UNCLASSIFIED

MASTER

OAK RIDGE NATIONAL LABORATORY

Operated By

UNION CARBIDE NUCLEAR COMPANY

 POST OFFICE BOX P
 OAK RIDGE, TENNESSEE

 EXTERNAL TRANSMITTAL
 AUTHORIZED

 ORNL
 CENTRAL FILES NUMBER

58-10-20

DATE: October 17, 1958

COPY NO. 41

 SUBJECT: Phase II - Foreign Reactor - Fuel Sample Irradiation of UC_2 -Al
 Dispersions. Irradiation Request ORNL-MTR-35

TO: R. J. Beaver

FROM: C. F. Leitten and W. C. Thurber

ABSTRACT

The following data highlight the results of calculations presented in this report to determine the irradiation testing of Irradiation Request ORNL-MTR-35. This phase of the program entails exposure of UC_2 -Al dispersions in the beryllium reflector of the MTR.

Capsule surface temperature	178°F
Central core temperature	180.4°F
Fissionable material (U_{235})	0.127 g/in.
Heat output	4.58×10^2 w/in.
Heat flux	1.613×10^5 Btu/hr-ft ²
Capsule OD and length	1.125-in. OD x 6.5 in.
Flux desired	unperturbed 1×10^{14} neutron/cm ² -sec
Hot-spot factor F_h	1.91
Hot-channel factor F_b	1.20

NOTICE

This document contains information of a preliminary nature and was prepared primarily for internal use at the Oak Ridge National Laboratory. It is subject to revision or correction and therefore does not represent a final report. The information is not to be abstracted, reprinted or otherwise given public dissemination without the approval of the ORNL patent branch, Legal and Information Control Department.

UNCLASSIFIED

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.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

- A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or
- B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission to the extent that such employee or contractor prepares, handles or distributes, or provides access to, any information pursuant to his employment or contract with the Commission.

PHASE II - FOREIGN REACTOR - FUEL SAMPLE IRRADIATION OF UC_2 -Al DISPERSIONS
Irradiation Request ORNL-MTR-35

C. F. Leitten and W. C. Thurber

I. INTRODUCTION

This report details Phase II of the irradiation program described in ORNL-CF-58-2-109¹ and entails determining the integrity and suitability of a 60 wt % dispersion of UC_2 in aluminum as a fuel for Foreign Reactor Fuel Elements.

II. DESIGN AND FABRICATION

This phase of the program involves irradiation of miniature fuel plates in the beryllium reflector of the MTR to burn-up values of 20, 40, 60 and 80% of the U_{235} atoms. The miniature fuel plates consist of a 20% enriched, 60 wt % dispersion of UC_2 in aluminum as the core material which is hermetically sealed in 1100 aluminum by hot-rolling. The manufacturing stages for the miniature fuel plates are outlined below.

1. Miniature cores 0.490 in. x 0.265 in. x 0.560 in. were machined from a powder-metallurgy master core. The master core was prepared by cold-pressing a blended mixture of UC_2 and Al at 33 tsi and sintering in vacuum for 3 hr at 600°C.

2. The cores were pressed into 1100-Al frames which were 0.250 in. in thickness and Al cover plates 0.193-in. thick were then attached by making fusion welds along the long sides of the sandwich. No filler rod was used.

3. The billets were then hot-rolled at 600°C using varying reductions per pass to a thickness of 0.055 in.

4. The hot-rolled plates were flux-annealed 1 hr at 610°C and then cold-rolled to 0.050 in.

5. After a stress-relief anneal at 570°C for 45 min, the plates were marked, sheared and machined to 1 in. x 6 in. x 0.050 in.

6. Radiographs were then made and the plates were given a final blister anneal for 1 hr at 610°C.

7. In light of the catastrophic corrosion effects observed with UC_2 , the miniature plates were inserted in 80°C water for 24 hr to ensure that

the UC_2 was heremetically sealed in aluminum. No dimensional changes were observed in these plates. Similiar plates possessing deliberate defects swelled badly under the same conditions.

8. Upon the completion of the corrosion tests, the plates were numbered and assembled in "leaky rabbit" test capsules.

The critical plate dimensions and fuel loadings for the UC_2 -Al dispersion fuel plates are listed in Table I. Measurements of finished core length and width were obtained from the radiographs.

