MASTER

UNC-5013

CARBIDE FUEL DEVELOPMENT

PROGRESS REPORT

PERIOD OF JANUARY 1, 1962
TO MARCH 31, 1962

A. Strasser
K. Taylor

May 28, 1962

Work Performed under Project 2178,
Contract AT(30-1)-2899 with the
United States Atomic Energy Commission

UNITED NUCLEAR
CORPORATION

DEVELOPMENT DIVISION
5 New Street,
White Plains, N.Y.

AND
THE CARBORUNDUM COMPANY

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UNITED NUCLEAR CORPORATION
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White Plains, New York

Subcontractor
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FOREWORD

The Carbide Fuel Development project is part of the AEC Fuel Cycle Development Program. The prime contractor is the United Nuclear Corporation, Development Division, formerly Nuclear Development Corporation of America (NDA), and the subcontractor is The Carborundum Company. The United Nuclear Corporation is performing the conceptual design, fuel evaluation, fuel irradiation, and irradiated fuel examination. The Carborundum Company is fabricating the fuel. Both companies are operating plutonium handling facilities.

The report covers progress from January 1, 1962 to March 31, 1962. Previous progress was reported in:

NDA 2140-2, Carbide Fuel Development – Phase I Report (Oct. 15, 1959)

NDA 2145-1, Carbide Fuel Development – Progress Report (Mar. 11, 1960)

NDA 2145-4, Carbide Fuel Development – Progress Report (June 13, 1960)

NDA 2145-5, Carbide Fuel Development – Progress Report (Aug. 30, 1960)

NDA 2145-6, Carbide Fuel Development – Phase II Report (Nov. 6, 1960)

NDA 2162-1, Carbide Fuel Development – Progress Report (Feb. 28, 1961)

NDA 2162-3, Carbide Fuel Development – Progress Report (June 1, 1961)

NDA 2162-5, Carbide Fuel Development – Phase III Report (Sept. 30, 1961)

UNC-5003, Carbide Fuel Development – Progress Report (Feb. 9, 1962).

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1. INTRODUCTION

Fuel made of a combination of UC and PuC has a potential of reducing the fuel cycle cost of existing fast breeder reactors. The fuel cycle cost reduction is anticipated because of increased burnup and increased power generation capability of UC-PuC, compared to presently available metallic fuels. The effect of high burnup and high power generation rate on the stability of UC-PuC is not known. However, UC irradiation tests show that carbide fuels are dimensionally more stable than metallic fuels. Because of the high melting point and good thermal conductivity of UC and the formation of solid solutions of UC-PuC with high melting points, there is further expectation that carbide fuels will be capable of high power generation rates.

The Carbide Fuel Development Program is concerned with the technology of the entire UC-PuC fuel cycle. The major goal of the program is to produce UC-PuC and to obtain data on its irradiation behavior for long burnups and at high power generation rates. In addition, other areas of the fuel cycle are being explored to discover potential problems. The program was initiated in May 1959 and as originally outlined covers a period of about four and one-half years. Program objectives are outlined below.

Conceptual Design

1. An analytical study of the effect of substitution of UC-PuC on heat transfer, physics, and cost of existing fast breeder reactors.

2. Conceptual design of rod-type, fuel element configurations which can be substituted directly in existing reactors.

The above was completed and was reported in NDA 2140-2.

Facility Design and Fabrication

1. Design and construction of a facility for carbide fabrication at The Carborundum Company.
2. Design and construction of a facility for carbide evaluation at United Nuclear.

The above was completed and reported in NDA 2145-6.

Fuel Fabrication and Evaluation

1. Explore various methods for preparation of fuel from powders and its fabrication into cylindrical pellets. High density (up to 13.1 g/cm^3) UC and UC-PuC pellets have been made.
2. Evaluate pellets by density measurement, chemical analysis, x-ray diffraction, hardness, metallography and fuel-cladding compatibility studies. This is complete for UC and initiated for UC-PuC. The UC-PuC studies will also include melting point, thermal stability, and coefficient of expansion measurements.

Fuel Irradiation

1. Irradiate clad fuel samples with burnup and maximum fuel temperature as the major variables. A minimum 2% burnup of all fuel atoms by fission and a minimum 650°C (1200°F) central fuel temperature were selected to establish the economic advantages of the fuel. This is complete for UC, and experiments for UC-PuC are under construction.
2. Make a post-irradiation examination. This is completed for UC.

2. SUMMARY

2.1 FUEL FABRICATION AND EVALUATION

Reproducibly high density UC-PuC pellets containing minor amounts of $(U,Pu)_2C_3$ were made with nickel sintering aid. Average densities of 95% and maximum densities of 96.6% of theoretical were obtained.

High density, single-phase UC-PuC pellets were made by sintering at high temperatures. Densities of 94% of theoretical appear to be attainable.

Improved identification of microstructural constituents was made by means of metallography, x-ray diffraction, and microprobe analysis. Development of specialized chemical analysis techniques continued.

2.2 IRRADIATION TESTS

The post-irradiation examination of the UC specimens was completed. Mass spectrometer analysis showed that 0.4% of the fission gases was released. Metallographic examination of the UC did not reveal any changes in structure. Metallographic examination of the metal specimen parts that were in contact with UC was completed. No evidence for interaction with niobium and slight evidence for interaction with stainless steel was found. The latter was not deleterious.

Perturbation factors were calculated for the UC-PuC experiments. Negotiations continued, in order to select the most suitable test reactor.

3. FUEL FABRICATION AND EVALUATION

3.1 INTRODUCTION

The goal of the fabrication studies is to produce high density solid solutions of PuC and UC by powder fabrication techniques. A high physical density (about 95% of theoretical) is desired to minimize fission gas release, give a high fuel density, and highest possible thermal conductivity.

