204 c
11.23
DWR- 1630

MASTER

ORNL-4600

UC-25 - Metals, Ceramics, and Materials

FUELS AND MATERIALS DEVELOPMENT PROGRAM

QUARTERLY PROGRESS REPORT

FOR PERIOD ENDING JUNE 30, 1970

THIS DOCUMENT CONFIRMED AS
UNCLASSIFIED
DIVISION OF CLASSIFICATION
BY 9 H Kahn / amh
DATE 12/1/70

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

P7566

DISCLAIMER

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

DISCLAIMER

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

2. ^X DEVELOPMENT OF ADVANCED LMFBR FUELS

P. Patriarca J. L. Scott

The goals of this program are to investigate the properties and behavior of those U- and Pu-based ceramic fuels that we term conductors -- such as the mononitrides, carbonitrides, and monocarbides -- and to compare their potential as liquid-metal fast breeder reactor (LMFBR) fuel with that of $(U,Pu)O_2$, which by comparison is an insulator. Since the thermal conductivity of the ceramic conductors is about ten times that of $(U,Pu)O_2$, it is theoretically possible to operate a conductor at ten times the power density with the same temperature at the center of the fuel. In practice, heat-transfer limitations, thermal stresses in the cladding, and high rates of swelling at high temperatures limit the power density that can be achieved with thermal conducting fuels to about two or three times that of $(U,Pu)O_2$ -- still a challenging improvement. Additionally, the margins for transient overpower in the ceramic conductors are much higher than those for $(U,Pu)O_2$.

We seek to provide the information necessary for evaluating the true potential of nitrides, carbonitrides, and carbides in comparison to each other and to mixed oxides. We must define the structures, composition, and quality control required to achieve 150,000 Mwd/metric ton at peak linear heat ratings of 30 to 50 kw/ft. We must also demonstrate the possibility of a low-cost fuel cycle for manufacturing fuel with the needed properties. And since austenitic stainless steel is a poor conductor of heat, we seek to establish the physical and thermodynamic criteria for the new cladding material that will probably be required to exploit the conducting fuels.

Our approach to establishing the potential of advanced LMFBR fuels is to emphasize irradiation testing of Na-bonded $(U,Pu)N$ at high heat ratings to high burnups. Tests are being conducted in both thermal and fast reactors. Fabrication processes are being studied to provide material for irradiation testing and to establish the range of fuel densities and pore sizes that can be reasonably achieved. Compatibility is being studied to assess the importance of impurities in fuel, Na, and cladding

on compatibility and to resolve problems that arise from compositional changes in the fuel at high burnup. Emphasis is placed on type 316 stainless steel cladding throughout the program.

Synthesis, Fabrication, and Characterization of Nitride Fuels

V. J. Tennery

Synthesis and Fabrication of Mixed Nitrides (E. S. Bomar, F. J. Homan)

The purpose of this portion of our program is to develop methods for preparing (U,Pu)N powders of high purity and to fabricate pellets of variable but controlled density for use in irradiation test capsules. We primarily use cold pressing and sintering but have also used a limited amount of hot pressing. Starting powders are either prealloyed (U,Pu)N or physically mixed UN and PuN.

The densities of pellets fabricated for preparation of pins for irradiation in the Engineering Test Reactor¹ (ETR) were analyzed statistically to determine the source of observed variations. We found that the mean densities for an array corresponding to furnace position fell well within their calculated confidence intervals but that half the mean densities for an array corresponding to sintering runs fell outside their calculated confidence intervals. This led to the conclusion that pellet position in the furnace did not influence the sintered density but that there was a significant difference between the 13 sintering runs.

In the second step of the analysis, we assumed that the pellets made from a given batch of powder were divided into five different populations. Analysis of individual sets of data selected from each of the five sets of data used confirmed the assumption that the five represented different populations. Reexamination of the mean densities for the 13 sintering runs showed that the means fell well within the calculated confidence intervals for the respective batches of powder with two exceptions. These analyses therefore indicate that the sintered density was independent of furnace position, that the density (within the 85 to 90% observed) is strongly dependent on the batch of powder from which the

pellet was fabricated, and that the sintering run may have some small effect.