III. FUEL-PLATE TEMPERATURE CALCULATIONS

A. Surface-Plate Temperature

Previous analysis of the water-flow velocity for miniature "rabbit-type" capsules, described in ORNL-CF-58-2-109, indicated a coolant velocity of 8.7 ft/sec. Employing this value and an unperturbed flux of 1×10^{14} neutron/cm²-sec, the fuel-plate surface temperature can be calculated by the following expression:

$$\frac{Q}{A_S} = h\Delta T = h(T_1 - T_2) \quad (1)$$

where:

- Q = heat generation in the plate (Btu/hr)
- A_S = total surface area of the core (ft)²
- h = heat-transfer coefficient of the coolant (Btu/hr-ft²-°F)
- T_1 = plate-surface temperature (°F)
- T_2 = bulk-coolant temperature (115°F)

The heat-transfer coefficient (h) is obtained from the modified Colburn equation,

$$h \frac{De_1}{k_f} = 0.023 \left(\frac{De_1 V_1 \rho}{\mu} \right)_f^{0.8} \left(Cp \frac{\mu}{k} \right)_f^{0.3} \quad (2)$$

TABLE I

PERTINENT DATA ON THE UC₂-Al DISPERSION-TYPE MINIATURE FUEL PLATES

Capsule and Plate No.	Finished Core Length (in.)	Finished Core Width (in.)	Finished Core Area (cm) ²	Core Weight (g)	U- Content of Core (g)	U ²³⁵ * Content of Core (g)	U ²³⁵ Content per Lin- ear In. (g/in.)	U ²³⁵ Surface Density g/cm ²	Finished Plate Width (in.)	Finished Plate Length (in.)	Finished Plate Thickness (in.)
OR-35-9	4.875	0.703	22.11	5.786	3.087	0.617	0.127	0.028	0.9943	6.001	0.050
OR-35-10	4.844	0.703	21.97	5.803	3.096	0.619	0.128	0.028	0.9940	6.001	0.051
OR-35-11	4.906	0.703	22.25	5.669	3.025	0.605	0.123	0.027	0.9943	6.003	0.051
OR-35-12	4.844	0.703	21.97	5.839	3.116	0.623	0.129	0.028	0.9943	6.002	0.051
OR-35-13	4.844	0.703	21.97	5.810	3.100	0.620	0.128	0.028	0.9945	6.002	0.051
OR-35-14	4.875	0.703	22.11	5.839	3.116	0.623	0.128	0.028	0.9946	6.002	0.051
OR-35-15	4.812	0.703	21.82	5.786	3.087	0.617	0.128	0.028	0.9941	6.003	0.051
OR-35-16	<u>4.781</u>	0.703	21.68	5.761	3.074	<u>0.614</u>	<u>0.128</u>	0.028	0.9950	6.002	0.051
Average	4.848					0.617	0.127				

*Based on an enrichment of 19.99% in U-235 isotope.

in which all the water properties were evaluated at the film temperature. The symbols are defined as follows:

- k = thermal conductivity of the film water (Btu/hr-ft²°F/ft)
- ρ = density of coolant (lb/ft³)
- μ = viscosity of the film water (lb/sec-ft or lb/hr-ft)
- De_1 = equivalent dia of the inside channel (ft)
- C_p = specific heat of the film water (Btu/lb-F°)
- V_1 = velocity of coolant (ft/sec)

The equivalent diameter is defined as:

$$De_1 = \frac{4A}{P} \quad (3)$$

where:

- A = cross-sectional area of the channel
- P = total surface perimeter in the channel which is in contact with the coolant

From the sketch in ORNL-CF-58-2-109, the equivalent diameter of the channel can be determined as follows:

$$A_1 = \frac{\pi}{4} (1)^2 - 0.050(0.996)$$

$$A_1 = 0.0051 \text{ ft}^2$$

$$P_1 = \pi (1) + 2(0.996) + 2(0.050)$$

$$P_1 = 0.436 \text{ ft}$$

$$De_1 = \frac{4(0.0051)}{0.436} = 0.047 \text{ ft}$$

The quantity of heat generated in the miniature plate can be found using the following expression for Q:

$$Q = 0.45 \times 10^{-10} m\theta \quad (4)$$

where:

m = amount of U_{235} present (g)

θ = perturbed neutron flux (neutron/cm²-sec)