The goal of the evaluation tests is to identify the material by density measurement, chemical analysis, x-ray diffraction, and metallography. Additional out-of-pile tests of fuel-cladding compatibility, thermal stability, melting point, and coefficient of expansion will survey properties of interest for in-pile tests.

At the beginning of February, work on this contract on the Fuel Fabrication and Out-of-Pile Evaluation task was stopped as per schedule and continued as work sponsored by the Joint U.S.-Euratom program. Work on these subjects from that date is being reported on separately.

3.2 CARBIDE POWDER PREPARATION AND PELLET FABRICATION

Fuel Fabrication and Evaluation

The major effort during the quarter was to make reproducibly high density UC-PuC pellets with nickel sintering aid. Work on producing high density UC-PuC pellets without nickel sintering aid was started on Contract AT(30-1)-2899 and continued on the Joint U.S.-Euratom Project.

A sintering experiment was carried out with the single-phase $(U_{0.8}Pu_{0.2})C_{0.95}$ powder, synthesized in one step at 1625°C from a 15-gram reaction batch. (See

experiment 2, Table 5, Report UNC-5003.) The reaction product was crushed in the "Spex" mixer and nickel added to constitute 0.12% of the powder by weight. The powders were then blended on the "Spec" mixer and pellets were pressed. These were sintered at 1550°C in helium, with a hold time of 1 hr. Results appear in Table 1.

Based on the good densities obtained from this powder, larger batches were synthesized for additional sintering studies. Two 50-gram batches were reacted. The reaction endpoint was determined by monitoring the exhaust gas drawn from the heated furnace, for carbon monoxide, as discussed in the previous progress report (UNC-5003).

Essentially single-phase material of about the intended carbon content was produced in these experiments. Results appear in Table 2.

The products shown in Table 2 were combined and ball-milled for 24 hr in a rubber-lined mill using stainless steel balls. Several sintering experiments were conducted with this product. In all sintering experiments the size of the pellet was 0.224 in. in diameter by about 0.200 in. long. The temporary binder was 1/4 w/o Carbowax-6000 dissolved in trichlorethylene, and the cold forming pressure was 30,000 lb/in.^{2} The pressed pellets were preheated to approximately 700 C in helium in an open tantalum crucible to volatilize the binder; the final sintering was then completed in a closed tantalum crucible.

The first experiment was conducted to determine the effect of small additions of free carbon on the sinterability of $(U_{0.8}Pu_{0.2})C_{0.95}$ compacts containing additional nickel. The results of previous experiments at Carborundum Co. with uranium carbide indicated free carbon to be a deterrent to densification. Companion pellets, with and without nickel additions, were carried along for purposes of comparison and also to confirm the results of previous experiments. Results appear in Table 3. The nickel sintering aid appears to be less effective in $(U_{0.8}Pu_{0.2})C_{0.95}$ pellets containing as much as 0.10 w/o additional carbon. The shorter, 1/2-hr sintering time at 1550°C produced pellets that had densities equiv-

Table 1 — Results of Sintering $(U_{0.8}Pu_{0.2})C_{0.95}$ + 0.12 w/o Ni Pellets

Composition of Pellet	Sintering Temperature and Hold Time	Average Density (five pellets), g/cm^3	Total Carbon, %	X-Ray Analysis
$(U_{0.8}Pu_{0.2})C_{0.95}$*	1550°C – 1 hr	13.00	4.70	Major (U,Pu)C Moderate $(U,Pu)_2C_3$ Faint $(U,Pu)O_2$ $a_0 = 4.968 \pm 0.001$ Å

*Chemical analysis of the product before sintering gave total carbon at 4.58%. X-ray analyses showed a single-phase (U,Pu)C solid solution with a unit cell $a_0 = 4.965 \pm 0.001$ Å.

Table 2 — Results of Synthesizing $(U_{0.8}Pu_{0.2})C_{0.95}$ Powder at 1625°C

Experiment No.	Reactants	Analysis of Reaction Mix Total Carbon, %	Hold Time, hr	Total Carbon in Product, %	X-Ray Analysis
1	UO_2, PuO_2, C	11.51	7½	4.61	Major (U,Pu)C Faint $(U,Pu)O_2$ $a_0 = 4.965 \pm 0.001$ Å
2	UO_2, PuO_2, C	11.51	5¼	4.60	Single phase (U,Pu)C $a_0 = 4.964 \pm 0.001$ Å

Table 3 — Results of Sintering Experiments with $(U_{0.8}Pu_{0.2})C_{0.95}$
(Effect of Nickel and Carbon Additions)

Experiment No.	Composition of Pellet	Temperature and Hold Time	Average Density g/cm^3	Total Carbon, %	X-Ray Analysis
1	$(U_{0.8}Pu_{0.2})C_{0.95}$	1550°C with a 1/2-hr hold	10.81	4.64	Major (U,Pu)C Faint $(U,Pu)_2C_3$
2	$(U_{0.8}Pu_{0.2})C_{0.95}$ +0.12 w/o Ni		12.90	4.68	Single phase (U,Pu)C* $a_0 = 4.9644 \pm 0.0003$ Å
3.	$(U_{0.8}Pu_{0.2})C_{0.95}$ +0.12 w/o Ni +0.10 w/o C		12.72	4.76	
4	$(U_{0.8}Pu_{0.2})C_{0.95}$ +0.12 w/o Ni +0.20 w/o C		12.74	4.88	
5	$(U_{0.8}Pu_{0.2})C_{0.95}$	1550°C with a one-hour hold	11.83	4.60	Major (U,Pu)C Faint $(U,Pu)_2C_3$
6	$(U_{0.8}Pu_{0.2})C_{0.95}$ +0.12 w/o Ni		12.91	4.77	Major (U,Pu)C Faint $(U,Pu)_2C_3$ $a_0 = 4.9644 \pm 0.0002$ Å
7	$(U_{0.8}Pu_{0.2})C_{0.95}$ +0.12 w/o Ni +0.10 w/o C		12.68	4.83	Major (U,Pu)C Weak $(U,Pu)_2C_3$ $a_0 = 4.966 \pm 0.001$ Å
8	$(U_{0.8}Pu_{0.2})C_{0.95}$ +0.12 w/o Ni +0.20 w/o C		12.75	4.94	Major (U,Pu)C Weak $(U,Pu)_2C_3$ $a_0 = 4.949 \pm 0.009$ Å

