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TITLE

PROPOSAL FOR IRRADIATING A TEST FUEL ELEMENT WITH FLUTED OUTER SURFACE: GEH 10-67 AND GEH 10-68.

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PROPOSAL FOR IRRADIATING A TEST FUEL ELEMENT WITH FLUTED
OUTER SURFACE
GEH 10-67 AND GEH 10-68

R. K. Marshall and D. P. O'Keefe

Fuel Design Operation
Hanford Laboratories Operation
General Electric Company
Richland, Washington

December 3, 1962

NOTICE

This test proposal is for the express purpose of providing information and data needed to obtain the required approvals for an experimental irradiation in the MTR or ETR. Subsequent to test approval, the decision to proceed with this irradiation test rests entirely with the experiment sponsor and is dependent upon program needs, internal scheduling priorities, and ETR or MTR scheduling requirements.

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PROPOSAL FOR IRRADIATING A TEST FUEL ELEMENT WITH FLUTED
OUTER SURFACE
GEH 10-67 AND GEH 10-68

I. PURPOSE

The fuel element for this test has been designed to extend the total exposure capability of metallic uranium fuel by altering the type of cladding deformations produced by uranium swelling. In a tubular fuel element of cylindrical geometry, irradiation-induced uranium swelling can cause splitting of the outer clad or buckling of the inner clad. With a fluted outer surface on a cylindrical element, a large amount of swelling may be accommodated by bending of the exterior cladding without the tensile strains which produce localized necking and failure. In addition, the fluted shape, by reducing the restraint on the fuel may direct the volume increases to the outer surface and decrease the likelihood of inner clad buckling.

For this test, the fluted surface concept has been adapted to a tubular element that nominally fulfills the requirements for N-reactor fuel. It is desired to observe the performance of an element with this geometry at an exposure that will produce about 5 percent swelling. It is requested, therefore, that GEH 10-67 and GEH 10-68 be irradiated to an exposure of about 3000 MWD/T in the 6X6 M-3 Loop of the ETR.

II. IRRADIATION UNIT CHARGES

Irradiation unit charges are to be made to Budget Activity Number 02-50-10-05-6.

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III. SUMMARY OF OPERATING CONDITIONS

TABLE I

Reactor and Loop Operating Conditions

Reactor Power Level	175 MW
Loop Flow Rate	175 gpm
Loop Pressure	2000 psi
Saturation Temperature	636 °F
Bulk Water ΔT (across fuel)	7 °F
Bulk Outlet Temperature	507 °F
Pressure Drop Across Assembly	9.0 psid

TABLE II

Experiment Operating Conditions

Coolant Velocity, Bore	22.0 ft/sec
Coolant Velocity, Annulus	17.1 ft/sec
Specific Heat Generation (peak)	126 kw/ft
Total Heat Generation	457,000 Btu/hr
Peak Heat Flux, Bore	395,000 Btu/hr-ft ²
Peak Heat Flux, Annulus	488,000 Btu/hr-ft ²
Water Film Heat Transfer Coefficient, Bore	7550 Btu/hr-ft ² °F
Water Film Heat Transfer Coefficient, Annulus	6800 Btu/hr-ft ² °F
Fuel Temperature °F	<u>Calculated</u> <u>Adjusted</u>
Clad Surface	
Bore	557 592
Annulus	581 627

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Table II (Cont'd.)

	<u>Calculated</u>	<u>Adjusted</u>
Core Surface		
Bore	680	715
Annulus	753	799
Maximum Core Temperature	942	988
Burnup Rate (<i>MAXIMUM</i>)	11.4 MW/T	

IV. DESCRIPTION AND FABRICATION OF FUEL SPECIMENS

There are two test fuel elements 10-1/2 inches long, internally-externally cooled, Zircaloy-2 clad and .947 percent enriched uranium cored. The assembly of the fuel element and its relation to the basket is shown in Figure 1. This element has the same uranium per unit length and has nominally the same cross-sectional geometry as that used in GEH 10-58, 59, and 60. In order to avoid the possibility of damaging amounts of warp, the length of the elements was kept as short as possible commensurate with specimen geometry requirements. A length of at least four times the diameter was desired to reduce the effects of end restraint on swelling deformations. A tapered joint between the cap and uranium was used to give a transition in clad bending strains between the cap and uranium.

The elements were formed by coextrusion of the .947 enriched uranium and the Zircaloy-2 cladding material. The fuel rods were cut from the central portion of the extrusion. The ends were recessed by chemical removal of the uranium to a depth of .470 inch. A 60° V-notch was machined in the uranium to a depth of .275 inch. The recess was then chemically etched for removal of another .020 inch of uranium. Zircaloy-2 caps were induction-brazed in place using a zirconium-5 percent beryllium brazing alloy. After brazing, the element ends were faced flat and a single weld pass was made

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over the cap-cladding joint by the TIG process. The Zircaloy-2 centering supports were spot-welded on the elements and sized to give a sliding fit between the elements and the basket.

