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HW-72507 RD

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January 29, 1962

GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

TITLE

PROPOSAL FOR CHARGING THE EIGHTH FUEL
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D. C. Kaulitz and
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PROPOSAL FOR CHARGING THE EIGHTH FUEL RUPTURE EXPERIMENT
GEH 10-57

D. C. Kaulitz and R. L. Call

January 29, 1962

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A

PROPOSAL FOR CHARGING THE EIGHTH FUEL RUPTURE EXPERIMENT
GEH 10-57

D. C. Kaulitz and R. L. Call

January 29, 1962

I. OBJECTIVE

The objective of this test is to prove a new method of intentional rupturing of fuel elements under conditions of high specific powers and central core temperatures. The fuel will be the inner component of an NPR fuel element assembly.

Permission to charge one tubular fuel element (GEH-10-57) into the GEH-P7 loop is requested. A hydraulic cylinder will be attached to the fuel basket assembly. The hydraulic cylinder will be used to move a cutter along the outer surface of the fuel element. The cutter is designed to machine through the Zircaloy cladding and expose the uranium core to the loop coolant.

II. IRRADIATION UNIT CHARGES

The irradiation unit charges for GEH-10-57 are to be made to the following Budget Activity Number:

02 - 50 - 10 - 05 - 6

III. SUMMARY OF TEST CONDITIONS

TABLE I

Reactor and Loop Operating Conditions

Reactor power level, MW	175
Desired flow rate	71.5 gpm (Figure 1)
Loop pressure, psia	2000
Saturation temperature of loop water, °F	635.8
Coolant velocity past fuel	See Figures 2 and 3
Bulk inlet and outlet temperatures	See Figure 4
Pressure drop across the fuel	~30 psi

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TABLE II

Experiment Operating Conditions

Maximum specific power generation	81.6 kw/ft (Figure 5)
Flow in the various channels	See Figures 6 and 7
Maximum surface heat flux	526,000 BTU/hr-ft ² (Figure 8)
Film coefficient	See Figures 9 and 10
Total heat generation	98.2 kw (Figure 11)
Maximum cladding surface temperature	587.5 °F (Figure 12)
Maximum uranium surface temperature	831.3 °F (Figure 13)
Maximum core center temperature	1062.6 °F (Figure 14)
Burnup rate	20.8 MWD/T (max.) (Figure 15)
Desired exposure	< 1 cycle

TABLE III

Fuel Element Information

Number of fuel elements	1
Cladding material	Zircaloy-2
Fuel material	0.947% enriched uranium
Weight of uranium, g/in	276
Total weight of uranium, g	5290
Weight of cladding, g	445
Weight of basket (in active zone) g	687
Fuel element dimensions	
Overall length, in	23.13
Uranium length, in	22.73
Cladding OD, in	1.253
Cladding ID, in	0.439
Uranium OD, in	1.173
Uranium ID, in	0.489
Cladding thickness	
Outer, in	0.040
Inner, in	0.025

The figures referenced above give values for the parameters when power generation is different from that used in the calculations.

IV. DESCRIPTION OF FUEL ELEMENT

One fuel tube, the inner component of an NPR fuel assembly, is proposed for this test. The element will be GEH-10-57. The fuel element was fabricated from 0.947 percent enriched uranium coextruded in Zircaloy-2 cladding. The cladding is 0.040 inches thick on the outer surface of

[REDACTED]

the tube and 0.025 inches thick on the inner surface. The fuel tube is 2313 inches long overall, and the uranium length is 22.73 inches. The 1/4-inch thick Zircaloy-2 end caps were brazed and welded into the counterbored ends of the fuel tube. The fuel tube has supports spot-welded to the outer surface to center the tube in the basket.

Figure 16 is a cross-section drawing of the fuel tube in the basket with the hydraulic cylinder that will be used to cut the fuel cladding.

V. DESCRIPTION OF RUPTURING MECHANISM

Details of the rupture device are shown in Figure 16 and 17 and the controls on Figure 18. The hydraulic milling machine consists of two cylinders. The large piston is used to move the cutter along the outer surface of the fuel element and the small piston moves the cutter in and out from the fuel surface. As external hydraulic pressure is applied on the main piston it is driven down; during this cycle the small piston not being powered causes a drag and pulls the cutter free from the surface of the fuel element. After completing the down stroke, cylinder pressure is reduced suddenly to atmospheric pressure and the loop system pressure drives the piston up on the cutting stroke. During this stroke, the water in the small cylinder is forced to exit through a very small orifice, building up a large pressure and causing a downward force on the small piston.

This force drives the cutting tool into the cladding during the cutting stroke.

This assembly has been tried out in a pressurized tube to determine its feasibility. These ex-reactor tests indicate that it should operate under conditions of pressure, temperature and reactor power.