*Metallographic examination showed a two-phase structure. X-ray analysis was apparently not sufficiently sensitive to pick up the second phase.

alent to those produced by 1 hr sintering time. To determine if a still better product could be produced at lower temperatures, an additional sintering experiment was conducted at 1500°C with a hold period of 1 hr. Equivalent densities were obtained. Results are shown in Table 4. Companion pellets of straight $(U_{0.8}Pu_{0.2})C_{0.95}$ were carried along for comparative purposes. Results of chemical analyses for total carbon indicated that no change of composition had occurred during the sintering in either case.

Higher temperature sintering experiments were also conducted with straight $(U_{0.8}Pu_{0.2})C_{0.95}$ pellets (no sintering aid) fabricated from the product shown in Table 2 of this report. Temperatures investigated were 1850 and 1950°C with hold times of 1/4 hr. Densities were fair, but less than those obtained with nickel at lower temperatures. Results appear in Table 5. Analyses made on the Euratom-sponsored project showed there was no loss or negligible loss of Pu even at these high sintering temperatures. Additional high temperature sintering studies are being continued on the Joint U.S.-Euratom project.

Two synthesis experiments to form PuC and UC, from the carbon reduction of PuO_2 and UO_2, were performed. The purpose of this study was to determine the approximate temperature at which reduction of the oxides will proceed. For this study, compacts of the reaction powders (PuO_2 +C in one experiment, and UO_2 +C in a separate experiment) were placed in an open graphite crucible and heated slowly in the graphite resistance furnace. The exhaust gas, drawn from the heated furnace, was monitored periodically for carbon monoxide. Results of the two experiments indicate that the carbon reduction of both PuO_2 and of UO_2 begins at about 1050°C.

3.3 METALLOGRAPHY AND MICROPROBE ANALYSIS

The metallography of pellets described in Table 10 of UNC-5003 and Table 1 and 3 of this report was completed. By a combination of microstructure and x-ray analysis results, an interpretation of the microstructure was made. The light etching phase shown in Fig. 5 of this report, and Figs. 3 and 4 of UNC-5003

Table 4 — Results of Sintering Experiment at 1500°C with a 1-hr Hold

Experiment No.	Composition of Pellet	Average Sintered Density g/cm^3	Total Carbon, %	X-Ray Analysis
1	$(U_{0.8}Pu_{0.2})C_{0.95}$	12.35	4.61	Major (U,Pu)C Faint $(U,Pu)_2C_3$ $a_0 = 4.965 \pm 0.001$ Å
2	$(U_{0.8}Pu_{0.2})C_{0.95}$ + 0.12 w/o Ni	12.96	4.64	Major (U,Pu)C Weak $(U,Pu)_2C_3$ $a_0 = 4.965 \pm 0.001$ Å

Table 5 — Results of Sintering Experiments with Straight $(U_{0.8}Pu_{0.2})C_{0.95}$ Compacts (Higher Temperature)

Experiment No.	Temperature and Hold Time	Average Density g/cm^3	Total Carbon, %	X-Ray Analysis
1	1850°C – 1/4 hr	12.29	4.61	Major (U,Pu)C Faint $(U,Pu)_2C_3$ $a_0 = 4.965 \pm 0.001$ Å
2	1950°C – 1/4 hr	12.39	4.65	Single phase (U,Pu)C $a_0 = 4.966 \pm 0.001$ Å

is most likely $(U,Pu)_2C_3$. The sesquicarbide was present in all samples made with nickel sintering aid. $(U,Pu)O_2$ was identified in nearly all early samples (Table 10 of UNC-5003), made prior to the installation of the CO monitor on the furnace. The samples made from solid solution oxides contained both oxide and a spherical phase which is probably a higher metal content phase. The carbon content of these samples was 4.52 and 4.58 w/o. They did not contain sesquicarbide.

The effect of excess carbon on microstructure was studied on samples described in Table 3. The structure of samples with excess carbon showed large agglomerates of sesquicarbide.

Microprobe analyses of samples shown in Figs. 3 and 4 of UNC-5003 were completed at Mound Laboratories. The conclusions were that there is no nickel in either the light or dark matrix phases, rather the nickel is in the grain boundary phase. There is a variation of plutonium content between the light phase and the dark phase. Additional microprobe analyses will be made with improved standards.

3.4 CHEMICAL ANALYSIS TECHNIQUES

Plutonium analyses have been in progress to determine the effect of processing conditions on final plutonium content. Initial difficulties of ignition (sparking) of the highly pyrophoric carbide powders during dissolution in aqueous solutions were solved by special handling techniques. These involved slow, controlled dissolution of the powders in a capsule, within double glass containers. Subsequent plutonium analyses were done on the Joint U.S.-Euratom project and are reported separately.

4. IRRADIATION TESTS

4.1 INTRODUCTION

In order for UC-PuC fuel to reduce fuel cycle cost, several conditions must be met. The fuel has to be able to achieve high burnup, high operating temperatures, and high power. At least 2 a/o burnup of all fuel atoms by fission is desirable at central fuel temperatures above 1200°F (650°C) with maximum dimensional stability of the fuel and minimum release of fission products. Power generation rate should be at least equivalent to that of presently planned fuels. The objective of the irradiation program is to determine whether the high burnup, temperature, and power density required can be achieved.

The irradiation tests measure temperatures, burnup, power generation rate, dimensional stability of the fuel, and fission gas release.

