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Westinghouse Astronuclear Laboratory

Contribution to the

Nineteenth High Temperature Fuel Committee Meeting

November 16, 17, and 18, 1964

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General Introduction

Advances in the high temperature fuel element materials program for the development of the NERVA nuclear rocket engine will be described. The principal topics will include: a review of current reactor tests, NERVA fuel element development, high temperature studies of fuel materials, and results from irradiation test experiments.

Nuclear Rocket Reactor Tests

Test firings of nuclear rocket reactors in the months of August and September have been by far the most successful and significant to date in the ROVER-NERVA program. On August 28, 1964, a reactor in the LASL-KIWI series (KIWI-B4E) operated smoothly at design conditions for 8 minutes. The liquid H₂ flow control system performed well and maintained the flow at 69 lb/sec until the dewars were almost empty. Fuel exit gas temperature was approximately 4000°R (1900°C). Essentially no reactivity loss was noted during the run. Because the run was smooth and examination of test data indicated no serious difficulties, a re-start test was performed on September 10, 1964. The primary objective was to examine re-start problems. The re-start test ran smoothly for a planned two minute operation.

Following this test, the NRX-A2 core was put on the test stand at the Nuclear Rocket Development Station proving ground north of Las Vegas, Nevada. The NRX-A series of nuclear reactors provide a thrust of 50,000 pounds at a power level of 1,000 megawatts. The NRX-A2 was the first hot test of a NERVA nuclear rocket reactor.

On September 24, 1964, the NRX-A2 test was successfully conducted with the reactor attaining 85 per cent of full power while operating for more than six minutes above 50 per cent of full power. The duration of the test was limited by available hydrogen gas needed for turbine drive and reactor cooldown.

Harold B. Finger, manager of the joint AEC-NASA Space Nuclear Propulsion Office, said of the test, "Combined with the KIWI B-4 test run earlier this year by the Los Alamos Scientific Laboratory, this NERVA reactor test is further clear proof that this country has achieved a major advance in rocket propulsion - nuclear rocketry.

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These tests prove that the nuclear rocket concept is sound and that nuclear rockets can achieve the previously predicted high performance that will be required for future space missions. "

Preliminary post-operative examination of the KIWI-B4E fuel elements indicates excellent performance of the NbC coated graphite fuel elements. A major problem area, defined by the test, involves peripheral leakage of hydrogen into the core of the reactor. The leakage hydrogen gives rise to corrosion on the external uncoated faces of the fuel elements. Post-operative examination of NRX-A2 is currently in progress.

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NERVA Fuel Element Development

The NERVA fuel material consists of a dispersion of PyC coated UC_2 particles in a graphite matrix. The 19-hole, 52 inch long, hexagonal shaped fuel element is fabricated by extrusion. The 19-coolant channels, called "bores", and a small portion of the hot exit end of the element are vapor coated with NbC to protect the graphite from hot hydrogen corrosion.

The nominal fuel element environment during full-power (1,000 megawatt) operation is as follows: fuel element temperatures vary from 190° to $2200^\circ C$; the hydrogen gas coolant temperatures vary from -170° to $2100^\circ C$ at a pressure of approximately 550 psi, and a nominal flow rate of 0.002 pounds hydrogen per second per coolant channel. The maximum specific power level in the fuel element is 4 kilowatts per cm^3 .

The direction of fuel development work since last May has been towards improved hydrogen corrosion resistance. Fuel elements are being made in production facilities at LASL, WANL, and Y-12 which may be expected to last over 30 minutes at the design operating conditions, based upon encouraging results from hydrogen corrosion testing of single elements.

I. Non-fueled Tip Development - C. E. Vogel, R. Patterson, and D. Puchy

Introduction

A significant advance in the fuel element design has been the bonding of a short non-fueled graphite tip to the end of the element before it is given its NbC protective coating. The quality of the coating over the non-fueled tip is superior to that over fueled graphite, particularly on external flat surfaces. Improved resistance to hydrogen corrosion at the hot exit end has resulted.

A major fuel materials problem in the NERVA fuel element has been poor coating adherence on the exit ends of elements. Since it was known that coatings reproducibly adhered to the ends of non-fueled elements, it was decided to attempt to attach non-fueled element sections to the exit ends of fueled elements with carbonaceous

cements. A butt type joint was chosen. Destructive testing showed that a butt type cement joint had sufficient strength for use in the NERVA reactor core. The NRX-A3 core will be assembled to elements with non-fueled wafer tips of 1/4 inch length.

