255	} 189
	Gas plenum	189	222	222	222	227	
Type 316	Fuel region	240	203	233	249	246	} 153
	Gas plenum	212	199	187	197	197	
Vanadium	Fuel region	161	143	148	151	112	} 136
	Gas plenum	142	158	164	166	149	

*Reaction zone.

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