4.2 POST-IRRADIATION EXAMINATION OF UC SPECIMENS

The post-irradiation examination of the UC irradiation test, capsule W1-1, was completed. A detailed description of the experiment and irradiation conditions was given in NDA 2162-5, pp. 29 to 41.

In summary, stainless steel and niobium-clad, 24% enriched UC specimens were irradiated. Each specimen contained 16 fuel pellets, 0.191 in. in diameter, for an approximate 3-in. fueled length. The void space was filled with helium. The UC pellets were made by cold pressing and sintering UC powder, which, in turn, was made by reacting UO_2 with carbon. The average density of the pellets was 94% of theoretical. Irradiation conditions were 760°C (1400°F) average central

fuel temperature; 315 w/cm (9.6 kw/ft) average heat generation rate; and 1.7 kw/cm^3 average power density.

Measurement of the length and diameter of four whole pellets from the niobium-clad specimen was completed (two pellets cracked during handling):

	Prior to Irradiation	After Irradiation
Avg. Diameter, in.	0.1910	0.1922
Avg. Length, in.	0.1833	0.1835

The 1-mil increase in diameter corresponds to a volume increase of 1.4%. The changes in dimension of these pellets are similar to the ones in the stainless steel clad specimen reported previously. The slightly increased diameter of the pellets corresponds to the ID of the cladding: 0.192 in. + 0.001-0.000.

Mass spectrometer analyses of the gas samples from the UC irradiation specimens were completed by ORNL. The sample taken from the stainless steel specimen showed that 0.4% of the fission gases generated during irradiation was released. This corresponds to the volume of fission gas that would leave the surface of the fuel specimen as a direct result of recoil of fission fragments. The sample bulb containing gas from the niobium sample had apparently leaked, and no analysis was obtained.

Post-irradiation metallography of the UC was completed by Battelle Memorial Institute. The structure and grain size of the UC appeared the same after irradiation as before (Figs. 1 and 2). The irradiated sample had slightly less free uranium metal in the structure; however, the difference probably is a difference between two separate samples.

Post-irradiation examination of metal specimen parts, which had been in contact with UC, was completed by the CANEL Facility of Pratt and Whitney Aircraft Division of The United Aircraft Corporation. The irradiation specimens operated for approximately 2800 hr at an average temperature of 760°C (1400°F) at the central thermocouple and center of the specimen end plug. The average

temperature at the ID of the stainless steel cladding was 490°C (920°F) and the niobium cladding 430°C (800°F).

No metallographic evidence for interaction was found between any of the niobium parts (central thermocouple well, cladding, end plug, washer) and UC. Microhardness measurements gave uniform hardness throughout the specimens. The material was ductile in a 90° bend test with the ID in tension. No cracks were seen by macroscopic examination.

A very slight interaction was noted between the type 316 stainless steel parts and UC. A narrow band of the stainless steel clad and end plug adjacent to the UC contained more intergranular carbide precipitate than was evident in the remainder of the specimen, which indicated a general carbide precipitation due to the thermal history of the specimen, rather than any interaction (Fig. 4). The stainless steel is shown prior to irradiation in Fig. 3. A comparison of microhardness measurements before and after irradiation showed hardening of the irradiated specimen due to the general carbide precipitation, and some preferential hardening of the surface near the UC. Results are given in Table 6. The material was ductile in a 90° bend test with the ID in tension. No cracks were seen by macroscopic examination. It can be concluded that the slight interaction is not sufficient to be deleterious to the properties of the cladding.

Burnup analyses by isotopic uranium and Cs^{137} determinations were completed by Phillips Petroleum Co., Idaho Falls, Idaho. The results are compared to calorimetric determination in Table 7.

The maximum burnup (at the pellet OD) and the minimum burnup (at the pellet centerline) were calculated using the average burnup based on isotopic analyses. The results are shown in Table 8.

4.3 DESIGN FOR UC-PuC IRRADIATION TESTS

The basic design conditions for the UC-PuC irradiation tests were given on pp. 32-33 of NDA 2162-5.

Two types of material will be irradiated:

Table 6 — Pre- and Post-Irradiation Hardness of Type 316 Cladding Tubing

Location	Hardness (DPH), 100-g Load	
	Unirradiated (Avg of 4 readings)	Irradiated (Avg. of 2 readings)
Near OD	263	307
Middle of wall	266	264
Near ID	308	430

Table 7 — Comparison of Average Burnup (MW-d/tonne) Determinations of the UC Specimens*

	Calorimetric (In-Pile)	Radiochemical (Cs^{137})	Isotopic (U)
Stainless Steel Clad Specimen	18,600	15,800	16,100
Niobium Clad Specimen	18,400	14,800	16,400

*One complete pellet from each specimen was dissolved for the radiochemical and isotopic analyses.

Table 8 — Radial Variation in Burnup

	Maximum Burnup, MW-d/tonne	Minimum Burnup MW-d/tonne
Stainless Steel Clad Specimen	24,600	9,000
Niobium Clad Specimen	24,900	9,200

1. 80% UC-20% PuC sintered at high temperatures (~1950°C) where densities of 12.6 to 12.8 g/cm^3 appear to be attainable (Fig. 5);
2. 80% UC-20% PuC-0.1% Ni sintered at low temperatures (~1500°C) where densities of 12.9 to 13.1 g/cm^3 are attainable (Fig. 6).

Since each capsule contains two specimens of 16 pellets each, each specimen will contain one type of material. They will be irradiated under essentially the same conditions.

Calculations were performed at United Nuclear to estimate the perturbation factor for the UC-PuC capsules in both the MTR and WTR. The WTR-MTR perturbation factor analyses were performed using the PALINDROME code, solving the one-velocity transport equation for cylindrical geometry in the P-3 approximation. WTR also performed transport calculations to compare with their critical experiment of one of the United Nuclear capsules. The results are compared to flux perturbation factors previously obtained by the Bradley Lewis correlation at United Nuclear in Table 9.