The elements were ultrasonic tested for bonding of the clad and caps to the uranium, and clad thickness was measured on wafers from the extrusion adjacent to each end of the elements. The completed elements were autoclaved in 400°C steam for 72 hours.

V. SPECIAL MEASUREMENTS AND INSTRUCTIONS

It will be necessary to correlate volume increase with exposure. This may be done by determining the weight of the element in water after successive increments of exposure. The elements should be weighed in the ETR canal prior to charging in the reactor. Subsequent weight measurements should be made after exposure increments of about 500 MWD/T.

It would also be desirable to obtain length and maximum diameter measurements with equipment in the ETR canal before charging in order to establish comparative results in dimension changes. Diameter measurements must be related to the specific peaks across which the diameter is measured.

In addition to weight and dimensions, the usual data log information for exposure and fuel temperature determinations will be required.

VI. OPERATING CONDITION REQUIREMENTS

The important parameters in this experiment are exposure and core temperature. The loop temperature should not be lower than requested in Section III for more than brief intervals.

VII. DESCRIPTION AND ASSEMBLY OF BASKET

The basket assembly is shown in Figure 2. This equipment is nominally

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the same as that used in GEH 10-58, 59, 60. One difference is in the location of the hasp which connects the flow spacer to the upper assembly. This connection is above the fuel elements. The other difference is in fuel centering which is accomplished by the supports on the fuel exterior for radial centering, and with a Zircaloy-2 dummy beneath the elements for longitudinal centering.

VIII. INSERTION AND REMOVAL OF EXPERIMENT

A marker is located near the top end of each element to assist in orienting the element relative to the reactor centerline (Figure 1). Because of the asymmetrical radial flux distribution through the element, it will be necessary to rotate the elements 180° away from their previous position each time they are charged into the reactor. This will provide for a more uniform exposure throughout the uranium.

IX. HAZARDS

The hazard which may be reasonably anticipated in this test will be in the performance of the outer cladding at high exposures where the anticipated five percent swelling is approached. The design of the outer surface is based on a swelling capacity of 11 percent. This volume increase must occur before the cladding will be strained as it is in a cylindrical element. At the goal volume increase of 5 percent, no damaging deformations are expected in the cladding.

The short length of the test pieces should eliminate warp as a possible hazard.

X. DISPOSITION OF SPECIMENS

On completion of the irradiation, the fuel elements should be sent to R. E. Olson, Radiometallurgy Operation, 327 Building, 300 Area, General Electric Company, Richland, Washington. --

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APPENDIX - THERMAL-HYDRAULICS ANALYSIS

I. NOMENCLATURE AND UNITS

<u>Symbol</u>	<u>Definition</u>	<u>Units</u>
A	Flow area	ft ²
b	Factor from Ref. 1	--
C _p	Specific heat	Btu/lb°F
D _e	Equivalent diameter	ft
D _{e1}	Equivalent diameter	in
f	Friction factor	None
f _r	Radial peaking factor	None
f _x	Axial peaking factor	None
G	Volume flow rate	gal/min
g	Gravity acc.	ft/sec ²
h	Heat transfer coefficient	Btu/hr-ft ² °F
k	Thermal conductivity	Btu/hr-ft°F
K ₁	Contraction loss coefficient	None
K ₂	Expansion loss coefficient	None
L	Length	ft
NRe	Reynolds No.	None
ΔP	Pressure drop	psid
P	Power	kw
Q	Volume flow rate	ft ³ /sec
q	Heat generation rate	Btu/hr
q'	Specific heat generation rate	Btu/hr-ft
q''	Surface heat flux	Btu/hr-ft ²

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<u>Symbol</u>	<u>Definition</u>	<u>Units</u>
r	Radius	ft
t	Temperature	F
Δt	Temperature difference	F
V	Velocity	ft/sec
w	mass flow rate	lb/hr
Σ	Summation	--
ρ	Density	lb/ft ³
μ	Viscosity	lb/hr-ft

Subscripts

a	Annulus
b	Bore

II. DIMENSIONS

Basket, Fuel Elements, Flow Restrictor

Process Tube ID:	3.441 inches
Basket OD:	3.313 "
Basket ID:	2.700 "
Effective OD of clad fuel	2.327 "
ID of clad fuel	1.286 "
Diameter of flow restrictor	0.680 "
Basket length	50. "
Cladding thickness	0.025 "
Length of uranium/fuel element	9.6 "
Wt. of uranium/ft	22.11 lbs/ft
Enrichment	0.949

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III. POWER GENERATION

The power generation is based on GEH-10-58 which was, based on operating data, estimated to be 178 kw. The axial flux distribution is assumed to be flat over the length of the fuel.