We also performed experiments to obtain additional information on the effect of several fabrication variables on the density and microstructure of sintered $(U_{0.8}, Pu_{0.2})N$ pellets. This is an extension of the work reported previously,^{1,2} in which we found that cold-pressed-and-sintered pellets that contained mechanical mixtures of UN and PuN densened only slightly when heated at 2100°C with a N₂ cover gas at 630 torr. The most weight lost during sintering at 2100°C was 4%.

Preparation of Pellets for Irradiation in the Experimental Breeder Reactor-II (E. S. Bomar)

We decided to prepare the (U,Pu)N pellets to be used in fabrication of fuel pins for irradiation in the Experimental Breeder Reactor-II (EBR-II) by conversion of prealloyed U and Pu by the hydride-nitride method.

We placed an order with WADCO for preparation by arc melting of 2 kg of 80% U-20% Pu alloy fully enriched in ²³⁵U. Shortly after the initial casting, a 500-g ingot spontaneously ignited and burned in a glove box with an air atmosphere. Neither the box nor its other contents were damaged. The remaining ingots were transferred to a box with an atmosphere of N₂ with an impurity level of 2 to 6% O₂ for storage before shipment. The ingots were subsequently found to have oxidized extensively while in storage.

We believe the pyrophoricity of the 20% Pu alloy is due to the presence of the ζ phase of the U-Pu binary. The composition of replacement castings was lowered to 18% Pu to reduce the amount of ζ phase that could be formed. Ingots prepared a year ago that contained 18.4% Pu, as analyzed at WADCO, gave no difficulty.

A set of metal containers with O-ring seals were fabricated in which to place additional castings for shipment. Shipment is scheduled for October 21, 1970.

Characterization of Nitride Fuels (V. J. Tennery)

X-Ray Diffraction Studies of Uranium-Plutonium Nitrides. - We determined the x-ray lattice constant of a control sample from each of the 13 lots of fuel pellets for the irradiation tests in the ETR. The parameter was determined by the use of a Straumanis film loading Debye-Scherrer camera and an extrapolation calculated on a digital computer. We have established that compositions in the pseudobinary UN-PuN system exhibit self-damage by irradiation down to at least 20 mole % PuN. The lattice constant increases with time approximately according to the observation of Bjorklund and Douglass³ for pure PuN except that the damage level at a given time decreases in the mononitrides in an apparently complicated fashion as a function of the Pu content. With $(U_{0.8}, Pu_{0.2})N$, a change of 0.0007 Å was observed in the lattice constant due to self-damage at 25°C in 79 days from the time the sample was cooled from high temperature. Therefore, the lattice constants reported here for the ETR capsule material were determined at sample ages between 39 and 68 days; the self-damage effect amounted to about 0.0006 Å over this range of ages. The measured constants and sample ages on the day the x-ray pattern was made are given in Table 2.1. Only minor variations were noted in the constant

Table 2.1. Lattice Constants of $(U_{0.81}, Pu_{0.19})N$ Fuel for Irradiation Tests in the Engineering Test Reactor

Sintering Lot	Pellet Numbers in Set	Period From Cooling to X-Ray Analysis (days)	Lattice Constant (Å)
1	155-169	48	4.89093 ± 0.00034
2	170-184	42	4.89095 ± 0.00032
3	185-199	43	4.89149 ± 0.00033
4	200-214	42	4.89086 ± 0.00037
5	215-229	39	4.89103 ± 0.00026
6	230-244	50	4.89093 ± 0.00027
7	245-259	49	4.89090 ± 0.00019
8	260-274	62	4.89093 ± 0.00019
9	275-289	68	4.89080 ± 0.00018
10	290-300	65	4.89099 ± 0.00026
11	301-315	64	4.89061 ± 0.00016
12	316-330	62	4.89109 ± 0.00055
13	331-339	62	4.89067 ± 0.00011

in the 13 sets. Five lots of powder were used to produce the pellets, and no correlation was found between the constant and the powder lot. The only diffraction lines observed on the patterns were those of the mononitride phase.