Both the United Nuclear and the WTR transport calculations are higher than the WTR critical experiment. Obvious reasons for the discrepancies were not apparent.

Neutron flux and exposure time requirements for the six capsules were estimated for both WTR and MTR using the Bradley Lewis correlation. Results are summarized in Table 10.

Based on the flux requirements obtained from the calculations, a complete cost and schedule analysis was made comparing the WTR, MTR, and ETR for the irradiation of the six capsules. The analysis indicated that the WTR would be the best, MTR second, and ETR third choice of reactors. At that time WTR shut down permanently and MTR informed United Nuclear that the number of experimental holes with the required fluxes were not available. Currently the ETR and GETR are being considered as potential test reactors for the program.

Table 9 — Comparison of Flux Perturbation Factors* for MTR and WTR

	Capsule 61 20% PuC-80% nat. UC, 0.191 in. OD fuel	Capsule 63 20% PuC-80% of 24% enr. UC, 0.173 in. OD fuel	Capsule 66 20% PuC-80% of 24% enr. UC, 0.191 in. OD fuel
MTR by Bradley Lewis correlation	0.135	0.090	0.083
MTR by transport calculation	0.259	0.193	0.161
WTR by Bradley Lewis correlation	0.165 (WTR)	0.119 (UNC) 0.116 (WTR)	0.108
WTR by UNC transport calculation	—	—	0.148
WTR by WTR transport calculation	—	—	0.178
WTR critical experiment	—	—	0.11

$$*\text{Flux perturbation factor} = \frac{\text{average flux in specimen}}{\text{unperturbed flux in experimental hole}} .$$

Table 10 — Flux and Exposure Time Requirements for UC-PuC Capsules

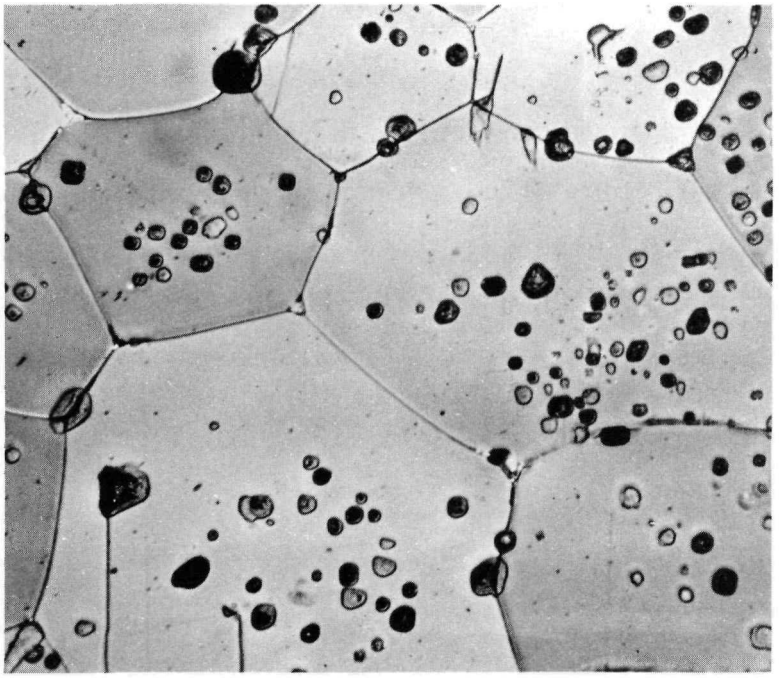
Capsule No.	Avg Burnup MW-d/tonne	Unperturbed Flux, nvt × 10^{-14}		Irradiation Time, days (cycles)	
		MTR	WTR	MTR	WTR
61	17,000	1.61	1.48	73 (5)	(4)
62	34,000	1.61	1.48	151 (10)	(8)
63	68,000	1.51	1.43	272 (17)	(13)
64	17,000	2.25	2.07	53 (4)	(3)
65	34,000	2.25	2.07	109 (7)	(6)
66	68,000	2.25	2.04	210 (14)	(10)

4.4 CONSTRUCTION OF UC-PuC EXPERIMENTS

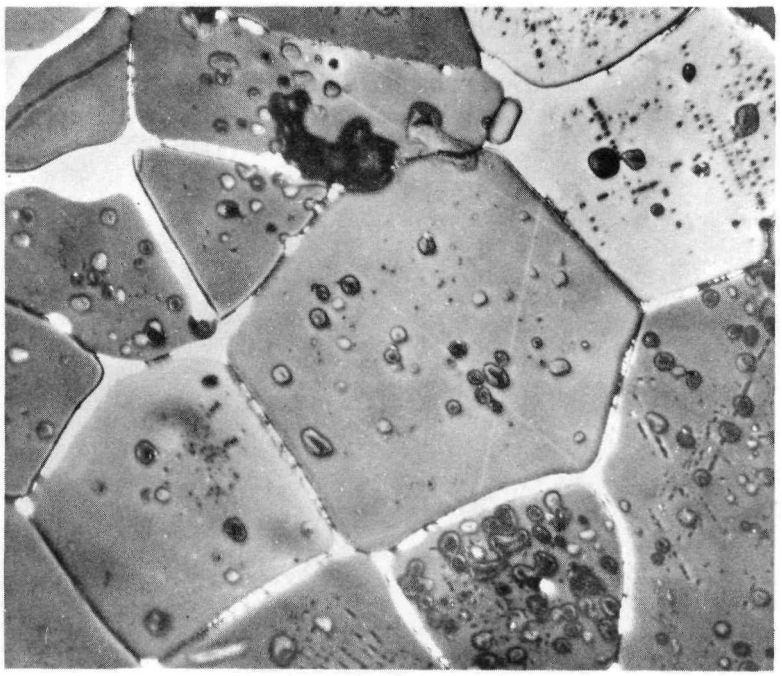
The synthesis of UC-PuC powder for the test specimens was initiated. The fabrication of pellets will be initiated and completed in the next quarter.