[REDACTED]

VI. DESCRIPTION OF FUEL BASKET

A double-wall stainless steel basket is used for this experiment. The overall length is 52-1/2 inches. The inner tube is 1.779 inches I.D. with a .035-inch thick wall. The outer tube has an I.D. of 2.000 inches and a wall thickness of 1/32-inch. The double wall basket has previously been used to obtain proper split in flow. The basket has been orificed so approximately 15 per cent of total loop flow will flow through the annulus between the basket and process tube to cool the two components. The rest of the flow will pass through the basket to cool the fuel element.

There will be no thermocouples installed for this test, but it will require a 1/4-inch stainless steel tube to actuate the hydraulic cylinder.

The 1/4-inch tube will be fed out the cap of the GEH-P7 pressure tube and the seal at the cap will be made with a 1/4-inch Conax seal. A 1/4-inch Conax seal will also be furnished to seal the tubing as it exits from the reactor pressure vessel.

VII. HAZARDS

Past rupture tests performed in the ETR on irradiated and unirradiated fuel elements have shown that corrosion rates are slow enough that neither the reactor nor the GEH-P7 facility will be in any danger as a result of this test. Loop instrumentation is adequate to detect any potentially serious condition in time to shut down the reactor before damage occurs. Burnout calculations for steady-state conditions have been made and the burnout fluxes are greater than the maximum surface heat fluxes by at least a factor of 7 over the range of fuel element powers and coolant inlet temperatures considered.

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During this test, the activity monitor should not be connected to the reactor scram circuit. The activity level scram will be manually controlled. Provisions are made to record coolant activity, temperature, flow and pressure in the loop. Should any of these parameters approach conditions outside the loop operating limits, the reactor will be scrammed before any damage to the reactor or loop components could occur.

The intentional rupture of the unirradiated tubular fuel element will not be hazardous to either the reactor or the loop facility.

VIII. PROCEDURES FOR ASSEMBLING FUEL ELEMENTS AND BASKET

The basket and hydraulic milling machine will be received at the ETR assembled. The fuel will be shipped in a separate container. It will be necessary to insert the fuel and make a circumferential weld on the basket. Personnel from Hanford will be present during this assembly to offer their services if needed.

IX. INSERTION OF THE TEST ASSEMBLY

After the reactor water level has been lowered and the GEH-P7 process tube cap has been removed, the process tube is ready for charging. Because the fuel to be loaded is unirradiated, it can be lowered in the pressure tube from the top of the reactor. The 1/4-inch line that feeds the hydraulic cylinder will be just long enough to clear the cap when it is in position. It will be necessary to use a 1/4-inch Swagelok coupling to attach additional 1/4-inch tubing to feed out of the reactor tank. This hydraulic line is to be connected to the hydraulic pump, as shown in Figure 18.

X. TEST PROCEDURE

As soon as the reactor has reached full power, the loop outlet tempera-

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ture is to be adjusted to the value determined from Figure 4. As soon as the reactor and loop have reached the operating conditions described in this report, the fuel is to be ruptured. The procedure below will be performed 15 times unless the D.W. system indicates failure has occurred with fewer times:

- 1) Check that all lines are filled with water.
- 2) Check to see that Valves 1, 2, and 3, as shown in Figure 18, are closed.
- 3) Open Valves 1 and 2 and operate pump rapidly until pressure builds up to about 4000 lbs.
- 4) Open Valve 3 and vent rapidly.
- 5) Close Valve 3 and repeat operation.

When the activity monitor indicates that the cladding failure has occurred, the reactor will be scrammed and the loop water temperature reduced. If there is no immediate indication of cladding failure, we request that the element be operated for a period of at least 24 hours.

XI. REMOVAL OF TEST ASSEMBLY

After the reactor top shielding has been removed and reactor water level lowered, the 1/4-inch hydraulic line is to be cut or separated just above the process cap. The 1/4-inch tubing line and test assembly are then to be removed as a unit from the pressure tube. If failure of the fuel is severe, an aluminum shipping can will be used in the reactor vessel. The 1/4-inch line will be cut just above the lifting bail and the fuel assembly is to be loaded into the shipping container for discharge from the reactor.

APPENDIX
SAMPLE HEAT TRANSFER CALCULATIONS

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I. NOMENCLATURE

- A - area (ft^2)
- c_p - specific heat at constant pressure ($\text{BTU}/\text{lb}\cdot^\circ\text{F}$)
- D_e - equivalent diameter (ft)
- D_o - outer diameter of either the outer or inner cladding (in)
- D_i - inner diameter of either the outer or inner cladding (in)
- e - the base of natural logarithms
- E - surface heat flux ($\text{BTU}/\text{hr}\cdot\text{ft}^2$)
- F - volume flow (ft^3/sec)
- F_b - hot channel factor
- F_h - hot spot factor
- G - mass velocity ($\text{lb}/\text{hr}\cdot\text{ft}^2$)
- h - heat transfer coefficient ($\text{BTU}/\text{hr}\cdot\text{ft}^2\cdot^\circ\text{F}$)
- k - thermal conductivity ($\text{BTU}/\text{hr}\cdot\text{ft}\cdot^\circ\text{F}$)
- Q - total power generation (kw)
- q - specific power (kw/ft or kw/in^3)
- r - radius (in)
- T - temperature ($^\circ\text{F}$)
- v - velocity (ft/sec)
- w - mass flow rate (lb/hr)
- μ - viscosity (lb/hr)
- ρ - density (lb/ft^3)

