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TITLE

GEH-4-63, 64; PROPOSAL FOR IRRADIATION
OF PRODUCTION BRAZED ZIRCONIUM CLAD
FUEL ELEMENTS

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GEH-4-63, 64; PROPOSAL FOR IRRADIATION OF
PRODUCTION BRAZED ZIRCALOY-2 CLAD FUEL ELEMENTS

by

J. C. Tverberg

Fuels Fabrication Development
Reactor & Fuels Research & Development
HANFORD LABORATORIES

May 18, 1961

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GEH-4-63, 65; PROPOSAL FOR IRRADIATION OF
PRODUCTION BRAZED ZIRCALOY-2 CLAD FUEL ELEMENTS

I. INTRODUCTION

A brazed end closure is currently being used on prototypical NPR fuel elements. The production closure will use a braze alloy composed of 5% Be + 95% Zry-2 to braze the Zircaloy-2 cap to the jacket and to the metallic uranium core. A similar MTR test, GEH-4-57, 58, used a braze alloy of the composition 4% Be + 12% Fe + 84% Zry-2 which melts at a lower temperature. In this previous test, element GEH-4-57 failed through a cladding defect located at the base of the braze heat affected zone. Because of this failure it would be desirable to subject a fuel element, which had been subjected to more severe brazing conditions, to the same conditions as GEH-4-57, 58. For this reason the thermal conditions of this test essentially match those of GEH-4-57, 58.

II. DESCRIPTION OF FUEL ELEMENTS

This irradiation test consists of two identical fuel elements. The fuel material is normal metallic uranium, Zircaloy-2 clad of the tubular geometry, NPR inner size. The fuel was coextruded at Hanford by General Electric's Fuels Preparation Department. Each element is 10.8 inches in length with flat Zircaloy-2 end caps brazed to the jacket and uranium core with the 5 Be + 95 Zry-2 brazing alloy, then TIG welded to further insure closure integrity. The elements are 1.254 inches OD and 0.439 inches ID. For hydraulic purposes a 0.343 inch diameter flow restrictor has been fitted into the central flow channel of both elements. A one-inch high perforated aluminum spacer supports the two elements. This entire assembly is contained in a standard GEH-4 basket which has had the fuel containing section slightly reduced in diameter to compensate for the smaller fuel element.

III. SUMMARY OF TEST CONDITIONS

Maximum Fuel Core Temperature	593 F (312 C)
Maximum Uranium Surface Temperature	379 F (193 C)
Maximum Jacket Temperature	217 F (103 C)
Normal Uranium Weight	0.750 lb/in
Total Uranium Weight (two elements)	15.75 lb (7,150 gm)
Weight U ²³⁵ (two elements)	0.11 lb (50 gm)
Maximum Power Generation	62.3 KW/ft
Maximum Heat Flux	502,600 BTU/hr-ft ²
Number of Fuel Elements	2
Number of MTR Cycles	6
Burnup Rate	13.8 MW/T
Water Flow Rate	15 gpm (Minimum 12 gpm)
Water Velocity	14.6 ft/sec

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Film Coefficient
External Fuel Diameter
Internal Fuel Diameter
Internal Basket Diameter of Fuel
Flow Restrictor Diameter

4,300 BTU/hr-ft²-°F
1.254 in.
0.439 in.
1.350 in.
0.343 in.

IV. FABRICATION

The fuel tubes were extruded by the conventional floating mandrel process at 1170 F and an extrusion ratio of 25:1. Extruded dimensions were 1.254 ± 0.005 inches OD and 0.439 ± 0.003 inches ID, with a nominal 0.035 inch thick outer Zircaloy-2 cladding and 0.025 inch thick inner Zircaloy-2 cladding. Transverse and longitudinal sections were examined for dimensions, uranium quality and bond quality. The extrusions, Nos. 189 and 190, were found suitable for irradiation. Extrusion copper was removed by nitric acid pickling. The extrusions were cut into 11.00 inch lengths and the uranium was removed to a depth of 0.300 inch on each end by etching in hydrochloric acid.

All fuel surfaces were then vapor blasted to remove any surface contamination, then bright etched in HF-HNO₃. The 0.250 inch thick Zircaloy cap was placed on the uranium and the walls crimped to hold the cap tightly against the uranium. A ring of braze material was placed on the cap and the assembly placed into the vacuum brazing chamber and evacuated to 10⁻³ mm of Hg. The closure was heated to 1040 C, and as soon as the braze metal melted, the vacuum valves were closed and the system pressurized with argon gas to force the braze alloy to flow under the cap.