The delivery of individually calibrated thermocouples was completed and the vendor's calibration was spot-checked.

As previously reported, the fabrication of all capsule parts has been completed. Assembly of the capsules will start in the next quarter.



Section from center of pellet



Section from near surface of pellet

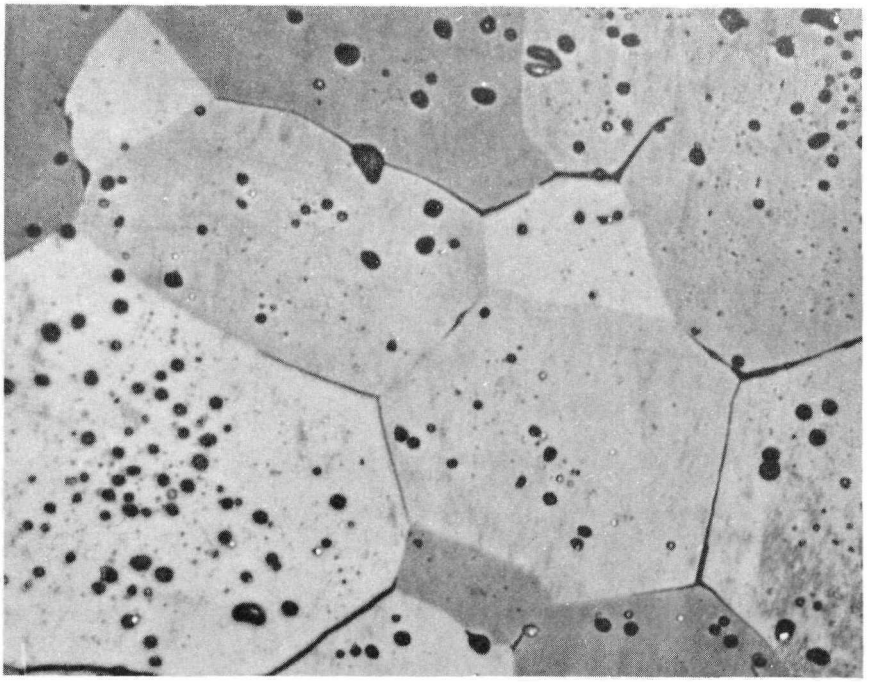
X-ray Analysis:
- Major UC
- Very faint UO_2
- No UC_2

Chemical Analysis:

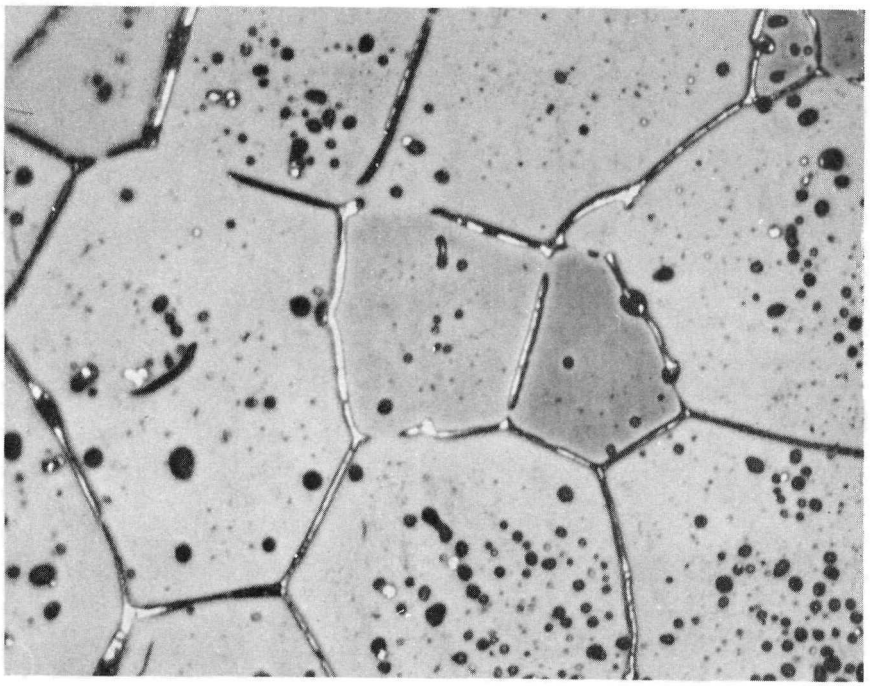
Total carbon	4.34%
Free carbon	< 0.05%
Uranium	95.37%
Iron	< 0.01%
Nitrogen	0.04%
Oxygen	0.32%

Enrichment: 24%

Fig. 1 — UC, unirradiated – 500× – nitric acid-acetic acid-water etch. The white phase at grain boundaries is metallic uranium. The dots are voids and oxide in the structure.

