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OAK RIDGE, TENNESSEE

CF-56-6-127

DATE: June 20, 1956

SUBJECT: FUEL MELTING EXPERIMENT - IRRADIATION
REQUEST ORNL-MTR-28, PHASE A

FROM: F. H. Neill

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DATE FEB 22 1957 *kl*
For The Atomic Energy Commission

H. F. Canale
Chief, Declassification Branch

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FUEL MELTING EXPERIMENT

IRRADIATION REQUEST ORNL-MTR-28

PHASE A

I. Introduction

An experimental program has been undertaken at ORNL to obtain information on the magnitude of fission product activity released, in gaseous form, upon the melting of a stainless steel clad fuel element⁽¹⁾. In order to work out suitable experimental procedures and to obtain results of a preliminary nature, very small slivers of low burn-up fuel plate material have thus far been used.

To make a more accurate analysis, it is now necessary to use fuel plate samples that more nearly duplicate the APPR-1 fuel plate in composition, thickness, cladding, and burn-up. The following irradiation program was initiated to obtain such samples.

II. Design and Fabrication of the Fuel Samples

The miniature fuel plates to be irradiated consist of fully enriched UO_2 particles with small amounts of B_4C particles uniformly dispersed and embedded in a matrix of sintered stainless steel powder and clad with wrought 304L-grade stainless steel 0.005 in. thick. Each sample contains 100 mg of U-235 and is approximately $5/8$ " x $1-3/4$ " x 0.030". Other critical dimensions are indicated in Table I.

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1. ORNL CF 55-2-123, Leichsenring, G. F., Proposed Fuel Plate Melting Experiment.

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TABLE I

IRRADIATION SAMPLE DIMENSIONS

Overall length	1-3/4"
Overall width	5/8"
Overall thickness	0.030"
Fuel matrix length	1-1/4"
Fuel matrix width	1/4"
Fuel matrix thickness	0.020"

Sets of 4 samples are housed in a perforated aluminum container suitable for placing in the MTR reflector.

III. Time of Irradiation

The desired burn-up will vary with each group of samples and will be specified at the time each group of samples are delivered to the MTR.

The estimated nvt required for a given burn-up is determined as follows:

$$\frac{dN}{dt} = - \frac{N\phi \bar{\sigma}_a}{\alpha} \quad (1)$$

where N = number of U-235 atoms per unit volume (cm^{-3})

$\bar{\sigma}_a$ = average absorption cross section in a Maxwellian velocity distribution (cm^2)

$$\bar{\sigma}_a = \frac{\sigma_a^f}{1.128} \quad (2)$$

ϕ = unperturbed flux ($\text{neut}/\text{cm}^2\text{-sec}$)

α = disadvantage factor = 1.30

t = time (seconds)

f = correction for a non $-1/V$ dependence.

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Solving Equation (1)

$$N(t) = N(0) e^{-\frac{\bar{\sigma}_a \phi t}{1.128 \alpha}} \quad (3)$$

or

$$\text{burn-up} = \frac{N_0 - N(t)}{N(0)} = 1 - \frac{\bar{\sigma}_a \phi t}{\alpha}$$

Assuming an average reflector temperature of 150° F, the average absorption cross section for U-235 is:

$$\bar{\sigma}_a = \frac{\sigma_a^f}{1.128} = \frac{(660)(0.979)}{1.128} = 572.8 \text{ barns} \quad (4)$$

From Equation (3) the Mwd of reactor operation required for various burn-ups was calculated as a function of flux levels and is shown in Fig. 1.

IV. Thermodynamic Considerations

The power generation in the fuel samples is given by:

$$P = 0.45 \times 10^{-10} M_F \phi \quad (5)$$

where

P = total power per sample (watts)

M_F = mass of U-235 per sample (grams)

φ = thermal flux (neut/cm²-sec)

The fuel matrix has an elliptical shape in the fuel plates, the surface area of which is given by:

$$\begin{aligned} A_F &= \pi ab \\ &= (\pi)(1/4)(1-1/4) = 0.982 \text{ in}^2 \\ &= 0.00682 \text{ ft}^2 \end{aligned} \quad (6)$$

Substituting the various neutron fluxes listed above in Equation (5) results in the following thermal output per sample.