Although work is still being carried out to optimize the strength of the cemented joint, a successful process has been developed. The carbonaceous cement consists of a mixture of graphite flour and varcum (partially polymerized furfuryl alcohol) catalyzed with maleic anhydride. Both fueled and non-fueled sections of the element are cured to 250°C after extrusion. The cement is applied to the end faces of both the fuel element and the non-fueled tip. The pieces are pushed together and held under slight pressure by a spring. Excess cement is removed. The element (with the tip on the end) is then cured to 140°C at 20°C/hr temperature rise. The spring fixture is then removed. Next, the element undergoes a carbonizing heat treatment to 850°C and a graphitizing heat treatment to 1900°C.

The improved adherence of the NbC to the tip of the element has greatly improved the corrosion behavior in the hot hydrogen tests at the exit end of the element. There is a sharp reduction in the flaking of NbC from the end coat.

High Temperature Studies of Fuel Materials

Introduction

The NERVA graphite fuel element contains up to 28,500 spherical fuel particles per cubic centimeter. Each fuel particle has a nominal 100 micron diameter UC_2 core which is coated with a 25 micron thick layer of PyC (pyrocarbon). The whole particle has a nominal 150 micron diameter. The pyrocarbon, when intact, protects the UC_2 against hydrolysis during storage and against reaction with the gases in the NbC vapor coating process. In addition, the pyrocarbon deters the migration of UC_2 through the graphite matrix to the NbC coating. The PyC coating can be damaged by heating to high temperatures where the UC_2 reacts with the coating and changes its structure to one more porous and permeable. It may also be damaged by mechanical treatment in the mixing, blending, and extrusion of the element. The NERVA fuel development program is concerned with both areas of uranium migration and mechanical damage.

I. Uranium Migration and Evaluation of Vendor Coated Particles - A. L. Feild

Work has continued on the evaluation of qualification, production, and development type coated particles from participating vendors. Thermal migration testing has been conducted at $2300^{\circ}C$ for 4 hours on all pre-production qualification and production lot samples. The development of a quantitative thermal migration test, based on micro-radiographic techniques, was described in detail in the report for the 18th High Temperature Fuels Committee meeting. The results of this standard migration test on recent sample lots are shown in Table I. These data indicate that the quality of fuel beads has continued to be excellent on production lot samples of both General Atomic and Carbon Products fuels. The $2300^{\circ}C$ - 4 hr. test requirement was met adequately and the minimum unaffected coating thickness after test was from 2-6 microns less than the 18-25 micron coating thickness before test. The percentage of particles completely migrated after this test was generally within the 3 per cent maximum set forth in WANL Specification 30050-B. The $\bar{X} = 1.6\%$ was considerably

greater than the 5 micron minimum required for acceptance and normally exceeded 10 microns.

In anticipation of future requirements for higher fuel operating temperatures, some work has been done in testing thermal migration characteristics above 2300°C for exposures of 1/2 hour. These results are shown in Table II. It must be remembered that the UC_2 core is molten at temperatures above approximately 2450°C , and the liquid can take a certain amount of carbon into solution. This means that the integrity of the coating is much more difficult to maintain than would be expected from the increased diffusion rate of uranium solely due to the higher temperature. The data show that none of the samples, all of which passed the 2300°C - 4 hour test, were able to survive either 2500°C or 2600°C for 1/2 hour with low percentages of total migration. Therefore, these results indicate that considerably more effort is needed in pyrocarbon coating development to produce fuel particles with acceptable migration resistance above 2300°C .

Another important aspect of fuel particle behavior is the effect of thermal cycling on the coating integrity. Reactor restart will introduce a thermal cycling effect which might be of considerable importance relative to fuel lifetime and performance. To simulate such a condition, thermal cycling of two particle samples was performed in the following manner:

Heat up to 2300°C	15 min.	
Hold at 2300°C	30 min.	Repeat 7 times
Cool down to black heat	15 min.	