Subscripts following any of the above symbols will indicate the following:

- a - Pertains to the annulus between the basket and the fuel element.
- b - Pertains to the bore of the fuel element.

(Continued.)

- d - Pertains to the point at which the hole is to be cut into the fuel element.
- o - Pertains to flow etc., outside the basket.
- p - Peak, or maximum value.
- t - total value.

II. HOT SPOT AND HOT CHANNEL FACTORS⁽¹⁾

A. Mechanical Tolerances	<u>F_h</u>	<u>F_b</u>	<u>ΔT Add</u>
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 of D variation	1	1.04	
B. Mechanical Distortion			
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 of D variation	1.02	1.02	
C. Correlation Equation on h	1	1.2	
D. Determination of Critical Parameters			
1. Flow measurement	1.03	1.03	1.4
2. Temperature measurement			
3. Pressure measurement			
4. Temperature measurements (coolant)			4.3
E. Flow Distribution Effects			
1. Plenum effects	1	1	
2. Edge effects	1	1	
F. Neutron and Gamma Flux Uncertainty			
1. Vertical distribution	1	1	
2. Horizontal distribution	1	1	
3. Internal distribution	1	1	
G. Neutron and Gamma Flux Undertainty			
1. Unperturbed flux source	1	1	
2. Average depression	1	1	
3. Internal distribution	1	1	
4. Experimental power generation variation timewise	1.1	1.1	

(Cont'd.)

(1) Nertney, R. J., Calculated Surface Temperatures for Nuclear Systems and Analysis of Their Uncertainties, IDO-16343, June, 1957, p. 15 and ff.

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	<u>F_h</u>	<u>F_b</u>	<u>ΔT Add</u>
H. Heat Flux Over Element Surface			
1. Fuel homogeneity	1	1.02	
2. Mechanical effects	1	1	
I. Effects Peculiar to Experiment	<u>1</u>	<u>1</u>	<u> </u>
Products of F _b and F _h columns and sum of ΔT add column	1.16	1.47	5.7

III. POWER GENERATION

The total power generated by GEH-10-44, 45 in the P7 Loop was 102 kw. (2)

The total gamma heat from the process tube in the active zone, from the basket, fuel element cladding, and the uranium was calculated to be 19 kw, leaving a total fission heat of 83 kw. For GEH-10-44, 45 the average specific heat was then calculated to be 34.5 kw/ft.

Fuel elements GEH-10-44, 45 are both natural uranium, NPR inner tube fuel elements. The fuel element for GEH-10-57 is an NPR inner tube also, but is 0.947 percent enriched. Let us assume that the ratio of power generation in the two tests will be the ratio of the enrichments with a five percent reduction in power because of the additional blackness of the enriched fuel element. Then, for this test

$$q(\text{ave}) = 34.5 \frac{\text{kw}}{\text{ft}} \times \frac{0.947}{0.72} \times 0.95 = 43.2 \frac{\text{kw}}{\text{ft}} .$$

For GEH-10-57 the mass of the basket in the active zone is:

$$\frac{\pi}{4} [(1.820)^2 - (1.758)^2] \times 36 \text{ in}^3 \times 0.237 \frac{\text{lb}}{\text{in}^3} \times \frac{1000\text{g}}{2.2 \text{ lb}} = 687\text{g} .$$

Let us assume that the average value of gamma heating in the loop is 1.1 watt/g. Then the gamma heat from the basket will be

$$287\text{g} \times 1.1 \text{ watt/g} = 0.8 \text{ kw.}$$

(2) Personal Communication from R. S. Kemper.

The mass of the cladding is

$$\frac{\pi}{4} [(1.253)^2 - (1.173)^2 - (.489)^2 - (.439)^2] \times 22.73 \text{ in}^3$$

$$\times 0.237 \frac{\text{lb}}{\text{in}^3} \times \frac{1000\text{g}}{2.2\text{lb}} = 445\text{g}$$

$$\text{Gamma heat} = 445\text{g} \times 1.1 \text{ watt/g} = 0.5 \text{ kw.}$$

The gamma heat from the end caps will be less than 0.1 kw so we will neglect that source of heat.