After brazing the elements were beta heat treated in chloride salt at 730 C and quenched into nitrate salt. The elements were then machined to the step contour required for welding, vapor blasted to remove the surface scale and bright etched. The elements were then TIG welded, inspected and autoclaved for 72 hours in 400 C (750 F) 1500 psi steam to insure adequate closure and corrosion resistant welds as well as good quality Zircaloy-2 surfaces. Final inspection included a visual check for marred surfaces, pre-irradiation measurements, radiographs of the closure region and an ultrasonic bond check.

V. FLUX WIRES

One flux wire shall be attached to the basket and remain attached throughout the irradiation. Two flux wires, one toward the reactor core and one away from the core, shall be placed in the flux wire slots and shall be replaced at the start-up of each cycle. It is imperative that this be done so that a complete record is obtained of the flux for each cycle and for the entire irradiation. The values shall be forwarded to the experimenter as soon as they become available.

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VI. MEASUREMENTS

If possible, the fuel elements shall be measured for length, diameter and warp on the ETR basin measuring device. These measurements shall be taken both before and after irradiation.

VII. REQUIRED FUEL EXAMINATION

Periscope photographs shall be taken as soon as possible following discharge of the irradiated elements in the canal. Further examination will be done at HAPO.

VIII. SPECIMEN INSERTION AND REMOVAL

The fuel elements and an MTR basket will be shipped to the MTR/ETR site after approval of this irradiation proposal. The elements will be placed in the basket with the numbered end up and with GEH-4-63 as the upper element. Inlet and outlet thermocouples will be provided.

IX. HAZARDS

There are no foreseen hazards associated with the irradiation of these test elements because of the very low uranium temperature. The cladding, braze alloy and welds have been subjected to autoclave conditions much more severe than actual reactor operating conditions.

X. APPENDICES

Details of all pertinent information are appended in the following order:

Appendix I: Thermal Aspects
Appendix II: Figures

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APPENDIX I - THERMAL ASPECTS

A. For calculation purposes, the specific power generation was taken from data obtained with the nuclearly similar test GEH-4-57, 58. This test operated at a measured average power generation of 74 KW/ft or 7.25 KW/lb of natural uranium. These particular elements, GEH-4-63, 64, contain 0.750 lb of natural uranium per inch, therefore a power generation of 65.25 KW/ft may be expected. The maximum heat flux will be approximately 502,600 BTU/hr-ft².

B. Sample Calculations

Basic Conditions:

Fuel Element

Uranium Core

Length	10.50 inches
Outer Diameter	1.184 inches
Inner Diameter	0.489 inch

Element with Cladding

Length	10.90 inches
Outer Diameter	1.254 inches
Inner Diameter	0.439 inch
Cap Thickness	0.20 inch
- Outer Clad Thickness	0.035 inch
Inner Clad Thickness	0.025 inch

1. Flow Areas

$$\begin{aligned} \text{Basket} &= \pi/4 (1.350)^2 = 1.431 \text{ in}^2 \\ &= 0.00994 \text{ ft}^2 \\ \text{Outer Annulus} &= \pi/4 \left[(1.350)^2 - (1.254)^2 \right] = 0.196 \text{ in}^2 \\ &= 0.00136 \text{ ft}^2 \\ \text{Inner Annulus} &= \pi/4 \left[(0.439)^2 - (0.343)^2 \right] = 0.05898 \text{ in}^2 \\ &= 0.0004096 \text{ ft}^2 \end{aligned}$$

2. Peripheries

$$\begin{aligned} \text{Basket} &= \pi (1.350) = 4.241 \text{ in} \\ &= .353 \text{ ft} \\ \text{Fuel OD} &= \pi (1.254) = 3.940 \text{ in} \\ &= .328 \text{ ft} \\ \text{Fuel ID} &= \pi (0.439) = 1.379 \text{ in} \\ &= .115 \text{ ft} \end{aligned}$$

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$$\text{Flow Restrictor} = \pi (0.343) = \begin{matrix} 1.078 \text{ in} \\ 0.0898 \text{ ft} \end{matrix}$$

3. Power Generation

Specific power generation in GEH-4-57, 58 was measured at 7.25 KW/lb of natural uranium.