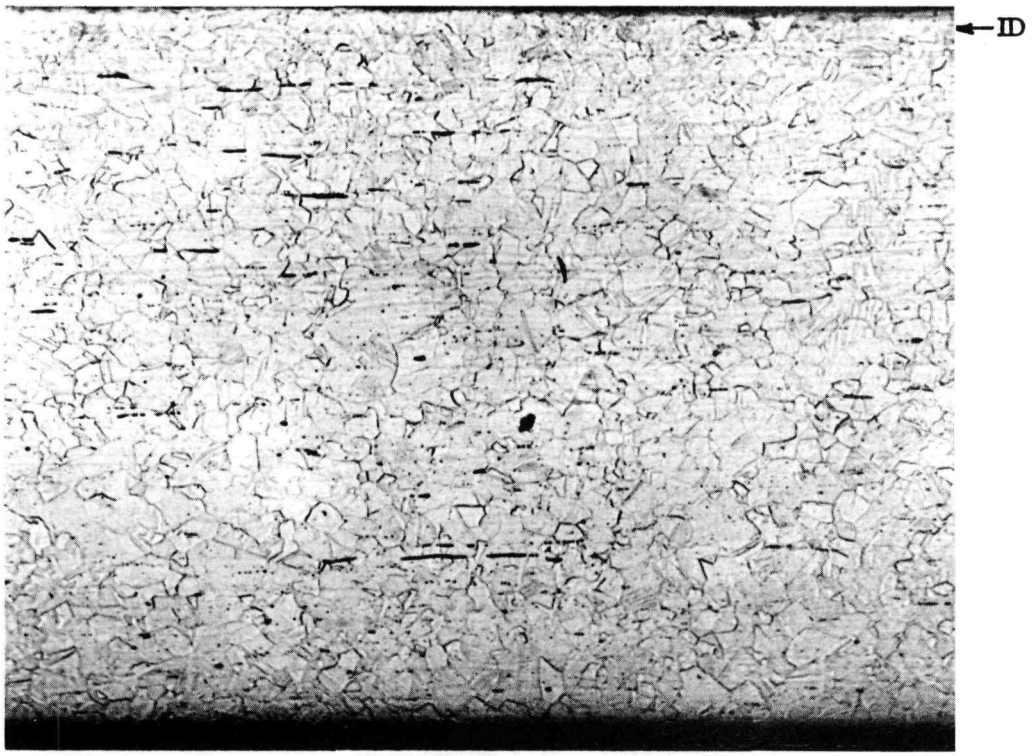


Section from center of pellet – irradiated ~10,000 MW-d/tonne – 1400°F

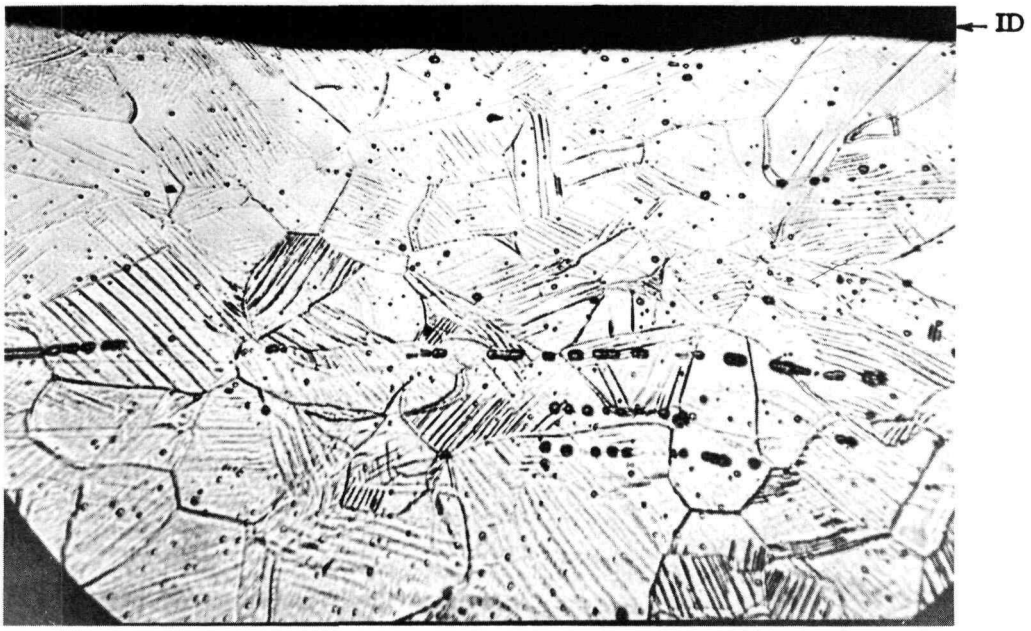


Section from near surface of pellet – irradiated ~24,000 MW-d/tonne – ~1150°F (at 2000 Btu/hr-ft^2-°F conductance)

Fig. 2 — UC, irradiated – 500× – nitric acid-acetic acid-water etch. Some of the metallic uranium oxidized (dark).