The mass of the uranium is

$$\frac{\pi}{4} [(1.173)^2 - (.489)^2] \times 1 \frac{\text{in}^3}{\text{in}} \times \frac{.061\text{in}^3}{\text{cm}^3} \times \frac{18.9\text{g}}{\text{cm}^3} =$$

$$276 \frac{\text{g}}{\text{in}} \times 22.73 \text{ in} = 6290\text{g}$$

$$\text{Gamma heat} = 6290\text{g} \times 1.1 \text{ watt/g} = 6.9 \text{ kw.}$$

The mass of the process tube in the active zone is:

$$\frac{\pi}{4} [(2.576)^2 - (2.156)^2] \times 36 \text{ in}^3 \times 0.29 \frac{\text{lb}}{\text{in}^3} \times \frac{1000\text{g}}{2.2\text{lb}} = 7420\text{g}$$

$$\text{Gamma heat} = 7420\text{g} \times 1.1 \text{ watt/g} = 8.2 \text{ kw}$$

$$\text{Total fission heat} = 43.2 \frac{\text{kw}}{\text{ft}} \times \frac{22.73}{12} \text{ ft} = 81.8 \text{ kw}$$

Gamma heat from uranium	6.9 kw
-------------------------	--------

Gamma heat from cladding	0.5 kw
--------------------------	--------

Total heat from fuel (Q) =	89.2 kw
----------------------------	---------

Average specific heat (fission plus gamma) is:

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$$q(\text{ave}) = 89.2 \text{ kw} \times \frac{12}{22.73 \text{ ft}} = 47.1 \frac{\text{kw}}{\text{ft}}$$

$$q_p(\text{fission plus gamma}) = 47.1 \frac{\text{kw}}{\text{ft}} \times 1.14 \times 1.52 = 81.6 \frac{\text{kw}}{\text{ft}}^{(3)}$$

$$q_d = 0.822q_p = 0.822 \times 81.6 \frac{\text{kw}}{\text{ft}} = 67.1 \text{ kw/ft}$$

$$Q(\text{top to defect}) = \frac{81.6 \text{ kw}}{1.14 \text{ ft}} \times \frac{22.73}{12} \text{ ft} \times 0.095 = 13.4 \text{ kw}^{(4)}$$

Total heat(fuel, basket and tube) = 89.2 kw + 0.8 kw + 8.2 kw = 98.2 kw.

IV. FLOW CALCULATIONS

Let us assume an inlet temperature of 475 F. At this temperature and with the flow controller on the P7 Loop set at 95 per cent, the flow will be 71.5 gpm⁽⁵⁾. At 475 F, and 2000 psi $\rho = 51.0 \text{ lb/ft}^3$ ⁽⁶⁾

-
- (3) The values of axial peaking factor (1.14) and horizontal peaking factor (1.52) used in the sixth rupture test will also be used herein. See Call, R. L., and D.C. Kaulitz, Proposal for Charging the Sixth Rupture Fuel Element Experiment: GEH-10-46, 47, HW-68262, January 1961, (Confidential).
 - (4) The factors used in obtaining q_d and $Q(\text{top to defect})$ are also discussed in HW-68262-RD.
 - (5) Personal communication from J. H. Johnson.
 - (6) The values of ρ , c_p , k and μ used in these calculations were obtained from Figure 1.3.9 and Tables 1.3.3, 1.3.5 and 1.3.6 respectively of The Reactor Handbook, Vol. 2, Engineering, 1st Ed., The Atomic Energy Commission, 1955.

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the total flow through the loop then is:

$$F_t = 71.5 \frac{\text{gal}}{\text{min}} \times \frac{60 \text{ min}}{\text{hr}} \times 0.1337 \frac{\text{ft}^3}{\text{gal}} \times 51.0 \frac{\text{lb}}{\text{ft}^3} = 29,200 \frac{\text{lb}}{\text{hr}}$$

Let us assume an average bulk temperature of 480 F. At this temperature and 2000 psia

$$c_p = 1.129 \text{ BTU/lb-}^\circ\text{F. Then}$$

$$\Delta T(\text{bulk}) = \frac{Q}{F_t \times c_p} = \frac{98.2 \text{ kw} \times 3413 \text{ BTU/kw-hr}}{29,200 \frac{\text{lb}}{\text{hr}} \times 1.129 \frac{\text{BTU}}{\text{lb-}^\circ\text{F}}} = 10.2^\circ\text{F}$$

We want to have the same ΔT around the basket as through it. Let us assume that all the heat from the process tube and one-half the heat from the basket go to the water flowing around the basket. Then the flow outside the basket will be

$$w_o = \frac{Q}{\Delta T c_p} = \frac{8.6 \text{ kw} \times 3413 \text{ BTU/kw-hr}}{10.2^\circ\text{F} \times 1.129 \text{ BTU/lb-}^\circ\text{F}} = 2,550 \text{ lb/hr}$$

Then the total flow through the basket is $W_t = 29,200 - 2550 = 26,650 \text{ lb/hr}$,

and the per cent of the total loop flow which flows around the basket is

$\frac{2550}{29,200} = 8.7\%$. We will assume that this percentage remains constant regardless of changes in inlet temperature and total loop flow.