$$Q = \frac{7.875 \text{ lb Nat'l U/element}}{10.50 \text{ in/element}} \times \frac{7.25 \text{ KW}}{1 \text{ lb Nat'l U}} \times \frac{12 \text{ in}}{\text{ft}} = 65.25 \text{ KW/ft}$$

4. Heat Flux (Assumed to be equal for all surfaces)

$$H = \frac{65.25 \times 3412}{(.328 + .115) \times 1} = 502,600 \text{ BTU/hr-ft}^2$$

5. Required Film Coefficient (Assume the average film $t_f = 117 \text{ F}$)

$$h_f = \frac{H}{\Delta t} = \frac{502,600}{117} = 4,300 \text{ BTU/hr-F-ft}^2$$

6. Flow Rate Required to Maintain Assumed t_f

Outer Annulus

a. Equivalent Diameter

$$D_e = \frac{4A}{S} = \frac{4 (0.196)}{4.241 + 3.940} = 0.0958 \text{ inch}$$

b. Required Velocity

$$h_f = \frac{150 [1 + 0.011 (t_b)]}{D_e^{0.2}} V_o^{0.8} \quad (\text{McAdams, 3rd Edition, p. 288})$$

$$\begin{aligned} V_o^{0.8} &= \frac{h_f D_e^{0.2}}{150 [1 + 0.011 (t_b)]} = \frac{(4,300) (0.0958)^{0.2}}{150 [1 + 0.011 (100)]} \\ &= \frac{(4300) (0.625)}{315} = 8.53 \end{aligned}$$

$$V = 14.6 \text{ ft/sec}$$

c. Flow

$$\begin{aligned} Q &= VA \\ &= 14.6 \text{ ft/sec} \times 60 \text{ sec/min} \times 62 \text{ lb/ft}^3 \times 1/8.33 \text{ gal/lb} \times \\ &\quad 0.00136 \text{ ft}^2 \\ &= 8.86 \text{ gal/min} \end{aligned}$$

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Inner Annulus

a. Equivalent Diameter

$$D_e = \frac{4 (0.05898)}{1.379 + 1.078} = 0.0960 \text{ inch}$$

b. Required Velocity

$$V_1^{0.8} = \frac{(4300) (0.0960)^{0.2}}{150 [1 + 0.011 (100)]} = \frac{(4300) (0.625)}{315} = 8.53$$

$$V_1 = 14.6 \text{ ft/sec}$$

c. Flow

$$\begin{aligned} Q &= VA \\ &= (14.6) (0.00041) (446) \\ &= 2.67 \text{ gal/min} \end{aligned}$$

7. Pressure Balance

Pressure drop estimates were determined by the "velocity head" method, with reference made to the Reactor Handbook, Engineering Volume.

a. Annular Velocities

$$\begin{aligned} \text{Outer} &= 14.6 \text{ ft/sec} \\ \text{Inner} &= 14.6 \text{ ft/sec} \end{aligned}$$

b. Velocity Head = $v^2/2g$

$$\text{Outer Annulus} = \frac{(14.6)^2}{2 (32.2)} \times \frac{62}{144} = 1.425 \text{ psi}$$

$$\text{Inner Annulus} = (14.6)^2 \times 0.006686 = 1.425 \text{ psi}$$

c. Reynolds Number

$$N_R = \frac{D_e V \rho}{\mu}$$

N_R = Reynolds Number

D_e = Equivalent Diameter (ft)

V = Velocity (ft/sec)

μ = Viscosity (water at 100 F = 4.58
x 10⁻⁴ lb/sec-ft)

ρ = Density (water at 100 F = 62 lb/ft³)

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Outer Annulus

$$N_R = \frac{(0.0958) (14.6) (62.0)}{(12) (4.58 \times 10^{-4})}$$

$$= 1.578 \times 10^4$$

Inner Annulus

$$N_R = \frac{(0.0960) (14.6) (62.0)}{(12) (4.58 \times 10^{-4})}$$

$$= 1.581 \times 10^4$$

- d. Friction Factor "f" Read from Reactor Handbook, Engineering Volume, Figure 1.5.2

Outer Annulus

$$f = 0.0275$$

Inner Annulus

$$f = 0.0275$$

- e. Friction Loss

$$\Delta P = \frac{(f) (L) (V_h)}{D_e}$$

f = Friction Factor
L = Fuel Column Length (inches)
D_e = Equivalent Diameter (inches)
ΔP = Pressure Drop (psi)