Assuming a perturbed flux of 8×10^{13} neutron/cm²-sec and a sample containing 0.617 g of U_{235} (the average loading for the miniature plates), a value for Q can be obtained:

$$\begin{aligned} Q &= 0.45 \times 10^{-10} m\theta \\ Q_{avg} &= 0.45 \times 10^{-10} (0.617)(8 \times 10^{13}) \\ Q_{avg} &= 2.22 \times 10^3 \text{ w or} \\ Q_{avg} &= 2.22 \times 10^3 \times 3.413 = 7.58 \times 10^3 \text{ Btu/hr} \end{aligned}$$

Assuming that the core is nominally 0.70-in. wide and 4.85-in. long, then the total surface area is

$$A_S = \frac{2(0.70)(4.85)}{144} = 0.047 \text{ ft}^2$$

Since the values for the various coolant properties which appear in equation (2) depend upon the film temperature defined in equation (1), a trial and error solution using these equations is required. Assuming various film temperatures as indicated in ORNL-CF-58-2-109 and consequently various values of μ , C_p , k , and ρ , equation (2) was solved for (h) and this value inserted into equation (1) to serve as a check on the assumed film temperature. It was found that a film temperature $T_f = 178^\circ\text{F}$ satisfied the solution. Evaluating the heat-transfer coefficient at this temperature yielded an $h = 2562 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$. This value was obtained using $Q_{avg} = 7.58 \times 10^3 \text{ Btu/hr}$. The check calculation is as follows:

for $T_1 = 178^\circ\text{F}$

$$h \left(\frac{0.047}{0.395} \right) = 0.023 \left(\frac{(0.047)(8.7)(61.4)}{2.4 \times 10^{-4}} \right)^{0.8} \left(\frac{(1.007)(0.85)}{0.395} \right)^{0.3}$$

$$h = 0.023(1.046 \times 10^5)^{0.8} (2.167)^{0.3}$$

$$h = 2562 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$$

then from equation (1),

$$\frac{Q}{A_S} = h(T_1 - T_2)$$

$$\frac{7.58 \times 10^3}{0.047} = (2562)(T_1 - 115)$$

$$T_1 = \frac{1.613 \times 10^5 + 115(2562)}{2562}$$

$$T_1 = 177.8^\circ\text{F}$$

B. Center-Line-Core Temperature

The core temperature at the center-line thickness can be estimated by first calculating the core-clad interface temperature. This calculation assumes heat flow only in the perpendicular direction to the plate surface from the core and can be found using the following expression.

$$\frac{Q}{A_S} = \frac{k}{L} (T_3 - T_1) \quad (5)$$

where:

k = conductivity of the cladding (118 Btu/hr-ft²-F°/ft)

L = the clad thickness (0.00125 ft)

T_3 = interface temperature (°F)

T_1 = plate-surface temperature (178°F)

$$161.3 \times 10^3 = \frac{118}{1.25 \times 10^{-3}} (T_3 - 178)$$

$$T_3 = \frac{161.3 \times 10^3 + 94.4 \times 10^3 (178)}{94.4 \times 10^3}$$

$$T_3 = 179.7^\circ\text{F}$$

Knowing the interface temperature and using similar assumptions, the center-line-core temperature can be calculated from the following expression:

$$(T_4 - T_3) = g \frac{\ell^2}{2k} \quad (6)$$

where:

T_4 = center-line-core temperature ($^\circ\text{F}$)

T_3 = interface core-clad temperature (179.7°F)

g = heat generation per unit volume ($\text{Btu}/\text{ft}^3\text{-hr}$)

ℓ = one-half the core thickness (0.000833 ft)

k = conductivity of the core ($\text{Btu}/\text{hr-ft}^2\text{-}^\circ\text{F}/\text{ft}$)

$$g = \frac{Q}{V}$$

$$Q = 7.58 \times 10^3 \text{ Btu/hr from equation (4)}$$

$$V = (\text{surface area of core}) \times (\text{core thickness})$$

$$V = \frac{0.70 \times 4.85 \times 0.020}{(12)^3} = 3.93 \times 10^{-5} \text{ ft}^3$$

$$g = \frac{7.28 \times 10^3}{2.93 \times 10^{-5}} = 1.93 \times 10^8 \text{ Btu}/\text{hr-ft}^3$$

The conductivity of the UC_2 -Al dispersion can be estimated as follows:²

$$k_{Al}(V\%_{Al}) + k_{UC_2}(V\%_{UC_2}) = k_{UC_2+Al}$$

$$(121)(0.7464) + 19.8(0.2536) = k_{UC_2+Al}$$

$$k_{UC_2+Al} = 95.34 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F/ft}$$