In the basket the annulus flow area is

$$A_a = \frac{\pi}{4} [(1.758)^2 - (1.253)^2] \text{ in}^2 = .00829 \text{ ft}^2$$

$$\text{Wetted perimeter} = \pi(1.758 + 1.253) \text{ in} = 0.788 \text{ ft}$$

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$$D_{e_a} = \frac{4 \times 0.00829 \text{ ft}^2}{0.788 \text{ ft}} = 0.0421 \text{ ft.}$$

The fuel element bore flow area is:

$$A_b = \frac{\pi}{4} (.439)^2 \text{ in}^2 = 0.001051 \text{ ft}^2$$

$$D_{e_b} = \frac{.439}{12} \text{ ft} = 0.0366 \text{ ft.}$$

Gumprecht⁽⁷⁾ has calculated the flow split for a full charge of NPR fuel elements. Because the I.D. of the basket used in this test is the same as the I.D. of an NPR outer fuel tube, we will use the ratio of flow in the bore to flow in the annulus between inner and outer fuel elements as the flow split for this test. Thus, the flow in the annulus will be 83.3 per cent and flow in the bore 16.7 per cent of the total flow through the basket. Then:

$$w_a = 0.833 w_t = 0.833 \times 26,650 \frac{\text{lb}}{\text{hr}} = 22,200 \frac{\text{lb}}{\text{hr}}$$

$$w_b = 0.167 w_t = 0.167 \times 26,650 \frac{\text{lb}}{\text{hr}} = 4,450 \frac{\text{lb}}{\text{hr}}$$

and

$$G_a = \frac{w_a}{A_a} = \frac{22,000 \text{ lb/hr}}{0.00829 \text{ ft}^2} = 2.68 \times 10^6 \text{ lb/hr-ft}^2$$

$$G_b = \frac{w_b}{A_b} = \frac{4,450 \text{ lb/hr}}{0.001051 \text{ ft}^2} = 4.24 \times 10^6 \text{ lb/hr-ft}^2$$

(7) Personal communication from R. O. Gumprecht.

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V. HEAT SPLIT

Gumprecht(8) has calculated the heat split for an NPR inner fuel tube to be:

65.6% to the annulus

34.4% to the bore

We will use these values in the calculations below.

VI. MAXIMUM COOLANT ΔT 's, TOP TO DEFECT AND TOP TO REACTOR MIDPLANE

Let us assume that the equivalent heat from the basket to the water inside the basket along the maximum heat generating line from the top of the fuel element to the defect point is 0.2 kw. Then, the local bulk ΔT along this line from the top of the fuel element to the defect point will be (at 475 F, $c_p = 1.122$ BTU/lb- $^{\circ}$ F)

$$\Delta T = \frac{0.656 Q_d + 0.2 \text{ kw}}{W_a c_p} = \frac{(0.656 \times 13.4 + 0.2) \text{ kw} \times 3413 \text{ BTU/kw-hr}}{22,200 \text{ lb/hr} \times 1.122 \text{ BTU/lb-}^{\circ}\text{F}}$$
$$= 1.2 ^{\circ}\text{F.}$$

Let us also assume the average water temperature in the annulus along the maximum heat generating line from the top of the fuel element to the reactor midplane is 478 $^{\circ}$ F. Then $c_p = 1.129$ BTU/lb- $^{\circ}$ F. The ΔT in the annulus from the top of the fuel element to the reactor midplane along the maximum heat generation line is:

$$\Delta T_a = \frac{\frac{0.656 Q_d}{2} \times 1.52 + \frac{\text{basket gamma heat}}{4} \times 1.52}{W_a c_p} =$$
$$\frac{\left(\frac{0.656 \times 89.2}{2} \times 1.52 + \frac{0.8}{4} \times 1.52 \right) \times 3413}{22,200 \times 1.129} = 6.1 ^{\circ}\text{F.}$$

(8) Personal communication from R. O. Gumprecht.

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With the F_b correction, this ΔT is $6.1^\circ\text{F} \times 1.16 = 7.1^\circ\text{F}$.

In the bore, let us assume a corresponding water temperature of 483°F .

Then $c_p = 1.134$ and the ΔT to the reactor midplane is

$$\Delta T_b = \frac{0.344 \times 89.2}{2} \times 1.52 \times 3413 = 15.8^\circ\text{F}$$

4450×1.134

With the F_b correction, this ΔT is $15.8^\circ\text{F} \times 1.16 = 18.3^\circ\text{F}$.