Outer Annulus

$$\Delta P = \frac{(0.0275) (21.8) (1.425)}{(0.0958)} = 8.917 \text{ psi}$$

Inner Annulus

$$\Delta P = \frac{(0.0275) (21.8) (1.425)}{(0.0960)} = 8.899 \text{ psi}$$

- f. Contraction and Expansion Loss

$$\Delta P = K (V_h)$$

Coefficient "K" read from Reactor Handbook, Engineering Volume, figures 1.5.6 and 1.5.7.

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Outer Annulus

$$\begin{aligned} A_o &= \text{Annulus Area} = 0.196 \text{ in}^2 \\ A_b &= \text{Basket Area} = 1.431 \text{ in}^2 \end{aligned}$$

$$\frac{A_o}{A_b} = \frac{0.196}{1.431} = 0.137$$

$$K_c = 0.388$$

$$\begin{aligned} \Delta P &= (0.388) (1.425) \\ &= 0.553 \text{ psi contraction loss} \end{aligned}$$

$$K_e = 0.72$$

$$\begin{aligned} \Delta P &= (0.72) (1.425) \\ &= 1.026 \text{ psi expansion loss} \end{aligned}$$

Inner Annulus

$$\begin{aligned} A_1 &= \text{Annulus Area} = 0.05898 \text{ in}^2 \\ A_b &= \text{Basket Area} = 1.431 \text{ in}^2 \end{aligned}$$

$$\frac{A_1}{A_b} = \frac{0.05898}{1.431} = 0.0412$$

$$K_c = 0.398$$

$$\begin{aligned} \Delta P &= (0.398) (1.425) \\ &= 0.567 \text{ psi contraction loss} \end{aligned}$$

$$K_e = 0.94$$

$$\begin{aligned} \Delta P &= (0.94) (1.425) \\ &= 1.340 \text{ psi expansion loss} \end{aligned}$$

Total Pressure Drop

<u>Loss</u>	<u>Outer Annulus</u>	<u>Inner Annulus</u>
Contraction	0.553	0.567
Expansion	1.026	1.340
Friction	8.917	8.899
Total	10.496 psi	10.806 psi

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8. Adiabatic Radius

$$r_{\max T} = \sqrt{\frac{r_o^2 - r_1^2}{2 \ln \frac{r_o}{r_1}}}$$

r_o = radius to outside of uranium
cylinder = 0.592 inch
 r_1 = radius to inside of uranium
cylinder = 0.2445 inch

$$\begin{aligned} r_{\max T} &= \sqrt{\frac{(0.592)^2 - (0.2445)^2}{2 \ln \frac{0.592}{0.2445}}} = \sqrt{\frac{0.3505 - 0.0598}{2 \ln 2.421}} \\ &= \sqrt{\frac{0.2907}{1.76836}} = \sqrt{0.1644} = 0.406 \text{ inch} \end{aligned}$$

Heat Flow Ratio

$$\begin{aligned} \text{Area to inner annulus} &= A_1 = \pi (r_{\max T}^2 - r_1^2) \\ &= \pi [(0.406)^2 - (0.2445)^2] \\ &= \pi (.10459) \\ &= 0.3285 \text{ in}^2 \end{aligned}$$

$$\begin{aligned} \text{Area to outer annulus} &= A_o = \pi (r_o^2 - r_{\max T}^2) \\ &= \pi [(0.592)^2 - (0.406)^2] \\ &= \pi (0.1861) \\ &= 0.5846 \text{ in}^2 \end{aligned}$$

Heat Flow to outer surface

$$= \frac{0.5846}{0.3285 + 0.5846} \times 100 = 64.0\%$$

Heat Flow to inner surface

$$= \frac{0.3285}{0.3285 + 0.5846} \times 100 = 36.0\%$$

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9. Temperature Drop through Cladding