The results of this treatment are shown in Table III. The performances of both the Union Carbide and General Atomic particles in this test were very good, with the former material apparently showing an actual improvement with thermal cycling. The average coating thickness of the two materials after the first cycle were within one micron of each other. After the seventh cycle, they remained essentially the same as the start. The GA material showed a greater total of completely migrated particles than

Table I
Thermal Migration Tests - Current Materials

Sample ⁽¹⁾	Test Condition	Avg. Coating Thick. (μ)	$\bar{X} - 1.6\sigma$ ⁽²⁾	Minimum Coating Thick. (μ)	% Particles Migrated
GA 44001	As-received	22.1	18.9	20.8	0
		19.1	17.2	17.8	0.3
GA 44002	As-received	20.7	16.9	18.3	0
		21.5	17.8	--	--
GA 44003	As-received	18.3	14.8	16.4	0
		19.6	16.1	--	--
GA 44001	2300°C - 4 hrs.	16.5	10.3	13.0	6.8
		17.5	13.3	14.4	2.8
GA 44002	2300°C - 4 hrs.	15.2	8.3	12.9	3.2
		17.2	12.2	14.4	3.2
GA 44003	2300°C - 4 hrs.	18.6	15.1	14.9	0.9
		19.3	14.8	16.3	0.9
		19.3	16.3	17.4	0.9
GA 44004	2300°C - 4 hrs.	21.5	15.5	19.3	0.7
		20.9	15.7	17.8	0.7
UC 43001	As-received	24.4	14.9	21.8	1.0
		23.3	15.6	20.1	1.2
UC 43002	As-received	21.7	11.3	15.9	2.4
		22.5	12.7	--	--
UC 43001	2300°C - 4 hrs.	17.4	8.8	12.7	3.8
		18.2	10.5	14.1	2.3
UC 43002	2300°C - 4 hrs.	16.2	6.1	12.3	0.36
		17.5	6.3	14.0	0.36
UC 43003	2300°C - 4 hrs.	16.9	7.5	14.4	1.4
UC 43004	2300°C - 4 hrs.	16.1	5.9	13.4	1.4

(1) GA -- General Atomic
UC -- Union Carbide (Carbon Products)
Dav -- Davison

(2) $\bar{X} - 1.6\sigma \geq 5\mu$ for acceptance

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Table II

Thermal Migration Tests at Temperature Above 2300°C

<u>Sample</u>	<u>Test Condition</u>	<u>Minimum Coating Thick. (μ)</u>	<u>% Particles Migrated</u>
GA 44002	2500°C - 1/2 hr.	5.6	44.0
GA 44003	2500°C - 1/2 hr.	6.9	52.0
UC 43002	2500°C - 1/2 hr.	--	100.0
UC 43001	2600°C - 1/2 hr.	--	100.0
LASL 588	2500°C - 1/2 hr.	--	83.0
UC D 49A	2500°C - 1/2 hr.	--	80.0
	2600°C - 1/2 hr.	--	100.0
UC D 54A	2600°C - 1/2 hr.	--	100.0
Y-12 Li 111	2600°C - 1/2 hr.	--	100.0

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Table III
Thermal Cycling Tests - Development Samples

<u>Sample</u>	<u>Test Condition</u>	<u>Avg. Coating Thick. (μ)</u>	<u>X - 1.65</u>	<u>Min. Coating Thick. (μ)</u>	<u>% Particles Migrated</u>
UC D 54A	2300°C 7 cycles at 1/2 hr. each	18.7	13.6	15.8	1.4
		19.1	14.7	16.1	0
		19.2	13.4	16.8	0.6
		19.6	15.8	17.0	0.6
		20.2	14.7	17.6	0
		19.5	14.7	18.0	0
		20.9	16.3	17.3	0
GA EK 43	2300°C 7 cycles at 1/2 hr. each	19.6	6.2	12.9	0.5
		19.2	12.8	15.3	0.7
		20.4	9.5	18.1	2.0
		19.3	12.3	15.6	1.1
		18.6	9.0	14.6	4.8
		19.2	11.4	16.3	4.9
		19.9	9.2	16.8	1.1

the UC material. The latter fuel showed practically no migration during any of the individual cycles.

The above cycling tests are preliminary. There is an apparent anomaly in the reported values of particles totally migrated due to variations in behavior of individual particles in samples. However, it appears that particles currently available in production quantities will be able to survive a large number of thermal cycles without appreciable deterioration.