Average Specific Power

$$(178) \frac{22.11}{39.6} \frac{.947}{1.12} = 84 \text{ kw/ft}$$

The .947 and 1.12 factors are the enrichment for this test and for GEH-10-58 respectively. The total weight of uranium in the GEH-10-58 was 39.6 lbs.

Maximum Specific Power

$$P_{\max} = P_{\text{ave}} f_r f_x$$

$$P_{\max} = (84)(1.5)(1) = 126 \text{ kw/ft}$$

Total Power

$$P = \frac{(84)(9.6)(2)}{12} = 134 \text{ kw}$$

IV. HYDRAULIC CHARACTERISTICS

Assumptions

Test section pressure: 2000 psi

Bulk inlet water temperature: 500 F

The pressure drops through each flow channel were calculated for three different arbitrarily selected flow rates using the following equation. The results are plotted in Figure 3.

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$$\Delta P = \frac{V^2 Z_0}{144(2g)} \quad \text{where } Z = \left[\frac{fL}{De} + \Sigma(K_1 + K_2) \right]$$

The factor $\Sigma(K_1 + K_2)$ is the total loss coefficient for all entrance and exit losses. The loss coefficients are taken from reference 2, pages 65 and 66.

$$f = \frac{(4)(.046)}{NR_e^{.2}} \quad \text{reference 3, page 155.}$$

Sample calculation of ΔP for fuel element bore.

Arbitrarily select a flow rate of 60 gpm.

$$V = \frac{Q}{A} = \frac{(60)(.002228)}{.0065} = 20.6 \text{ ft/sec}$$

$$f = .184 \left[\frac{\mu}{VDe\rho} \right]^{.2} = .0122$$

$$Z = \left[(.0122) \left(\frac{31.5}{.606} \right) + .38 + \left(\frac{.0065}{.0057} \right)^2 (.10 + .75) \right]$$

$$Z = .635 + .38 + 1.10$$

$$Z = 2.11$$

$$\Delta P = \frac{(20.6)^2 (2.11) (46.5)}{(144)(2)(32.2)} = 4.48 \text{ psid}$$

The annulus curve is based on the assumption that the flutes on one fuel element are lined up with the troughs on the adjacent fuel element.

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The pressure drop vs flow across the orifice was calculated from the equation:

$$G = 236 d_o^2 C \sqrt{\frac{\Delta P}{\rho}} \quad \text{where } d_o = \text{orifice diameter and } C \text{ is the orifice coefficient.}$$

The equation was taken from reference 4, page 3-14.

See Figure 3 for plot of hydraulic characteristics.

Fuel element flow split

The flow split can be determined from the Bore and Fuel Element Annulus curves of Figure 3.

Example:

At a ΔP across the fuel column of 5 psid the flow through the bore would be 64 gpm, the flow through the annulus 78 gpm.

$$\% \text{ through bore} = \frac{(64)(100)}{142} = 45.0\%$$

$$\% \text{ through annulus} = \frac{(78)(100)}{142} = 55.0\%$$

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V. HOT SPOT AND HOT CHANNEL FACTORS

	<u>Hot Channel Factor, Fb</u>	<u>Hot Spot Factor, Fh</u>	<u>Additive Factors ΔT Add</u>
A. Mechanical Tolerances (Hydraulic Effects)			
(1) Effect on Mass Flow- External Effects	1	1	
(2) Effect on Mass Flow- Entrance and Exit Effects	1	1	
(3) Effect on mass flow- Channel Effects	1	1	
(4) Effect on D Variation	1	1.01	
B. Mechanical Distortion			
(1) Effect on Mass Flow- External Effects	1	1	
(2) Effect on Mass Flow- Entrance Effects	1	1	
(3) Effect on Mass Flow Channel Effects	1	1	
(4) Effect on D Variation	1.03	1.03	
C. Correlation Equation on h	1	1.20	
D. Determination of Critical Parameters			
1. Flow Measurement			
2. Temperature Measurements (metal)	1.03	1.03	2
3. Pressure measurements			
4. Temperature measurements (coolant)	1	1	4

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V. HOT SPOT AND HOT CHANNEL FACTORS (Cont'd.)