Therefore, using equation (6) the center-core temperature is:

$$T_4 = T_3 + g \frac{r^2}{2k}$$

$$T_4 = 179.7 + \frac{1.93 \times 10^8 (8.33 \times 10^{-4})^2}{2(95.34)}$$

$$T_4 = 180.4^\circ\text{F}$$

C. Additional Heat Data

Heat generation

The heat generation in the fuel sample appears on page 6 and can be calculated using equation (4). The value obtained for a fuel sample containing 0.617 g of U_{235} and in a perturbed flux of 8×10^{13} neutron/cm²-sec is

$$Q = 0.45 \times 10^{-10} \text{ m0}$$

$$Q = 2.22 \times 10^3 \text{ w}$$

Heat output in w/in.

The heat output in w/in. can be obtained as follows:

$$\frac{2.22 \times 10^3 \text{ (w)}}{4.85 \text{ (in.)}} = 4.58 \times 10^2 \text{ w/in.}$$

Thermal flux

The thermal flux can be calculated by the ratio $\frac{Q}{A_S}$ which appears in equation (1),

where:

$$\begin{aligned} Q &= \text{heat generation Btu/hr} \\ A_S &= \text{total surface area (ft)}^2 \end{aligned}$$

$$\frac{Q}{A_S} = \frac{(2.22 \times 10^3)(3.413)(144)}{2 \times 0.70 \times 4.85} = 1.613 \times 10^5 \text{ Btu/hr-ft}^2$$

IV. TIME OF IRRADIATION

The estimated nvt and time of irradiation to achieve 20, 40, 60, and 80% burn-up of U_{235} atoms is listed in Table II. These values are based on an assumed perturbed flux of 8×10^{13} neutron/cm²-sec. A sample calculation of the values indicated in Table II for the 20% burn-up specimens is as follows: the estimated nvt required for the 20% burn-up of U_{235} atoms in samples OR-35-9 and OR-35-10 (Table I) is given by:

$$\frac{dN}{dt} = -N\theta\alpha \quad (7)$$

where:

$$\begin{aligned} N &= \text{number of } U_{235} \text{ atoms/unit volume (cm}^{-3}\text{)} \\ \sigma &= \text{average absorption cross section of } U_{235} \text{ (cm}^2\text{)} \\ \theta &= \text{unperturbed flux (neutron/cm}^2\text{-sec)} \\ \alpha &= \text{reciprocal of the disadvantage factor (0.80)} \\ t &= \text{time (sec)} \end{aligned}$$

Solving equation (7),

$$N(t) = N_0 e^{-\alpha\theta\sigma t}$$

TABLE II

ESTIMATED NVT AND TIME OF IRRADIATION DATA

Sample No.	Perturbed Neutron Flux (Neutron/cm ² -sec)	Desired U ₂₃₅ atom burn-up	Estimated nvt	Estimated Exposure Time Days	Estimated Number of 15 Day Cycles
OR-35-9	8×10^{13}	20%	3.92×10^{20}	81	4
OR-35-10	8×10^{13}	20%	3.92×10^{20}	81	4
OR-35-11	8×10^{13}	40%	8.96×10^{20}	185	9
OR-35-12	8×10^{13}	40%	8.96×10^{20}	185	9
OR-35-13	8×10^{13}	60%	16.08×10^{20}	332	16
OR-35-14	8×10^{13}	60%	16.08×10^{20}	332	16
OR-35-15	8×10^{13}	80%	28.24×10^{20}	584	27
OR-35-16	8×10^{13}	80%	28.24×10^{20}	584	27

or burn-up,

$$1 - \frac{N(t)}{N_0} = 1 - \rho^{-\alpha\theta\sigma t}$$

Assuming an average reflector temperature of 115°F, the average absorption cross section for U_{235} is

$$\bar{\sigma}_a = 596 \times 10^{-24} \sqrt{\frac{293}{319}}$$

$$\bar{\sigma}_a = 570 \times 10^{-24} \text{ cm}^2$$

Then solving the equation for the total integrated flux, the required nvt for 20% burn-up is:

$$t = \frac{\ln N}{-\alpha\theta\sigma}$$

$$t = \frac{\ln(0.8)}{-(0.80)(1 \times 10^{14})(570 \times 10^{-24})}$$

$$t = \frac{0.223}{4.56 \times 10^{-8}} = 4.9 \times 10^6 \text{ sec}$$

$$\text{nvt} = (8 \times 10^{13})(4.9 \times 10^6) = 3.92 \times 10^{20}$$

Assuming 70% operation and a perturbed flux of 8×10^{13} neutron/cm²-sec in the reflector, the estimated time in days is:

$$\frac{3.92 \times 10^{20}}{(8 \times 10^{13})(3600)(24)(0.70)} = 81 \text{ days}$$