II. Mechanical Damage Studies - W. Brizes

During the fabrication of NERVA fuel elements, the elements are leached in HCl to remove some of the exposed UC_2 before the NbC vapor coating step. Leach losses vary, but range from 7 to 10 per cent of the uranium in the extruded elements. The majority (a calculated maximum of 6.1 per cent) of the loss is thought to be from particles whose PyC coating was cut during the machining of the graphitized elements to final dimensions. It is postulated that the excess loss is from fuel particles whose PyC coatings are damaged during fabrication prior to leaching and the UC_2 , below the exposed surface, is subsequently leached.

To determine whether damage is occurring during the first stages of fabrication, i.e., the mixing, grinding or extrusion processes, samples of fuel mixes (particles, varcum, lampblack, flour), were taken from each of the above mentioned processes and leached to remove all of the exposed uranium. Two types of leaching mediums were used, chlorine gas at $1000^{\circ}C$ and a 50/50 nitric acid solution at $90^{\circ}C$. Both methods appeared equally effective in leaching out the exposed uranium. Also, both methods indicated that as fabrication proceeded fuel particle damage increased up to a maximum of 0.4 per cent. The leaching results are given in Table IV for the final extrusions only since they contain the cumulative damage of all prior processes.

The maximum of 0.4 per cent uranium leachable will certainly not account for the uranium lost in the fuel element leaching process. Thus, if damage is occurring to the particles below the exposed surface, it must be between the final extrusion and

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Table IV
Uranium Leachable from Damaged Particles Separated from
As-extruded Elements

<u>Sample</u>	<u>Application</u>	<u>% U Leachable by Chlorine at 1000°C for 3 hrs.</u>	<u>% U Leachable by 1 N HNO₃ at 90°C for 4 hrs.</u>
GA pre-production batch	NRX A-3 NRX A-4	0.392*	
CP pre-production batch	NRX A-3 NRX A-4	0.395	
NUMEC Batch 1119	NRX A-2		0.220
NUMEC Batch 1111	NRX A-2		0.324
NUMEC Batch 1115	NRX A-2		0.266
NUMEC Batch 1112 Batch 1253	NRX A-2		0.072 and 0.047 (2nd analysis)
Davison No. 25019			0.128 and 0.145 (2nd analysis)
3M No. 12017			0.164 and 0.169 (2nd analysis)

* As-received particles showed 0.0081 per cent leachable uranium. Heat treatment for four hours at 2300°C increased the leachable uranium to 0.280 per cent.

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the leaching processes, i. e., one of the heat treatments (cure, bake, or graphitization) and, or, the machining of the element to final dimensions. Work is presently being directed to determine if damage is actually occurring during one of these later processes.

III. Reflection Electron Microscope Studies

The reflection method of electron microscopy which has been applied to fuel studies at Hanford Laboratories, has been used to observe the surface topography of sectioned and polished PyC coated UC_2 fuel particles in a graphite matrix by a reflected electron beam. The beam makes a low angle of incidence and reflection ($<30^\circ$) with the sample and then strikes the photographic plate at right angles. Thus, when looking at the photomicrographs, one sees the coated particle at a shallow angle relative to the surface of the sample and the circular coated particle appears elliptical.

Electron photomicrographs of two migrated fuel particles are shown in Figures 1 and 2 with corresponding photomicrographs made with an optical microscope. It is clear that the topography of the coated particles becomes apparent when viewed by this technique. In Figure 1 for example; 1) the graphite matrix is well bonded to the pyrocarbon coating, 2) there is an apparent difference in height across the delamination, 3) the migration zone appears rough, 4) the UC_2 core is raised due to differences in hardness of the carbide and pyrocarbon in polishing, thus casting a shadow into the foreground, 5) the lines on the UC_2 core are probably ppt's of UC, and 6) the surface is contaminated with polishing debris.

High temperature studies can be made with this technique since the actual sample is used, not a replica. Temperatures approaching $2700^\circ C$ have been achieved at the Hanford Laboratory with a modification of the JEM 6A microscope used here; and thus, it seems probable that migrating coated particles can be observed at high temperature with the use of electron reflection techniques.