	<u>Hot Channel Factor, F_b</u>	<u>Hot Spot Factor, F_h</u>	<u>Additive Factors ΔT Add</u>
E. Flow Distribution Effects			
1. Plenum Effects	1	1	
2. Edge Effects	1	1	
F. Neutron and Gamma Flux Distribution			
1. Vertical Distribution	1	1	
2. Horizontal Distribution	1	1	
3. Internal Distribution	1.05	1.05	
G. Neutron and Gamma Flux Uncertainty			
1. Unperturbed Flux Source	1	1	
2. Average Depression	1.01	1.01	
3. Internal Distribution	1	1	
4. Experimental Power Generation Variation Timewise	1.1	1.1	
H. Heat Flux Over Element Surface			
1. Fuel Homogeneity	1	1.02	
2. Mechanical Effects (Mech and Clad Thickness)	1	1	
I. Effects Peculiar to Experiment			
	<u>1</u> 1.24	<u>1</u> 1.53	<u>6</u>

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VI. THERMAL CHARACTERISTICS

Total loop flow:	175 gpm
Fuel element bore	64 gpm
Fuel element annulus:	78 gpm
Basket-loop tube annulus	33 gpm
Test section ΔP @ 175 gpm from Figure 3	9.0 psid

Water film heat transfer coefficient

$$h = \frac{b \cdot v^{.8}}{De_1^{.2}} \quad \text{Reference 1, page 52.}$$

1. Bore

$$h_b = \frac{(575)}{(.606)^{.2}} \left[\frac{(64)(.002228)}{(.0065)} \right]^{.8}$$

50 (175)^{.8} = 9.7

$$h_b = 7550 \text{ Btu/hr-ft}^2\text{ }^{\circ}\text{F}$$

6200 @ 50 gpm the bore

2. Annulus

$$h_a = \frac{(575)}{(.373)^{.2}} \left[\frac{(78)(.002228)}{.01015} \right]^{.8}$$

$$h_a = 6800 \text{ Btu/hr-ft}^2\text{ }^{\circ}\text{F}$$

Heat Split

An IBM 7090 computer program was used to calculate the heat split.

Fraction to bore - 0.309

Fraction to annulus - 0.691

Maximum Surface Heat Flux

$$1. \text{ Bore: } \frac{(126)(3413)(.309)(12)}{\pi(1.286)} = 395,000 \text{ Btu/hr-ft}^2$$

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2. Annulus: $\frac{(126)(3413)(.691)(12)}{\pi(2.327)} = 488,000 \text{ Btu/hr-ft}^2$

Temperature Drop

	<u>Calculated</u>	<u>Adjusted</u>
<u>Bore</u>		
Fuel ΔT (from IBM Program)	261	
Bond $\Delta T = \frac{(126)(3413)(.309)(12)}{(\pi)(1.336)(30,000)} =$	13.	
Cladding $\Delta T = \frac{(126)(3413)(.309) \ln \frac{1.336}{1.286}}{2\pi(7.5)}$	110.	
Water film $\Delta T = \frac{(126)(3413)(.309)(12)}{\pi(1.286)(7550)}$	52 ⁶³	80
Bulk water temperature rise $= \frac{(134)(3413)(.309)}{(64)(8.02)(46.5)(1.252)}$	5.	12.

Annulus

Fuel ΔT (from IBM Program)	189.	
Bond $\Delta T = \frac{(126)(3413)(.691)(12)}{\pi(2.277)(30,000)}$	17.	
Cladding $\Delta T = \frac{(126)(3413)(.691) \ln \frac{2.327}{2.277}}{2\pi(7.5)}$	155.	
Water film $\Delta T = \frac{(126)(3413)(.691)(12)}{\pi(2.327)(6800)}$	72.	110.
Bulk water temperature rise $= \frac{(134)(3413)(.691)}{(78)(8.02)(46.5)(1.252)}$	9.	17.

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	<u>Temperatures °F</u>			
	<u>Bore</u>		<u>Annulus</u>	
	<u>Calc.</u>	<u>Adj.</u>	<u>Calc.</u>	<u>Adjusted</u>
Inlet water temperature	500	-	500	-
Cladding temperature	557	592	581	627
Maximum fuel surface temperature	680	715	753	799
Maximum fuel temperature	941	976	942	988
Sat. temp @2000 psig	636	-	636	-

Since the axial flux distribution was assumed to be flat, the temperatures were based on conditions at the end of the fuel column.

Test ΔT

$$\frac{(134)(3413)}{(142)(8.02)(46.5)(1.252)} = 7.0^{\circ}\text{F}$$

Loop ΔT

$$\frac{(134)(3413)}{(175)(8.02)(46.5)(1.252)} = 6.0^{\circ}\text{F}$$

$$P = \frac{\Delta T (175)(8.02)(46.5)(1.17)}{3413}$$

REFERENCES

1. Nertney, R. J., Editor, Calculated Surface Temperatures for Nuclear Systems and Analysis of Their Uncertainties, IDO-16343, June 1, 1957, Unclassified.
2. The Reactor Handbook, Vol. 2, Engineering, USAEC, May 1955, AECD-3646.
3. McAdams, W. H., Heat Transmission, 34d Edition.
4. Flow of Fluids Through Valves, Fittings, and Pipe, Crane Technical Paper No. 410.

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SEC. A-A

FIGURE 1

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Figure 2

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2 X 2 CYCLES

Figure 3

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