We produced several samples of $(U_{1-x}, Pu_x)N$, in which x had nominal values of 0.2, 0.4, 0.6, 0.8, and 1.0, by mechanically blending UN and PuN. We uniaxially pressed and heat treated pellets of these compositions at $1800^\circ C$ in about 600 torr N_2 . We then crushed, ground, and refabricated the pellets into right cylinders about 1/4 in. in outside diameter and 1/4 in. high. We ground these pellets to -80 mesh and again heat treated them at $1800^\circ C$ for 6 hr in about 600 torr N_2 . The specimens densened to about 80% of theoretical. The lattice constants of these mononitride compositions were determined within a few days after the heat treatment to preclude the effect of self-damage through irradiation on the measured lattice constants. The chemical content of each composition was determined and is given in Table 2.2 along with a typical spectrographic analysis. We assumed that the O and C impurities occupied N lattice sites in fixing the composition of the various pellets used in the study. The x-ray data obtained are shown in Fig. 2.1 and listed below:

<u>PuN Content (mole %)</u>	<u>Lattice Constant (Å)</u>
0	4.88918 ± 0.00017
19	4.89060 ± 0.00022
38	4.89470 ± 0.00015
58	4.90018 ± 0.00025
79	4.90382 ± 0.00013
100	4.90486 ± 0.00008

The lattice constant of these solid solutions does not obey a Vegard relationship as reported by Anselin.⁴ The lattice constants measured agree very well with published values for UN (ref. 5) and PuN (refs. 6, 7) when they have impurity levels of 800 ppm O or less. We believe the disagreement between our data and those of Anselin is due to the high O contents of the nitride samples reported.⁴

Table 2.2. Results of Chemical Analysis of (U,Pu)N Fuel for Irradiation in Engineering Test Reactor

Sintering Lot	Pellet Number	Aliquot Number	Composition, wt %					Mass Balance ^a
			U	Pu	N	O	C	
1	167	1	76.68	18.23	5.14	0.070	0.029	100.15
		2	76.42	18.19	5.15	0.070	0.018	99.86
2	182	1	76.13	18.51	5.10	0.030	0.021	99.80
		2	76.14	18.45	5.15	0.044	0.027	99.82
3	197	1	76.28	18.15	5.49	0.062	0.043	100.03
		2	76.12	18.36	5.63	0.081	0.034	100.23
4	212	1	76.31	18.22	5.30	0.075	0.039	99.94
		2	76.16	18.43	5.32	0.075	0.031	100.02
5	217	1	76.35	18.28	5.20	0.078	0.029	99.95
		2	76.24	18.34	5.25	0.065	0.031	99.93
6	242	1	75.70	18.29	5.46	0.029	0.019	99.50
		2	75.84	18.26	5.41	0.029	0.048	99.59
7	257	1	75.46	18.73	5.22	0.059	0.020	99.50
		2	76.14	18.86	5.08	0.048	0.027	100.16
8	272	1	76.36	18.42	5.19	0.098	0.013	100.08
		2	76.34	18.36	5.12	0.092	0.015	99.93
9	287	1	76.19	18.26	5.25	0.072	0.023	99.80
		2	76.49	18.30	5.29	0.089	0.011	100.18
10	298	1	75.99	18.48	5.32	0.029	0.015	99.84
		2	76.28	18.45	5.35	0.030	0.015	100.13
11	313	1	76.24	18.27	5.21	0.026	0.020	99.77
		2	76.32	18.36	5.39	0.020	0.020	100.12
12	328	1	76.46	18.34	5.19	0.039	0.018	100.06
		2	76.16	18.34	5.11	0.041	0.019	99.68
13	337	1	76.28	18.36	5.10	0.019	0.017	99.78
		2	76.46	18.40	5.12	0.021	0.016	100.02

^aMass balances are arbitrarily the sums of the numbers shown for a particular aliquot number. These balances do not include about 0.02% of trace metal impurities that were present.

Fig. 2.1. Lattice Constant in the UN-PuN System. [Refs. F. Anselin, *J. Nucl. Mater.* 10(4), 301-320 (1963); R. Lorenzelli, *Contribution to the Study of the System (U,Pu)C,N*, ORNL-tr-3019 (March 1969); M. W. Shupe et al., *Quarterly Status Report on the Advanced Plutonium Fuels Program*, April 1 to June 30, 1969 and Third Annual Report, FY 1969, LA-4284-MS, p. 71.]

We are now measuring the self-damage by irradiation in the x-ray samples; preliminary results indicate that the self-damage at 25°C in UN-PuN solid solutions is not proportional to the Pu content.