The number of 15 day irradiated cycles can be obtained as follows:

$$\frac{81 \times 0.70}{15} = 4 \text{ cycles}$$

V. HOT-SPOT HOT-CHANNEL TABULATION

The hot-spot hot-channel tabulation is shown in Table III for the miniature encapsulated fuel plates. This analysis yields a hot-channel factor $F_b = 1.20$ and a hot-spot factor (F_h) = 1.91. These values were obtained by merely multiplying together the individual variables listed in Table III.

TABLE III

HOT-SPOT AND HOT-CHANNEL FACTORS

	Hot-Channel Factor, F_b	Hot-Spot Factor, F_h	Additive Factors ΔT_{add}
A. Mechanical Tolerances (Hydraulic Effects)			
1. Effect on mass flow-External Effects (End boxes, etc.)	1.0	1.0	
2. Effect on mass flow-Entrance and Exit Effects	1.0	1.0	
3. Effect on mass flow-Channel Effects	1.0	1.0	
4. Effect of D Variation	1.0		
B. Mechanical Distortion			
1. Effect on mass flow-External Effects (End boxes, etc.)	1.0	1.0	
2. Effect on mass flow-Entrance Effects	1.0	1.0	
3. Effect on mass flow-Channel Effects	1.0	1.0	
4. Effect of D Variation	1.0	1.0	
C. Correlation Equation on h	1.0	1.25	
D. Determination of Critical Parameters			
1. Flow measurement	1.0	1.0	
2. Temperature measurements (metal)	1.0	1.0	
3. Pressure measurements	---	---	
4. Temperature measurements (coolant)	1.20	1.15	
5.			
E. Flow Distribution Effects			
1. Plenum Effects	1.0	1.0	
2. Edge Effects	1.0	1.0	

TABLE III (continued)

	Hot-Channel Factor, F_b	Hot-Spot Factor, F_h	Additive Factors ΔT_{add}
F. Neutron and Gamma Flux Distribution			
1. Vertical Distribution	1.0	1.0	
2. Horizontal Distribution	1.0	1.0	
3. Internal Distribution	1.0	1.0	
G. Neutron and Gamma Flux Uncertainty			
1. Unperturbed Flux Source	---	1.2	
2. Average Depression	---	1.0	
3. Internal Distribution	---	1.0	
4. Experimental Power Generation Variation Timewise	---	---	
H. Heat Flux Over Element Surface			
1. Fuel Homogeneity	---	1.06	
2. Mechanical Effects (Meat and Clad Thickness)	---	1.02	
I. Effects Peculiar to Experiment			
1. Variation in Sample Loading	---	1.02	
2.			
3.			
4.			
FACTORS	1.20	1.91	

REFERENCES

1. C. F. Leitten and W. C. Thurber, Phase I - Foreign Reactor Fuel Sample Irradiation, ORNL-CF-58-2-109 (Revised), (October 13, 1958).
2. "General Properties," Reactor Handbook, 3, Materials (Section 1) Technical Information Service, U. S. Atomic Energy Commission, (March 1955) p. 121.

DISTRIBUTION

1. R. E. Adams	17. J. B. Philipson, AEC, IDO
2. M. H. Bartz, PPCO	18. A. E. Richt, IDO
3. R. J. Beaver	19. H. M. Roth, AEC, OROO
4. D. S. Billington	20. M. J. Skinner
5. J. E. Cunningham	21. W. C. Thurber
6. R. Carson Dalzell, AEC, Washington	22. R. B. Van Sice, PPCO
7. R. L. Doan, PPCO	23. R. C. Waugh
8. J. H. Erwin	24. I. F. Zartman, AEC, Washington
9. J. I. Federer	25. M. C. Ziemke
10. J. H. Frye, Jr.	26. ORNL Doc. Reference Library
11. W. J. Kucera	27. Health Physics Library
12. C. F. Leitten, Jr.	28-29. Central Research Library
13. T. S. Lundy	30-39. Laboratory Records Dept.
14. W. D. Manly	40. Laboratory Records-ORNL-RC
15. J. H. McKenna, PPCO	41-55. TISE, AEC
16. F. L. McMillan, PPCO	