VII. WATER FILM COEFFICIENTS

The Modified Colburn Equation⁽⁹⁾ which will be used to calculate the film coefficients is:

$$\frac{h}{(k)_f} \frac{De}{\mu_f} = 0.023 \left(\frac{DeG}{\mu_f} \right)^{0.8} \left(\frac{c_p \mu}{k} \right)_f^{0.3}$$

where $()_f$ indicates that all physical properties within the parentheses are evaluated at the film temperature.⁽¹⁰⁾

Let us assume a film temperature at the defect of 513°F . At this temperature and 2000 psi:

$$c_p = 1.176 \text{ BTU/lb-}^\circ\text{F}$$

$$k = 0.3506 \text{ BTU/hr-ft-}^\circ\text{F}$$

$$\mu = 0.253 \text{ lb/hr-ft}$$

(9) Nertney, op. cit.

(10) The film temperature is defined as the average of the local bulk water temperature and the temperature of the heat transfer surface.

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then,

$$h_d = \frac{0.023 \times 0.3506 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}}{0.0421 \text{ ft}} = \left(\frac{0.0421 \text{ ft} \times 2.68 \times 10^6 \text{ lb/hr-ft}^2}{0.253 \text{ lb/hr-ft}} \right)^{0.8} \\ \left(\frac{1.176 \text{ BTU/lb-}^\circ\text{F} \times 0.253 \text{ lb/hr-ft}}{0.3506 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}} \right)^{0.3} = 6010 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$$

Assume a film temperature of 527 °F at the peak heat generating point at the reactor midplane in the annulus. At this temperature

$$c_p = 1.198 \text{ BTU/lb-}^\circ\text{F}$$

$$k = 0.3338 \text{ BTU/hr-ft-}^\circ\text{F}$$

$$\mu = 0.245 \text{ lb/hr-ft}$$

and,

$$h_a = \frac{0.023 \times 0.3338}{0.0421} \left(\frac{0.0421 \times 2.68 \times 10^6}{0.245} \right)^{0.8} \left(\frac{1.198 \times 0.245}{0.3338} \right)^{0.3} \\ = 5930 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}.$$

Assume a corresponding film temperature in the bore of 536 °F,

then

$$c_p = 1.217 \text{ BTU/lb-}^\circ\text{F}$$

$$k = 0.3392 \text{ BTU/hr-ft-}^\circ\text{F}$$

$$\mu = 0.241 \text{ lb/hr-ft}$$

and,

$$h = \frac{0.023 \times 0.3392}{0.0366} \left(\frac{0.0366 \times 4.24 \times 10^6}{0.241} \right)^{0.8} \left(\frac{1.217 \times 0.241}{0.3392} \right)^{0.3} \\ = 9110 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}.$$

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VIII. ΔT's ACROSS THE COOLANT FILM

$$\Delta T_a = \frac{q_d}{h_d A_a} = \frac{0.656 \times 67.1 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr}}{\frac{6010 \text{ BTU}}{\text{hr/ft}^2 \cdot ^\circ\text{F}} \times 1.253 \pi \times 12 \text{ in}^2 \times \frac{1 \text{ ft}^2}{144 \text{ in}^2}} = 76.2 ^\circ\text{F}$$

$$\Delta T_a = \frac{0.656 \times 81.6 \times 3413}{5930 \times 1.253 \pi \times 1/12} = 93.7 ^\circ\text{F}$$

With the F_h correction $\Delta T_a = 93.7 ^\circ\text{F} \times 1.47 = 136.8 ^\circ\text{F}$.

$$\Delta T_b = \frac{0.344 \times 81.6 \times 3413}{9110 \times 0.439 \pi \times 1/12} = 91.7 ^\circ\text{F}$$

With the F_h correction $\Delta T_b = 91.7 ^\circ\text{F} \times 1.47 = 135.8 ^\circ\text{F}$.

IX. ΔT's ACROSS THE CLADDING

Assume that at the maximum heat generating point at the reactor midplane the conductivity (k) of both the inner and outer cladding is 8.18 BTU/hr-ft-°F. (11) Then:

$$\Delta T_a = \frac{0.656 q_p \ln \frac{D_o}{D_i}}{2\pi k} = \frac{0.656 \times 81.6 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr} \times \ln \frac{1.253}{1.173}}{2\pi \times 8.18 \text{ BTU/hr/ft-}^\circ\text{F}}$$

$$= 236.5 ^\circ\text{F}$$

$$\Delta T_b = \frac{0.344 \times 81.6 \times 3413 \times \ln \frac{0.489}{0.439}}{2\pi \times 8.18} = 197.7 ^\circ\text{F}$$

(11) This is a reasonable value to use at the cladding temperature involved. See Merckx, K. R., (ed) Fuel Element Design Handbook, HW-51000, Sept. 1958, p. 302. 1T.