Studies of the Analytical Techniques for Uranium-Plutonium Nitrides. -

The chemical contents of a pellet selected randomly from each sintering lot of the ETR fuel were determined by the procedures described previously.⁸

The major analytical problem remaining for the nitrides represented by $(U_{0.8}, Pu_{0.2})N$ is the dissolution method used in the analyses for N. The method now used consists of sealing a weighed amount of sample into a glass tube with a mixture of HCl and HF and heating the tube in a furnace for 24 to 36 hr. After complete dissolution, the sample is analyzed by the Kjeldahl method in the conventional manner. Nitrogen is often lost during this dissolution procedure. Recently, we developed a new

dissolution method for UN that is far simpler, provides greatly improved precision, and requires only about 15 min.

We determined the U and Pu contents of the ETR samples coulometrically and checked the results by titrating known solutions made from high-purity standard Pu and U. The O analyses were performed by inert-gas fusion, and the C was analyzed by a conventional combustion technique. The chemical contents of selected pellets corresponding to the sets listed in Table 2.1 are given in Table 2.2. Duplicate analyses were performed for each of the major elements, and, in some cases, triplicate analyses were performed when there was any question about the precision, primarily in the analyses for N. The values in Table 2.2 are those accepted as representative of the fuel for the irradiation test in the ETR.

We recently observed that UN could be readily dissolved in phosphoric acid in a matter of minutes in a beaker on a hot plate. Several of these specimens dissolved this way were analyzed for N by the Kjeldahl method. We also dissolved UN in phosphoric acid with a small addition of Devarda's alloy, a reducing agent often used in nitride dissolutions to prevent the transformation of the N to gaseous N_2 . Essentially the same results were obtained with and without the use of Devarda's alloy. The phosphate apparently complexes the N in such a way that very little or none is converted to N_2 gas during dissolution and thus lost to the analysis. To date we have complete results for all major elements in four UN samples and results for N only, determined both with and without the use of Devarda's alloy, for two samples. Since each UN sample was sintered according to a slightly different program of temperature and N_2 pressure, some real variation may be expected in the N contents. The results are shown in Table 2.3. The nonmetal sums are very near the theoretical value for UN (5.556%). This dissolution method is now being tested for its applicability to mixed nitride specimens.

The results of chemical analyses for the various UN-PuN compositions listed previously are given in Table 2.4.

Table 2.3. Results of Nitrogen Analysis by Kjeldahl Method for Sintered UN Specimens Dissolved in Phosphoric Acid

Sample Number	Composition, wt %			Sum of Nonmetals (wt %)
	U	N	O + C	
1	94.61	5.40	0.07	5.47
2	94.60	5.38	0.07	5.45
3	94.52	5.45	0.05	5.50
4	94.49	5.42	0.09	5.51
5		5.34	0.04	5.38
		5.37	0.04	5.41
5 ^a		5.40	0.05	5.45
		5.45	0.05	5.50
6		5.48	0.04	5.51
		5.46	0.04	5.50
6 ^a		5.50	0.05	5.55
		5.48	0.05	5.53

^aDevarda's alloy added to phosphoric acid for these aliquots.

Table 2.4. Results of Chemical Analyses of UN-PuN Solid Solutions

Specimen Number	PuN Content (mole %)	Composition, wt %				
		U	Pu	N	O	C
1	0	94.45	0	5.41	0.09	0.030
2	19	77.00	17.40	5.47	0.08	0.026
3	38	58.60	35.90	5.45	0.05	0.020
4	58	40.20	54.80	5.45	0.04	0.030
5	79	21.35	73.10	5.30	0.04	0.020
6	100		94.39	5.40	0.10	0.020

A typical spectrographic analysis for the specimens used in the UN-PuN x-ray study is shown below for specimen 2 from Table 2.4 which had the composition $(U_{0.8}, Pu_{0.19})N$:

<u>Element</u>	<u>Concentration (ppm)</u>
Al	60
B	
Cr	
Cu	
Fe	120
Mn	
Mo	
Ni	
Pb	
Si	
Sn	
Ti	
V	
Zr	

The Pu and U were removed from the sample by ion absorption before analysis.

Compatibility of Mixed-Nitride Fuels with LMFBR Cladding Alloys

J. M. Leitnaker

Our approach to compatibility in high-performance fuel systems is to characterize interactions that can occur between the fuel and cladding material. A basic understanding of these reactions may point to methods for tailoring the fuel so that the reactions cannot occur and perhaps to methods for reducing reaction rates.

