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SUBJECT: Phase III - Foreign Reactor - Fuel Sample Irradiation of U_{308} -Al
 Dispersions. Irradiation Request ORNL-MTR-35

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ABSTRACT

The following data highlight the results of calculations presented in this report to determine the irradiation testing for Phase III of Irradiation Request ORNL-MTR-35. This phase of the program entails exposure of U_{308} -Al dispersions in the beryllium reflector of the MTR.

Capsule surface temperature	170°F
Central-core temperature	172.8°F
Fissionable material (U_{235})	0.109 g/in.
Heat output	3.93×10^2 w/in.
Heat flux	1.35×10^5 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
Cycles or nvt desired	See Table II
Hot-spot factor F_h	1.91
Hot-channel factor F_b	1.20

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PHASE III - FOREIGN REACTOR - FUEL SAMPLE IRRADIATION OF U_3O_8 -Al DISPERSIONS
Irradiation Request ORNL-MTR-35

C. F. Leitten and W. J. Kucera

I. INTRODUCTION

This report describes Phase III of the Foreign Reactor Fuel Irradiation program described in ORNL-CF-58-2-109.¹ It consists in determining the integrity and suitability of a 35 vol % U_3O_8 dispersion in aluminum as a fuel for foreign reactor fuel elements.

II. DESIGN AND FABRICATION

This phase entails 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 fuel plates consist of a 20% enriched, 35 vol % dispersion of U_3O_8 in aluminum as the core material which has been hermetically sealed in 1100 aluminum by hot-rolling. The manufacturing stages for the miniature fuel plates are outlined below:

1. Miniature cores 0.570 in. x 0.423 in. x 0.324 in. were machined from a powder-metallurgy master core. The master core was prepared by cold-pressing a blended mixture of U_3O_8 and Al at 31 tsi and sintering in air at 400°C for 1 hr.
2. The cores were pressed into 1100 aluminum frames which were 0.309 in. in thickness and aluminum cover plates 0.100-in. thick were then attached by making fusion welds along both sides of the sandwich, employing no filler rod.
3. The billets were then hot-rolled at 590°C using approximately a 10% reduction in thickness per pass to a thickness of 0.055 in.
4. The hot-rolled plates were annealed for 1 hr at 600°C and then cold-rolled to 0.050 in.
5. After a stress-relief anneal at 600°C for 1 hr, the plates were marked, sheared and machined to 1 in. x 6 in. x 0.050 in.
6. Radiographs were then taken of each plate prior to insertion into the "leaky rabbit" capsules.

The critical plate dimensions and fuel loadings for the U_3O_8 -Al dispersion-type fuel plates are listed in Table I.

III. FUEL-PLATE TEMPERATURE CALCULATIONS

A. Surface-Plate Temperature

Previous analysis of the coolant-flow velocity for miniature "rabbit-type" capsules described in ORNL-CF-58-2-109 indicated a coolant velocity of 8.7 ft/sec. Thus employing this value and assuming an unperturbed flux 1×10^{14} neutron/cm²-sec, the fuel-plate surface temperature can be obtained from 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 core (ft²)
- 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,

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

with all water properties evaluated at the film temperature,

where:

- 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)

TABLE I

PERTINENT DATA ON THE U_3O_8 -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} Surface Density g/cm ²	U_{235} Content per Linear Inch g/in.	Finished Plate Length (in.)	Finished Plate Width (in.)	Finished Plate Thickness (in.)
OR-35-17	4.750	0.719	22.03	5.083	2.703	0.522	0.024	0.110	6.001	0.9960	0.050
OR-35-18	4.875	0.719	22.61	5.238	2.786	0.538	0.024	0.110	6.003	0.9986	0.050
OR-35-19	4.938	0.719	22.90	5.166	2.747	0.531	0.023	0.108	6.001	0.9984	0.050
OR-35-20	4.875	0.703	22.10	5.147	2.737	0.529	0.024	0.109	6.001	0.9980	0.050
OR-35-21	4.750	0.719	21.98	5.151	2.739	0.529	0.024	0.111	6.002	0.9972	0.050
OR-35-22	4.813	0.703	21.82	5.168	2.748	0.531	0.024	0.110	6.002	0.9971	0.050
OR-35-23	4.875	0.719	22.61	5.123	2.724	0.526	0.023	0.108	6.002	0.9968	0.050
OR-35-24	<u>4.813</u>	<u>0.719</u>	22.32	5.040	2.680	<u>0.518</u>	0.023	<u>0.108</u>	6.002	0.9967	0.050
	4.836	0.715				0.528		0.109			

De_1 = equivalent dia of the inside channel (ft)
(see ORNL-CF-58-2-109)

C_p = specific heat of the film water (Btu/lb-F°)

V_1 = velocity of coolant (ft/sec)

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

$$Q = 0.45 \times 10^{-10} m\theta \quad (3)$$

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 on a sample containing 0.528 g. of U_{235} , which is the average loading for the miniature plates, a value for Q can be obtained.