X. ΔT's ACROSS THE BOND LAYER

Let us assume a heat transfer coefficient of 30,000 BTU/hr-ft²-°F for the cladding-core bond layers.⁽¹²⁾ Then:

$$\Delta T_a = \frac{0.656 x q_p}{h A_a} = \frac{0.656 x 81.6 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr}}{30,000 \frac{\text{BTU}}{\text{hr-ft}^2\text{-}^\circ\text{F}} \times \frac{1.173 \times 12 \text{ in}^3}{\text{ft}} \times \frac{1 \text{ ft}^2}{144 \text{ in}^2}} = 20.0^\circ\text{F}$$

$$\Delta T_b = \frac{0.344 x 81.6 x 3413}{30,000 x 0.489 \times 12 \frac{\text{in}^3}{\text{ft}}} = 25.2^\circ\text{F}$$

XI. ΔT ACROSS THE CORE

Geering⁽¹³⁾ has developed the following equation for calculating the temperature drop through a tubular fuel element:

$$\Delta T = \frac{q}{4k} \left[\frac{r_a^2 - r_b^2}{\ln \frac{r_a}{r_b}} \times \ln \left(\sqrt{\frac{r_a^2 - r_b^2}{2 \ln \frac{r_a}{r_b}}} \right) + r_a^2 - \frac{r_a^2 - r_b^2}{2 \ln \frac{r_a}{r_b}} \right]$$

where ΔT, q, k, and r are in a consistent system of units.

Assume an average core temperature of 930 °F at the peak heat generating point in the fuel element. At this temperature

$$k = 20.7 \frac{\text{BTU}}{\text{hr-ft-}^\circ\text{F}} \text{ (14) or } k = 1.723 \frac{\text{BTU}}{\text{hr-in-}^\circ\text{F}} \text{ . Also,}$$

$$q_p = 81.6 \text{ kw/ft} \times \frac{1 \text{ ft}}{\frac{\pi}{4} [(1.173)^2 - (0.489)^2] \times 12 \text{ in}^3} = 7.62 \text{ kw/in}^3.$$

(12) Ibid, p. A100-5.

(13) Geering, G. T., Tubular Fuel Elements, Part One, Operation of Prototype Elements in the ETR Hanford Loop, HW-59940, April, 1959., Confidential.

(14) Tebo, F. V., Selected Values of the Physical Properties of Various Materials, ANL-5914, September 1958, p. 92.

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Then:

$$\Delta T = \frac{7.62 \frac{\text{kw}}{\text{in}^3} \times 3413 \frac{\text{BTU}}{\text{kw-hr}}}{4 \times 1.723 \frac{\text{BTU}}{\text{hr-in-}^\circ\text{F}}} \left[\frac{\left(\frac{1.173}{2}\right)^2 - \left(\frac{0.489}{2}\right)^2}{\ln \frac{1.173}{0.489}} \right] \times$$

$$\ln \left(\sqrt{\frac{\left(\frac{1.173}{2}\right)^2 - \left(\frac{0.489}{2}\right)^2}{2 \ln \frac{1.173}{0.489}}} \right) + \left[\left(\frac{1.173}{2}\right)^2 - \frac{\left(\frac{1.173}{2}\right)^2 - \left(\frac{0.489}{2}\right)^2}{2 \ln \frac{1.173}{0.489}} \right] \text{in}^2$$

$$= 231.3 ^\circ\text{F}.$$

XII. TEMPERATURE DISTRIBUTIONS

At the Defect

Inlet temperature
 ΔT_a (top to defect)
 Local bulk water temperature
 ΔT_a (film)
 Cladding surface temperature

Best Estimate

475 $^\circ\text{F}$
 1.2
 476.2 $^\circ\text{F}$
 76.2
 552.4 $^\circ\text{F}$

At Reactor Midplane - From the Annulus Inward

	<u>Best Estimate</u>	<u>F_b and F_h Adjusted Temperature</u>
Inlet temperature	475 $^\circ\text{F}$	475 $^\circ\text{F}$
ΔT_a (top to midplane)	6.1	7.1
Local bulk water temperature	481.1 $^\circ\text{F}$	482.1 $^\circ\text{F}$
ΔT_a (film)	93.7	136.8
ΔT add		5.7
Cladding surface temperature	574.8 $^\circ\text{F}$	624.6 $^\circ\text{F}$
ΔT_a (cladding)	236.5	236.5
Bond layer temperature	811.3 $^\circ\text{F}$	861.1 $^\circ\text{F}$
ΔT_a (bond layer)	20.0	20.0
Uranium surface temperature	831.3 $^\circ\text{F}$	881.1 $^\circ\text{F}$
ΔT (core)	231.3	231.3
Core center temperature (maximum)	1062.6 $^\circ\text{F}$	1112.4 $^\circ\text{F}$

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At Reactor Midplane - From the Bore Outward