The Pu-Cr-N System (J. P. De Luca)

The first sample in the Pu-Cr-N system, a 1/4-in.-diam pellet with a Cr:PuN ratio of about 2:1, was heated at 1725°C for 2 hr under

200 torr N_2 . It was then examined by optical microscopy and x-ray analysis. From Fig. 2.2 it appears that a liquid formed in the Cr-PuN pellets; however, the pellet did not slump during annealing. The phases have not yet been individually identified. X-ray analysis revealed predominantly PuN and also Cr_2N and several unidentified lines. Chromium

Fig. 2.2. Plutonium Nitride (-325 Mesh) and Chromium Metal (-325 Mesh) in a Molar Ratio of 1:2. Heated at $1700^{\circ}C$ for 2 hr under 200 torr N_2 .

metal was not identified, although the x-ray lines for PuN may have masked the Cr lines. The unidentified lines may indicate the presence of a Pu-Cr- N_2 phase analogous to the U-Cr- N_2 phase identified by Spear and Leitnaker.⁹

A second pellet of Cr_2N and PuN in a 1:1 molar ratio was annealed 8 hr under 200 torr N_2 at $1500^{\circ}C$. The resulting pellet is being analyzed.

Thermodynamic Investigations of High-Performance Fuel Systems

J. M. Leitnaker

The goal of our thermodynamics program for high-performance fuels, the carbides, nitrides, and carbonitrides, is to be able to predict the limiting equilibrium behavior of fuel and cladding in fast-reactor environments. Our experiments, though limited to measurements out-of-reactor, are of value for defining and understanding the factors that affect fuel performance.

Thermodynamics of Plutonium Nitride (J. P. De Luca)

For our study of the compatibility of mixed nitride fuels with cladding materials, we are building a system for measuring N_2 pressures over the nitrides and carbonitrides of Pu to define the effects of temperature, pressure, and composition on the behavior of the fuel toward cladding materials. The system was described previously.¹⁰

We continued to calibrate and check this piece of equipment. The residual-gas analyzer and microbalance were both put into operation and found to work satisfactorily.

Irradiation Testing of Nitride Fuels for LMFBR Applications

T. N. Washburn

The objective of this phase of the program is to establish the irradiation performance of the U-Pu nitrides and carbides. The nitrides and carbides have a thermal conductivity about 5 to 10 times higher, a theoretical density 30% higher, and a metal content in the compound 7% higher than the oxides. These properties make the nitrides and carbides strong contenders as advanced fuel for an LMFBR.

Thermal-Flux Tests (C. F. Sanders)

The initial series of thermal-flux irradiation tests consists of two uninstrumented capsules of four fuel pins each that will be

irradiated in the ETR. These tests are of the "screening" type and are being used to evaluate the performance of (U,Pu)N fuel synthesized from metal. The fuel pins operate at a peak linear heat rating of 30 kw/ft to burnups of 30,000 and 60,000 Mwd/metric ton.

The fuel is cold-pressed-and-sintered (U,Pu)N pellets with densities from 86 to 91% of theoretical. The fuel pin has a 0.010-in. radial gap between the fuel pellet and the inside surface of the type 316 stainless steel cladding. The gap will be filled with NaK-19 (Na-19% K) to enhance heat transfer.

We completed the fabrication of ten fuel pins, eight of which will be irradiated in capsules 43-N1 and 43-N2. The fuel and NaK-19 were loaded into the cladding, and the fuel pins were transferred in a sealed container for final closure welding. The NaK-19 was loaded¹¹ into each fuel pin and found to contain less than 95 ppm O. The quality and level of the NaK bond were determined by standard eddy-current techniques. At the time the fuel was loaded, we did not have a complete chemical analysis of the fuel; therefore, we decided to load each pin from one batch of fuel. This gave us a fuel column about 2.5 in. high instead of the design 3 in. high.

During the final inspection of the fuel pins, we found three defects. In one pin (45-1-C) the cold NaK level was about 3/32 in. below the top of the fuel column because the NaK loading tube was short. In another pin (45-1-J) we detected a broken pellet. Since this pellet had not been broken before loading, we concluded that it had been broken during handling or centrifuging of the fuel pin. The third defect (pin 45-1-G) was a void in the NaK-19 bond between fuel and cladding; we centrifuged the pin four more times but could not remove the void.

Since we had to use one of the pins with a defect, we chose pin 45-1-C, which had the low NaK level, because our calculations showed that the NaK level during operation will be above the fuel. The fabrication of the capsule will be completed next quarter.