Outer Annulus

$$\Delta t_o = \frac{543.5 Q}{K} \ln \frac{r_2}{r_1} \quad \text{(Fuel Element Design Handbook, Page A-100-6)}$$

$$\begin{aligned} Q &= \text{Specific Power} = 65.25 \times 64.0\% = 41.8 \text{ KW/ft} \\ r_2 &= \text{Larger Radius} = 0.627 \text{ inch} \\ r_1 &= \text{Smaller Radius} = 0.592 \text{ inch} \\ K &= \text{Thermal Conductivity - Zircaloy-2 at 212 F} \\ &K = 8.12 \text{ BTU/hr-ft-}^\circ\text{F} \end{aligned}$$

$$\begin{aligned} \Delta t_o &= \frac{(543.5)(41.8)}{8.12} \ln \frac{0.627}{0.592} = 2797.82 \ln 1.059 \\ &= 160.0 \text{ F} = 88.9 \text{ C} \end{aligned}$$

Inner Annulus

$$\begin{aligned} Q &= 62.25 \times 36\% = 22.4 \text{ KW/ft} \\ r_2 &= 0.244 \\ r_1 &= 0.219 \end{aligned}$$

$$\begin{aligned} \Delta t_i &= \frac{(543.5)(22.4)}{8.12} \ln \frac{0.244}{0.219} = 1499.31 \ln 1.114 \\ &= 161.8 \text{ F} = 89.9 \text{ C} \end{aligned}$$

10. Temperature Drop through Zircaloy-2 - Uranium Interface

Assumed to have no Δt

11. Uranium Surface Temperature

Outer Surface

$$\begin{aligned} T_s &= t_{\text{bulk}} + \Delta t_f + \Delta t_o \\ &= 100 + 117 + 160 = 377 \text{ F} = 192 \text{ C} \end{aligned}$$

Inner Surface

$$\begin{aligned} T_s &= t_{\text{bulk}} + \Delta t_f + \Delta t_i \\ &= 100 + 117 + 162 = 379 \text{ F} = 193 \text{ C} \end{aligned}$$

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12. Temperature Drop through the Uranium

Temperature drop from adiabatic radius to outside of tubular core
(Reference: ANL-4690, page 9, Case III)

$$\Delta t (r_o - r_{max} T) = \frac{q}{2K} \left[\frac{(r_o^2 - r_{max}^2)}{2} - r_{max}^2 \ln \frac{r_o}{r_{max} T} \right]$$

where

$$q = \frac{q/ft}{Vol/ft} = \frac{(62.25) (3412) (4)}{(12) (1.184^2 - 0.489) (\pi)}$$

$$= 19,380 \text{ BTU/hr-in}^3$$

K = Thermal conductivity of uranium

$$= \frac{1196 + t (^{\circ}\text{C})}{83.4} = \frac{1196 + 200}{83.4} = 16.8 \text{ BTU/hr-}^{\circ}\text{F-ft}$$

$$= 1.4 \text{ BTU/hr-}^{\circ}\text{F-in}$$

$$\Delta t (r_o - r_{max}) = \frac{19380}{(2) (1.4)} \left[\frac{(0.592)^2 - (0.406)^2}{2} - (406)^2 \ln \frac{.592}{.406} \right]$$

$$= 6921.4 \left[\frac{0.1861}{2} - 0.1648 \ln 1.458 \right]$$

$$= 6921.4 (0.0930 - 0.0621)$$

$$= 214 \text{ F} = 119 \text{ C}$$

13. Temperature Summary

	<u>Outer Annulus</u>	<u>Inner Annulus</u>
Water	100 F (38 C)	100 F (38 C)
Clad Surface	217 F (103 C)	217 F (103 C)
Uranium Surface	377 F (192 C)	379 F (193 C)
Uranium Maximum	591 F (311 C)	593 F (312 C)

The flow rate required for a film drop of 117 F is only

$$8.86 + 2.67 = 11.53 \text{ gpm.}$$

Actual flow conditions should be 15 gpm. Therefore a velocity increase of $\frac{15}{11.53} = 1.30$ will exist. This will increase the h_f and thereby

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reduce the Δt_f to: $\frac{117}{(1.30)0.8} = \frac{117}{1.234} = 95$ F. The maximum surface temperature then will be 195 F instead of 217 F. There will be a corresponding decrease in internal temperatures also.

14. Basket Water ΔT

At 62.3 KW/ft power generation or 131 KW total power, 15 gpm water flow and 100 F inlet temperature, the water temperature rise over the elements should be 35°F.

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FIGURE 1

GEH-4-23, 64 Fuel Basket Assembly

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FIGURE 2
Cross Sectional View of GEH-4-83, 84 Fuel Element

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