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TABLE II
POWER GENERATION AND HEAT FLUX PER SAMPLE

ϕ (Neut/cm ² -sec)	P(watts)	q/A(Btu/ft ² -hr)
0.5×10^{14}	225	56,295
1.0×10^{14}	450	112,590
1.5×10^{14}	675	168,885

A sketch of the aluminum carrier for the samples to be irradiated is shown in Fig. 2. It is assumed that the carrier will be placed in a standard "X" basket in the beryllium reflector of the MTR. Other assumptions are listed in Table III:

TABLE III
THERMAL MODEL DIMENSIONS

Inside diameter of "X" basket	$d_1 = 1.25$ in.
Outside diameter of sample container	$d_2 = 1.125$ in.
Inlet water temperature	$T_1 = 140^\circ$ F
Water velocity in annulus	$V = 30$ ft/sec
Overall fuel plate thickness	$t = 0.030$ in.
Fuel plate clad thickness	$t_c = 0.005$ in.
"X" basket active length	$L_x = 36$ in.
Sample container length	$L_s = 4$ in.
Number of samples per container	$N = 4$

The maximum coolant temperature, assuming a full "X" basket loading of samples and a uniform vertical flux distribution can be determined as follows:

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Total number of samples:

$$N_T = \frac{NL_x}{L_S} = \frac{(4)(36)}{4} = 36 \quad (7)$$

Total power output per sample (based on $\phi = 1.5 \times 10^{14}$; Table II):

$$q = (675 \text{ watts})(3.413 \text{ Btu/watt-hr}) = 2305 \text{ Btu/hr}$$

Weight rate of flow (w)

$$\begin{aligned} w &= V\rho A \\ &= (30 \text{ ft/sec})(61.1 \text{ \#/ft}^3) \left[(1.25)^2 - (1.125)^2 \right] \frac{\pi}{(4)(144)} \\ &= 2.94 \text{ \#/sec} \\ &= 10,570 \text{ \#/hr} \end{aligned} \quad (8)$$

Temperature rise in "X" basket channel:

$$\begin{aligned} \Delta T &= \frac{q}{C_p w} \\ &= \frac{(2305 \text{ Btu/hr})(36)}{(0.999 \text{ Btu/\#-}^\circ\text{F})(10,570 \text{ \#/hr})} = 7.9^\circ \text{ F} \end{aligned} \quad (9)$$

Hence the maximum channel coolant temperature is:

$$T_M = T_i + \Delta T = 140 + 7.9 = 147.9^\circ \text{ F} \quad (10)$$

With the assumptions stated above combined with the values given in Table III, it is possible to estimate the temperature at the centerline of the fuel sample as follows:

Cross sectional area of the flow channel in region of sample:

$$A_S = \frac{\pi d_1^2}{4} = \frac{\pi}{4} \frac{(1.25)^2}{144} = 0.00850 \text{ ft}^2 \quad (11)$$

Flow velocity in the region of the sample:

$$V_S = \frac{w}{A_S \rho} = \frac{(2.94 \text{ \#/sec})}{(0.0085 \text{ ft}^2)(61.1 \text{ \#/ft}^3)} = 5.76 \text{ ft/sec} \quad (12)$$

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The equivalent diameter:

$$D_e = \frac{4A_S}{(\text{per})} \frac{(4)(0.0085 \text{ ft}^2)}{(\pi)(\frac{1.25}{12}) + (4)(\frac{76}{360})(\pi)(\frac{0.25}{12})} = 0.0890 \text{ ft} \quad (13)$$

The coefficient of heat transfer:

$$\begin{aligned} h &= \frac{0.023 K}{D_e} \left[\frac{V D_e \rho}{\mu} \right]^{0.8} \left[Pr \right]^{0.4} \quad (14) \\ &= \frac{(0.023)(0.385)}{0.089} \left[\frac{(5.76 \times 3600)(0.089)(61.1)}{(0.970)} \right]^{0.8} \left[2.51 \right]^{0.4} \\ &= 1630 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F} \end{aligned}$$

The maximum centerline temperature of the fuel element is then

$$\begin{aligned} T_o &= \frac{q}{A} \left[\frac{1}{h} + \frac{L_c}{K_c} + \frac{L_M}{4K_M} \right] + T_M \\ &= (1.689 \times 10^5) \left[\frac{1}{1630} + \frac{(0.005)}{(12)(9.4)} + \frac{0.030}{4(9.4)(12)} \right] \\ &= 122.2 + 147.8 \\ &= 270.0^\circ \text{ F} \end{aligned}$$

FHNeill:vj
June 20, 1956

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Percent Burn-up of U-235

Burn-up of U-235

$\phi = 1.5 \times 10^{14}$
 $\phi = 1 \times 10^{14}$
 $\phi = 0.5 \times 10^{14}$

0	2000	4000	6000	8000	10,000	12,000	14,000	16,000	MWD
0	3	6	9	12	15	18	21	24	Cycles
0	2	4	6	8	10	12	14	16	Months

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Sect: A-A

NOTES

Material: 2S Aluminum
Item 3 Same as Item 1
Except grooves are on
both sides

SAMPLE CONTAINER 3
FHN 6-7-56

Item 1
Double Scale