IV. Further Notes on Diffusion in Niobium Carbides - W. Brizes and L. Cadoff

An equation expressing the interdiffusion coefficient for diffusion of carbon through NbC was given at the 18th High Temperature Fuel Committee meeting. The

600X Optical Microscope

3500X Electron Reflection Microscope (JEM GA)
at Room Temperature

Figure 1 - Photomicrographs of a Coated UC_2 Particle from
Element 9-11269, 51 inches As-graphitized

600X Optical Microscope

3500X Electron Reflection Microscope (JEM GA)
at Room Temperature

Figure 2 - Photomicrographs of a Coated UC_2 Particle from
Element 9-11269, 51 inches As-graphitized

interdiffusion coefficient for carbon through Nb_2C was recently measured. Carbon diffuses through Nb_2C much more rapidly than through NbC . As a result of the new measurements, the previously reported results have been corrected for carbon diffusion in Nb_2C . The new D_{NbC} coefficients are a factor of 3.5 to 4.0 times higher. The general equations fitting the experimentally determined carbon interdiffusion coefficients in NbC and Nb_2C phases in the temperature range of 1600° to 2200°C follow:

$$D_{\text{NbC}} = 1.0 \exp(-76 \text{ kcal/RT})$$
$$D_{\text{Nb}_2\text{C}} = 0.09 \exp(-55 \text{ kcal/RT})$$

A compilation of D_{NbC} and $D_{\text{Nb}_2\text{C}}$ data is given in Figure 3.

Inert marker (microhole) experiments have shown a very slight movement of the markers in the NbC phase during carburization. The movement can be related to the volume expansion of the NbC on carburizing. However, by assuming the marker movement was entirely due to Nb diffusion, it was determined that the diffusivity of Nb in NbC is at least a factor of ten lower than that of C in NbC . Therefore, the intrinsic carbon diffusivity in NbC , or $D_{\text{NbC}}^{\text{C}}$, can be calculated to be equal to two times the interdiffusion coefficient D_{NbC} . A topical report, WANL-TNR-185, describes the results and methods of calculations.

V. High Temperature Mechanical Properties of NbC - C. A. Anderson

An investigation of the high temperature strength of NbC was undertaken to assist in the interpretation of the behavior of vapor deposited NbC coatings. Exploratory measurements were made by compression, tensile, and burst tests on various types of NbC samples. The preliminary data obtained are given in Table V. It is important to note that the strength values given are the minimum values obtained for each type of test. Significant ductility of NbC was apparent in all tests at 2200°C and above.

Figure 3 - Diffusion Coefficients in Niobium Carbides
 as a Function of Reciprocal Temperature

Table V
Results of Preliminary Examination of High Temperature
Mechanical Properties of NbC

<u>Test</u>	<u>Specimen</u>	<u>Temperature (°C)</u>	<u>Minimum Strength Value (psi)</u>
Compression	Vapor deposited NbC tubes (1 mil wall thickness x 0.1 inch diameter)	2500	2200 (compression)
Burst test	Carburized Nb foil (C/Nb 0.96; 0.3 inch diameter x 1-2 mils thick)	2200	5000 (tension)
Tensile test	Carburized Nb wire (5 inches long x 50 mils diameter)	2500	7000 (tension)

Steady-State Irradiation of NERVA Fuel Material - D. Jacobs and C. Glassmire

Introduction

This work is a continuation of the work reported at the last meeting. The earlier results were based on metallographic examination of three fuel pellets. Sixteen NERVA fuel samples have now been evaluated, and the results are compiled below.

I. Evaluation of Results

A total of sixteen NERVA fuel samples have been irradiated. Each sample received one of the following pre-irradiation treatments:

1. Heated in vacuum to 2400°C for 0.5 hour then held at 100°C in 100 per cent relative humidity for 24 hours.
2. Machined from a NERVA fuel element which had been subjected to a 2.75 minute hydrogen corrosion test at 2500°C (surface temperature).
3. No pre-irradiation treatment.

The test conditions and results are summarized in Tables VI, VII, and VIII. The untreated NERVA fuel material exhibited excellent stability during the irradiation. All untreated samples retained their structural integrity and suffered no dimensional changes. Temperatures and irradiation times are given in the tables.

The irradiated, hydrogen-corroded and hydrolized materials suffered considerable damage. All samples became friable and were severely cracked. Those hydrogen corroded samples which remained intact suffered significant dimensional changes ($\geq 4\%$). Cracking was also observed in those unirradiated hydrogen corroded and hydrolized samples which were subjected to the longer heat treatments (≥ 130 minutes). However, this cracking was considerably less severe than that observed in the irradiated samples.

The crack patterns which developed in the irradiated hydrogen corroded samples were indicative of radial and tangential stresses. It is believed that these stresses were the results of (a) thermal gradients and (b) the residual stresses inherent to the anisotropic fuel material.