$$Q = 0.45 \times 10^{-10} m\theta$$

$$Q = (0.45 \times 10^{-10})(0.528)(8 \times 10^{13})$$

$$Q = 1.90 \times 10^3 \text{ w or}$$

$$Q = 1.90 \times 10^3 \times 3.413 = 6.49 \times 10^3 \text{ Btu/hr}$$

Assuming that the core is nominally 0.715-in. wide and 4.836-in. long, which is the average dimension for the test plates shown in Table I, the total surface area is,

$$A_s = \frac{2(0.715)(4.836)}{144} = 0.048 \text{ ft}^2$$

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 for the film temperature and heat-transfer coefficient 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 $T_1 = 170^\circ\text{F}$ satisfied the solution. Evaluating the heat-transfer coefficient at this temperature yielded $h = 2477 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$. The check calculation is as follows:

$$\text{for } T_1 = 170^\circ\text{F}$$

$$h \left(\frac{0.047}{0.392} \right) = 0.023 \left(\frac{(0.047)(61.4)(8.7)}{2.5 \times 10^{-4}} \right)^{0.8} \left(\frac{(0.9)(1.0065)}{0.392} \right)^{0.3}$$

$$h = 0.192(10.043 \times 10^5)^{0.8} (2.311)^{0.3}$$

$$h = 2477 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$$

then from equation (1),

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

$$\frac{6.488 \times 10^3}{0.048} = 2477(T_1 - 115)$$

$$T_1 = \frac{1.352 \times 10^5 + 115(2562)}{2562}$$

$$T_1 = 169.6^\circ\text{F}$$

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:

$$\frac{Q}{A_S} = \frac{k}{L} (T_3 - T_1) \quad (4)$$

where:

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

L = the clad thickness (0.00108 ft)

T_3 = interface temperature °F

T_1 = plate-surface temperature 170°F

$$1.352 \times 10^5 = \frac{118}{1.08 \times 10^{-3}} (T_3 - 170)$$

$$T_3 = \frac{1.352 \times 10^5 + (1.093 \times 10^5)(170)}{1.093 \times 10^5}$$

$$T_3 = 171.3^\circ\text{F}$$

Knowing the interface temperature and based on similiar assumptions, the center-line-core temperature can be calculated using the following expression:

$$(T_4 - T_3) = g \frac{\ell^2}{2k} \quad (5)$$

where:

T_4 = center-line-core temperature (°F)

T_3 = interface temperature (171.3°F)

g = heat generation per unit volume (Btu/hr-ft³)

ℓ = one-half the core thickness (0.0010 ft)

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

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

$$Q = 6.49 \times 10^3 \text{ Btu/hr}$$

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

$$V = \frac{(0.715)(4.836)(0.024)}{(12)^3} = 4.80 \times 10^{-5} \text{ ft}^3$$

$$g = \frac{6.49 \times 10^3}{4.80 \times 10^{-5}} = 1.35 \times 10^8 \text{ Btu/hr-ft}^3$$

The conductivity of the U_3O_8 -Al dispersion can be estimated as follows:

$$k_{\text{Al}}(\text{vol \% Al}) + k_{\text{U}_3\text{O}_8}(\text{vol \% U}_3\text{O}_8) = k_{\text{U}_3\text{O}_8 + \text{Al}}$$

Since no data were available for the conductivity of U_3O_8 , it is assumed to be similar to that of UO_2 thus:

$$(121)(0.352) + (2.52)(0.648) = k_{\text{U}_3\text{O}_8 + \text{Al}}$$

$$k_{\text{U}_3\text{O}_8 + \text{Al}} = 44.22 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F/ft}$$

Therefore, using equation (5), the center-core-temperature is:

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

$$T_4 = 171.3 + \frac{(1.35 \times 10^8)(1 \times 10^{-6})}{2(44.22)}$$

$$T_4 = 172.8^\circ\text{F}$$

C. Additional Heat Data

Heat generation

The heat generation in the fuel sample can be calculated using equation (3). The value obtained for a fuel sample containing 0.528 g of U_{235} and in a perturbed flux of 8×10^3 neutron/cm²-sec is:

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

$$Q = 1.90 \times 10^3 \text{ w}$$

Heat output in w/in.

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

$$\frac{1.90 \times 10^3}{4.836} = 3.93 \times 10^2 \text{ w/in.}$$

Thermal flux

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

where:

Q = heat generation Btu/hr

A_S = total surface area (ft)²

$$\frac{Q}{A_S} = \frac{6.488 \times 10^3}{0.048} = 1.35 \times 10^5 \text{ Btu/hr-ft}^2$$

IV. TIME OF IRRADIATION

The estimated nvt and time of irradiation to achieve 20, 40, 60, and 80% burn-up of the 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 can be obtained in ORNL-CF-58-2-109.

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 multiplying together the individual variables listed in Table III.

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 No. 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

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	
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	---	---	

TABLE III (continued)

	Hot-Channel Factor, F_b	Hot-Spot Factor, F_h	Additive Factors ΔT_{add}
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.	---	---	
FACTORS	1.20	1.91	

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1. C. F. Leitten and W. C. Thurber, Phase I - Foreign Reactor Fuel Sample Irradiation, ORNL-CF-58-2-109 (Revised) (October 13, 1958).

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