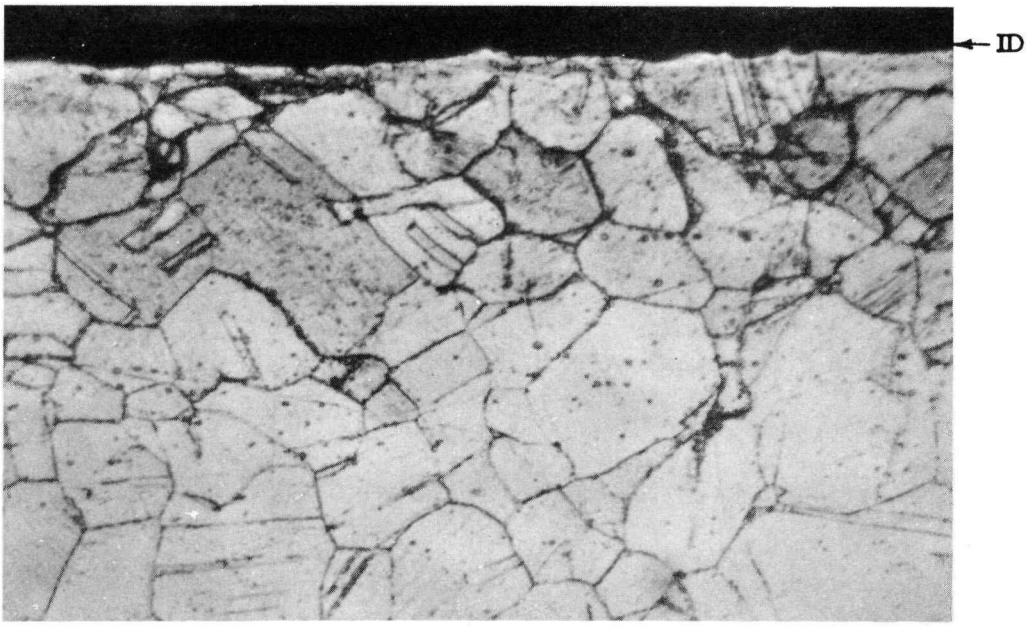
100×



500×

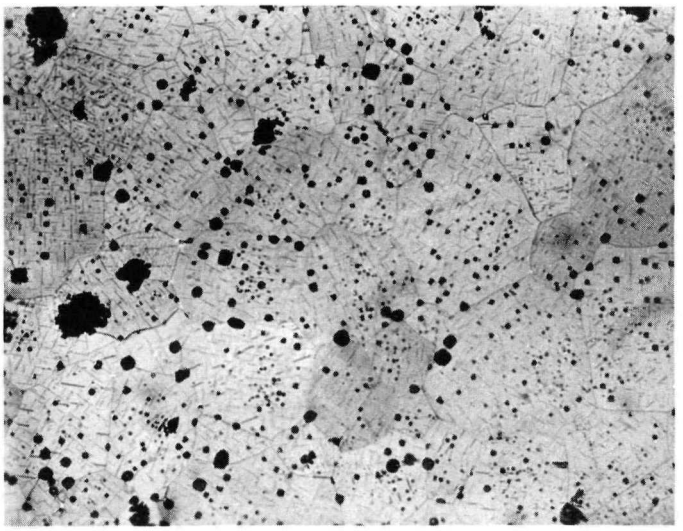
Fig. 3 — Type 316 stainless steel cladding, unirradiated – electrolytically etched

100×

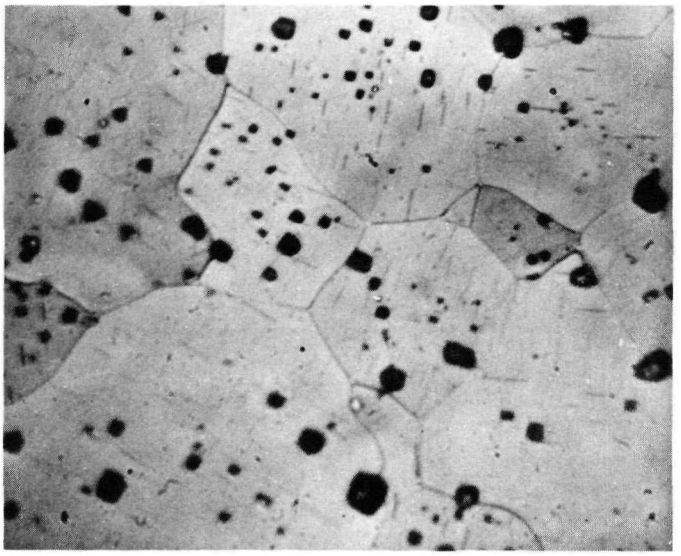


500×

Fig. 4 — Type 316 stainless steel cladding, irradiated – Kallings etch

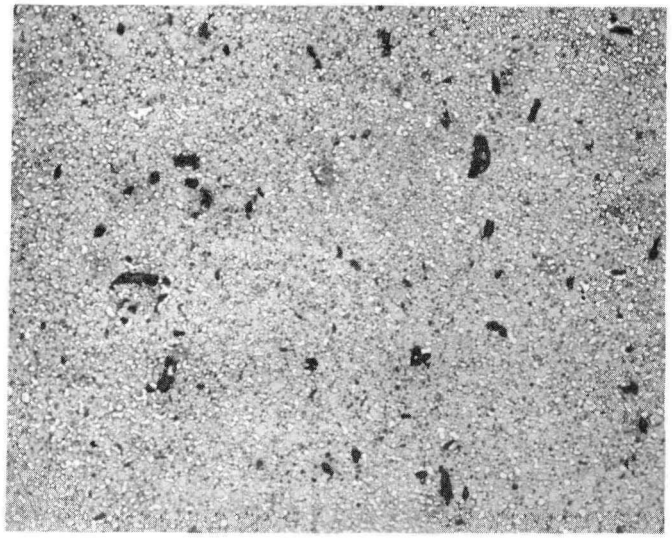


150×

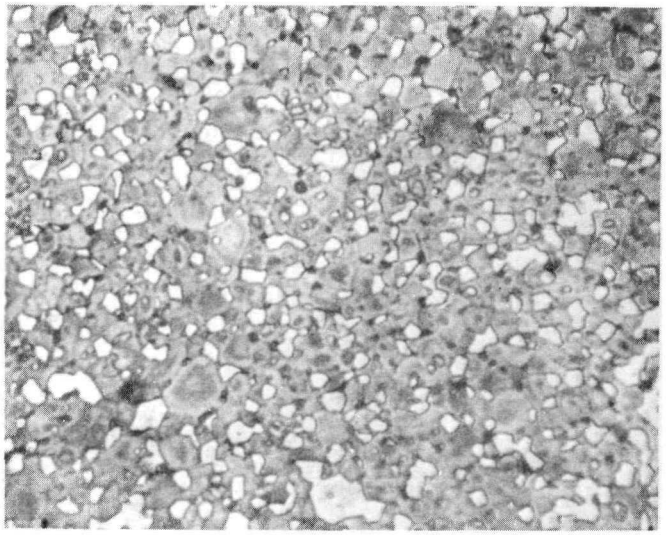


600×

Fig. 5 — 80% UC-20% PuC – nitric acid-acetic acid-water etch. Sintered 1950°C, 1 hr in helium – 12.8 g/cm^3, 4.5% C. X-ray analysis: single-phase (U,Pu)C, $a_0 = 4.961 \pm 0.003$ Å



150×



600×

Fig. 6 — 80% UC-20% PuC-0.1% Ni – nitric acid-acetic acid-water etch. Sintered 1550°F, 1 hr in helium – 12.91 g/cm^3, 4.77% C. X-ray analysis: major (U,Pu)C; faint $(U,Pu)_2C_3$, $a_0 = 4.9644 \pm 0.0002$ Å

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