	<u>Best Estimate</u>	<u>F_b and F_h Adjusted Temperature</u>
Inlet temperature	475 °F	475 °F
ΔT _b (top to midplane)	15.8	18.3
Local bulk water temperature	490.8 °F	493.3 °F
ΔT _b (film)	91.7	135.8
ΔT add		5.7
Cladding surface temperature	582.5 °F	634.8 °F
ΔT _b (cladding)	197.7	197.7
Bond layer temperature	780.2 °F	832.5 °F
ΔT _b (bond)	25.2	25.2
Uranium surface temperature	805.4 °F	857.7 °F
ΔT (core)	231.3	231.3
Core center temperature (maximum)	1036.7 °F	1089.0 °F

XIII. MAXIMUM SURFACE HEAT FLUXES

$$E_a = \frac{0.656 q_p}{A_a} = \frac{0.656 \times 81.6 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr}}{\frac{1.253\pi \times 12 \text{ in}^2}{\text{ft}} \times \frac{1 \text{ ft}^2}{144 \text{ in}^2}} = 352,000 \frac{\text{BTU}}{\text{hr-ft}^2}$$

$$E_b = \frac{0.344 \times 81.6 \times 3413}{0.439\pi \times 12 \times \frac{1}{144}} = 526,000 \text{ BTU/hr-ft}^2$$

XIV. BURNUP RATES (Fission Heat Only)

$$\begin{aligned} \text{Average burnup rate} &= \frac{q}{\text{weight of fuel}} = \frac{81.8 \times 10^{-3} \text{ MW}}{6.29 \text{ kg} \times \frac{2.2 \text{ lb}}{\text{kg}} \times \frac{\text{ton}}{2000 \text{ lb}}} \\ &= 11.8 \frac{\text{MW}}{\text{T}} \end{aligned}$$

Maximum burnup rate = Ave. burnup rate x peaking factors

$$= 11.8 \frac{\text{MW}}{\text{T}} \times 1.14 \times 1.52 = 20.5 \frac{\text{MW}}{\text{T}}$$

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XV. ADDITIONAL CALCULATIONS

The foregoing calculations were repeated at fuel element powers 60 and 140 per cent of the powers used above. The data thus obtained were used to plot the curves of Figures 4, 5, 8, 11, 12, 13, 14, and 15. These calculations will not be repeated here. The inlet temperature at each fuel element power was chosen so as to give a cladding surface temperature of 552 at the defect.

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FIGURE 1

TOTAL LOOP FLOW AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 95%)

FIGURE 2

AVERAGE COOLANT VELOCITY IN THE ANNULUS BETWEEN THE BASKET AND FUEL ELEMENT AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 95%)

FIGURE 3

AVERAGE COOLANT VELOCITY IN THE BORE OF THE FUEL ELEMENT AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 95%)

INLET AND OUTLET TEMPERATURES AS A FUNCTION OF BULK WATER TEMPERATURE RISE IN THE LOOP

FIGURE 4

FIGURE 5

MAXIMUM SPECIFIC POWER GENERATION (FISSION PLUS GAMMA HEATING)
AS A FUNCTION OF BULK WATER TEMPERATURE RISE IN THE LOOP

FLOW IN THE BORE OF THE FUEL ELEMENT AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 95%)

FIGURE 7

FLOW IN THE ANNULUS BETWEEN THE BASKET AND THE FUEL ELEMENT AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 95%)

FIGURE 6

FIGURE 8

MAXIMUM SURFACE HEAT FLUX AS A FUNCTION OF BULK WATER TEMPERATURE
RISE IN THE LOOP

FILM COEFFICIENT IN THE BORE OF THE FUEL ELEMENT AS A FUNCTION OF INLET TEMPERATURE

FIGURE 10

FILM COEFFICIENT IN THE ANNULUS BETWEEN THE BASKET AND THE FUEL ELEMENT AS A FUNCTION OF INLET TEMPERATURE

FIGURE 9

TOTAL HEAT GENERATION (FISSION PLUS GAMMA IN FUEL, BASKET, AND PROCESS TUBE) AS A FUNCTION OF BULK WATER TEMPERATURE RISE IN THE LOOP

FIGURE 11

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FIGURE 12

MAXIMUM CLADDING SURFACE TEMPERATURES AS A FUNCTION OF BULK
WATER TEMPERATURE RISE IN THE LOOP

FIGURE 13

MAXIMUM URANIUM SURFACE TEMPERATURES AS A FUNCTION OF BULK WATER TEMPERATURE RISE IN THE LOOP

FIGURE 14

MAXIMUM FUEL ELEMENT CORE CENTER TEMPERATURE AS A FUNCTION OF
BULK WATER TEMPERATURE RISE IN THE LOOP

FIGURE 15

BURNUP RATE (FISSION HEAT ONLY) AS A FUNCTION OF BULK WATER TEMPERATURE RISE IN THE LOOP

FIG-17

HYDRAULIC CYLINDER
ASSY

FIG - 18

HYDRAULIC CONTROLS

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