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OAK RIDGE NATIONAL LABORATORY

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POST OFFICE BOX P
OAK RIDGE, TENNESSEE

ORNL
CENTRAL FILES NUMBER

CF- 56-12-117

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"External Transmittal Authorized"
COPY NO. 20

DATE: December 27, 1956

SUBJECT: Irradiation Experiment on Aluminum Fuel Element
Containing Uranium Dicarbide Dispersion.

TO: Listed Distribution

FROM: R. J. Beaver

Ref: (a) ORNL-MTR-30, AEC Form 320, dated June 5, 1956.

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I. Purpose

In accordance with reference (a) a request was submitted to evaluate the irradiation behavior of aluminum plate-type fuel elements, in which the uranium was incorporated as a dispersion of uranium dicarbide in aluminum. The uranium dicarbide is one of the important materials in establishing reliable and economical procedures for manufacturing aluminum MTR-type fuel elements under the Atoms for Peace Plan in which uranium, with a maximum enrichment of 20% in the U-235 isotope, is specified. This enrichment specification necessitates a high total uranium investment in each fuel element. Uranium dicarbide offers the potential of accomplishing this objective, however, its behavior under actual operating conditions must be determined.

II. Design Test Component

The test assembly contains 19 plates spaced at 116 mil intervals. The design is patterned after the MTR Mark XI fuel element. The seventeen inner plates are 50 mils thick, with a clad-core-clad thickness of 15-20-15 mils. The top and bottom plates are 65 mils thick, with a clad-core-clad thickness of 22 1/2-20-22 1/2. Pertinent design details are listed in Table I. Materials data of the fuel core are itemized in Table II. The enrichment factor may vary between 19.0 and 20.0%, which will reflect a slight change in the mass of total uranium.

III. Physical and Nuclear Data

The physical and nuclear data for the test element are listed in Table III. The test component contains approximately 3743 grams of aluminum, 967 grams of uranium, 17 grams of silicon, and 98 grams of carbon, calculated on the basis of a 60 cm effective length. The total absorption cross-section of these materials for

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thermal (0.025 ev) neutrons is 402.92 cm^2 and 354.86 cm^2 with and without water, respectively. The test component contains 190 grams of U-235, resulting in a fission cross-section of 280.00 cm^2 . The thermal and fission cross-section values used for these calculations were 680 and 575 barns, respectively.

IV. Construction of Test Component

The test plates will be fabricated with techniques similar to those used in roll bonding MTR fuel plates. The fuel plates will be curved and joined into an assembly by brazing, utilizing current procedures for brazing MTR assemblies. The test assemblies will be identified as ORNL-MTR-3 and ORNL-MTR-4, respectively, and will contain markings on the side plates to permit machining of individual plates from the assembly for post irradiation examination.

V. Time of Irradiation

It is desirable to irradiate ORNL-MTR-3 to a minimum burnup of U-235 atoms of 30%, or 5.5×10^{20} nvt, and ORNL-MTR-4 to a minimum burnup of U-235 atoms of 60%, or 14.4×10^{20} nvt.

R. J. Beaver

RJB: lmp

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TABLE I

DESIGN DATA ON URANIUM DICARBIDE-ALUMINUM
TEST COMPONENT FOR MTR IRRADIATION-ORNL-MTR-30

Number of Elements	2
Total Irradiation, nvt	ORNL-30-3 5.5×10^{20}
	ORNL-30-4 14.4×10^{20}
Number of Fuel Plates Per Element	19
Spacing Between Plates, mils	$116 \pm 10\%$
Width of Fuel Section, inches	$2.470 \pm 1/32$
Length of Fuel Section, inches	$23 \frac{5}{8} \pm 1/2$
Thickness of Fuel Section, mils	20
Thickness of Cladding, mils	long plates: $22 \frac{1}{2}$
	short plates: 15
Overall Plate Width and Length, inches	long plates: $2.8 \times 28 \frac{5}{8}$
	short plates: $2.8 \times 24 \frac{5}{8}$
Core Mixture Composition, wt %	UC ₂ - 59.08
	Al - 40.29
Core Mixture Composition, vol. %	UC ₂ - 25.65
	Al - 74.35
U-235 Enrichment, %	19.5 ± 0.5
U-235 Content per plate, gms	10.00 ± 0.10
U-235 Content per Assembly, gms	190 ± 1.00
U-235 Fuel Distribution, gm/cm ²	Minimum: 0.0255
	Maximum: 0.0277
Type of MTR Assembly	Mark XI
Side Plates	ORNL Dwg. D-21090

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TABLE II

MATERIALS DATA OF POWDERED FUEL CORE COMPACT
 CORE VOLUME BASED ON ONE PLATE - 1.181-in.^3 or 19.34 cm^3

<u>Material</u>	<u>cm³ Volume</u>	<u>% Volume</u>	<u>gms Weight</u>	<u>% Weight</u>	<u>Gm-Mole Fraction</u>	<u>Atomic %</u>
Uranium Dicarbide	4.96	25.65	56.05	59.08	--	--
U-235	--	--	10.00	--	--	--
U-234 and U-238	--	--	40.89	--	--	--
U-Total	--	--	50.89	--	0.215	10.32
Carbon	--	--	5.16	--	0.430	--
Aluminum	<u>14.38</u>	<u>74.35</u>	<u>38.82</u>	<u>40.92</u>	<u>1.439</u>	<u>--</u>
Totals	19.34	100.00	94.87	100.00	2.084	10.32

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TABLE III

PHYSICAL AND NUCLEAR DATA ON URANIUM DICARBIDE-
ALUMINUM TEST COMPONENT FOR MTR IRRADIATION-ORNL-MTR-30

<u>Material</u>	<u>cm² Cross-Sectional Area</u>	<u>cm³ Volume (1)</u>	<u>gms Mass</u>	<u>cm² Absorption Cross-Section (2)</u>	<u>gms Fission Cross-Section</u>
Uranium-235	--	--	190.00	330.98	280.00
Uranium-234 and 238	--	--	776.91	5.45	--
Carbon	--	--	98.04	0.01	--
Uranium Dicarbid	1.57	94.24	1064.95	336.44	--
Aluminum	23.10	1386.37	3743.17	18.38	--
Silicon	0.12	7.19	16.77	0.04	--
Water (3)	37.47	2248.20	--	48.06	--
<hr/>					
Total Solids or Metals	24.79	1487.80	4824.89		
Metal to Water Volume Ratio	0.66				
Total Absorption Cross-Section, cm ²				402.92	
Fission Cross-Section, cm ²					280.00

(1) Based on 60 cm effective length

(2) Cross-section of 0.025 ev neutrons

(3) Includes one-half of water spacing between adjacent MTR fuel assemblies

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