During the irradiation the largest contribution to the developed stresses was undoubtedly made by the thermal gradient. However, thermal gradients were essentially absent during heat treatment and the stresses developed within the control samples were derived primarily from residual stresses.

In light of the above and considering the experimental results it is clear that hydrogen corrosion and UC_2 hydrolysis both serve to degrade the mechanical strength of NERVA fuel. The effects of hydrogen attack are readily explained by conventional corrosion theory, i.e., a general loss and degradation of matrix material. However, the effect of UC_2 hydrolysis is not so clearly understood. Preliminary studies indicate that a hydrolysis treatment, similar to that received by the irradiation samples, reduces the room temperature, compressive strength of NERVA fuel material by approximately 30 per cent.

Neglecting the water of hydration, the molar volume of UO_2 is 6.63 per cent greater than that of UC_2 . Consequently, if the UC_2 and graphite matrix are in intimate contact, hydrolysis would result in the formation of high local stresses. These would contribute locally to the residual stresses thereby inducing micro-cracking and/or serving to reduce the apparent strength of the fuel material.

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Table VI
Test Results - As-Received Fuel Material

Pellet	<u>Mean Temp., °C</u>		<u>Exposure Time (min.)</u>	<u>Burn-up₃ (fiss/cm³) × 10¹⁷</u>	<u>Results</u>
	<u>Center</u>	<u>Surface</u>			
3M	2420	2125	25.2	0.334	Pellet intact and exhibited good structural integrity. No dimensional changes. No significant UC ₂ migration.
3M-B*	2250	2250	25.2	---	Same as 3M
4M	2255	1990	100.2	1.22	Same as 3M
4M-B*	2220	2220	100.2	---	Same as 3M
6M-B*	1980	1980	270.0	---	Same as 3M
5M-B*	1860	1860	130.8	---	Same as 3M
5M	1580	1420	130.8	0.964	Same as 3M
6M	1380	1240	270.0	1.65	Same as 3M
9-1**	3050	2650	76.5	1.36	Pellet intact and exhibited good structural integrity. No dimensional changes. Metallography not completed.
9-2**	2900	2530	76.5	1.22	Same as 9-1
9-3**	2730	2400	76.5	1.10	Same as 9-1
9-4**	2560	2780	76.5	0.98	Same as 9-1

* Unirradiated controls

** Data are preliminary.

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Table VII

Test Results - Pellets Machined from an Element which
 Failed the Hydrogen Corrosion Test

Pellet	Mean Temp., °C		Exposure Time (min.)	Burn-up ₃ (fiss/cm ³) x 10	Results
	Center	Surface			
3T	2240	1970	25.2	0.299	Pellet was intact but covered with a network of cracks. Increase in diam: 4.3 %.
4T	2140	1890	100.2	1.13	General description same as 3T. Pellet extremely friable and broke during normal handling. Increase in diam: 4 %.
3T-B*	2080	2080	25.2	---	Pellet retained structural integrity.
4T-B*	2040	2040	100.2	---	Pellet retained structural integrity.
6T-B*	1740	1740	270.0	---	Pellet intact but fine cracks present.
5T-B*	1700	1700	130.8	---	Pellet intact but fine cracks present.
5T	1500	1360	130.8	0.883	Same as 4T
6T	1240	1120	270.0	1.44	Same as 3T. Increase in diam: 4.3 %. Increase in height: 3.4 %.

* Unirradiated control pellets.

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Table VIII
Test Results - Fuel Material with Hydrolyzed UC₂

Pellet	Wt. Gain (%)	Mean Temp., °C		Time (min.)	Burn-up (fiss/cm ³) x 10 ⁻¹⁷	Results
		Center	Surface			
3B	0.235	2555	2250	25.2	0.375	Pellet friable. Broken while still in capsule. Sample covered with network of cracks.
3B-B*	0.276	2420	2420	25.2	---	Good structural integrity.
4B-B*	0.238	2400	2400	100.2	---	Good structural integrity.
4B		2375	2100	100.2	1.31	Same as 3B
6B-B*	0.307	2020	2020	270.0	---	Sample intact but some fine cracks present.
5B-B*	0.316	2020	2020	130.8	---	Same as 6B-B
5B	0.224	1760	1580	130.8	1.11	Same as 3B
6B	0.204	1520	1380	270.0	1.88	Pellet broken into extremely small pieces (almost a powder)

* Unirradiated control pellets.

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Atomic Energy Act 1954