Table 2.5 lists the design conditions, and Table 2.6 gives a summary of the measured characteristics of each fuel pin fabricated.

Table 2.5. Design Conditions for Mixed Nitride Capsules N-1 and N-2

Fuel	(U _{0.8} ,Pu _{0.2})N
Fuel diameter, in.	0.245
Fuel density, % of theoretical	88-90
Fuel length, in.	2.5
Cladding material	Type 316 stainless steel
Cladding diameter, in.	0.3
Cladding wall thickness, in.	0.0175
Bond between fuel and cladding	NaK-19
Gap between fuel and cladding, in.	0.010
Capsule material	Type 304 stainless steel
Capsule diameter, in.	0.501
Capsule wall thickness, in.	0.025
Instrumentation	None
Heat rating, kw/ft	30
Temperature of fuel center, °C	925
Temperature of fuel surface, °C	610
Temperature of inner surface of cladding, °C	585
Temperature of outer surface of cladding, °C	495
Number of pins per capsule	4
Number of capsules per test	1

The safety analysis for the irradiation experiment was completed and subsequently reviewed and approved by the ETR safety personnel.

After the Be reflector was changed in the ETR during April, there were uncertainties in peak flux in various locations. Therefore, we performed a new flux mapping for these experiments and decided to use positions L-12-SE and F-10-NW.

Table 2.6. Fuel Pins for Mixed Nitride Series I Irradiation Experiment

Tube	Fuel Pin	Pellet Batch	Pellet Density (% of theoretical)	Fuel Height (in.)	Pellet Diameter (in.)	Remarks
45-1-C	N1-1	A-7-C-2	90.2	2.473	0.240-0.243	NaK-19 level about 3/32 in. low
45-2-F	N1-2	A-6-C-2	87.2	2.453	0.245-0.248	
45-1-K	N1-3	A-6-C-1	87.8	2.448	0.245-0.248	
45-1-E	N1-4	A-5-C-2	85.5	2.573	0.240-0.245	
45-1-J	Sample	A-4-C-1	85.7	2.593	0.239-0.243	Broken pellet
45-1-D	N2-1	A-8-C-2	88.5	2.523	0.240-0.244	
45-2-G	N2-2	A-8-C-1	88.5	2.508	0.2405-0.244	
45-1-F	N2-3	A-7-C-1	89.2	2.487	0.240-0.243	
45-2-E	N2-4	A-6-C-3	86.8	2.441	0.247-0.248	
45-1-G	Sample	A-5-C-3	86.1	2.573	0.2405-0.244	Void in NaK-19

Notes

1. E. S. Bomar, Fuels and Materials Development Program Quart. Progr. Rept. March 31, 1970, ORNL-4560, pp. 40-45.
2. E. S. Bomar, Fuels and Materials Development Program Quart. Progr. Rept. Dec. 31, 1969, ORNL-4520, pp. 58-66.
3. C. W. Bjorklund and R. M. Douglass, Quarterly Status Report on the Advanced Plutonium Fuels Program, April 1 to June 30, 1969, and Third Annual Report, FY 1969, LA-4284-MS, pp. 73-75.
4. F. Anselin, J. Nucl. Mater. 10(4), 301-320 (1963).
5. J. M. Leitnaker, R. A. Potter, K. E. Spear, and W. R. Laing, High Temp. Sci. 1(4), 389-400 (1969).
6. M. W. Shupe et al., Quarterly Status Report on the Advanced Plutonium Fuels Program, April 1 to June 30, 1969, and Third Annual Report, FY 1969, LA-4284-MS, p. 71.
7. R. Lorenzelli, Contribution to the Study of the System (U,Pu)C,N, ORNL-tr-3019 (March 1969).
8. V. J. Tennery, J. L. Botts, and J. P. De Luca, Fuels and Materials Development Program Quart. Progr. Rept. March 31, 1970, ORNL-4560, pp. 49-55.
9. K. E. Spear and J. M. Leitnaker, "Phase Investigations in the U-C-N System." Submitted for publication.
10. J. P. De Luca, Fuels and Materials Development Program Quart. Progr. Rept. Sept. 30, 1969, ORNL-4480, p. 52.
11. C. F. Sanders and J. D. Sease, Fuels and Materials Development Program Quart. Progr. Rept. March 31, 1970, ORNL-4560, p. 61.