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PROPOSAL FOR CHARGING THE FIFTH RUPTURE
FUEL EXPERIMENT: GEH-10, 34, 35

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August 25, 1960

PROPOSAL FOR CHARGING THE FIFTH RUPTURE FUEL EXPERIMENT
GEH-10; 34, 35

by

R. L. Call
Fuel Element Design
Hanford Laboratories

D. C. Kaulitz
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Hanford Laboratories

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PROPOSAL FOR CHARGING THE FIFTH RUPTURE FUEL
EXPERIMENT: GEH-10; 34, 35

R. L. Call and D. C. Kaulitz

I. OBJECTIVE

The objective of this irradiation is to further verify the corrosion rate of tubular-type fuel elements under conditions of high specific power and central core temperatures. This fuel will be the inner tube only of an NPR fuel assembly. As in previous tests, this inner tube rupture will be used to further substantiate the rupture detection instrumentation that is being used in the development of the NPR. Previously unirradiated fuel will be used in this test. The reactor is to operate at full power during the test.

Permission is requested for charging two tubular elements. The top element will have attached to it a hydraulic mechanism for opening a defect in the outer surface of the tube. The second, or bottom element, will be used as a heater element to maintain loop temperature.

II. SUMMARY OF TEST CONDITIONS

Maximum Fuel Temperature	-	See Figure 1.
Maximum Surface Temperature	-	Outer Surface - See Figure 2. Bore - See Figure 3.
Weight of Uranium, gm/in	-	394
Total Weight of Uranium, gm	-	9,450
Maximum Power Generation, kw/ft	-	See Figure 4.
Number of Fuel Element Assemblies	-	2
Water Flow Rate	-	17,200 lb/hr (41 gal/min at 435 F)
Water Velocity	-	See Figure 5.
Film Coefficients, BTU/hr-ft ² -°F ⁽¹⁾		
Annulus	-	7,900
Bore	-	11,600
Maximum Loop Pressure, psia	-	2,000

(1)

These coefficients vary slightly with changes in power and inlet temperature, but for the conditions of this test the variations are so small they can be neglected.

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-2-

HW-66603

III. DESCRIPTION OF THE FUEL ELEMENTS

Two NPR inner tubes are proposed for this irradiation. The top element will be number 34, and the bottom element will be number 35. The fuel elements are fabricated from natural uranium and coextruded with Zircaloy-2 jackets 0.020 inch thick. The tubes have an O.D. of 1.421 inches and an I.D. of 0.526 inches, they are 12.5 long with 12.0 inches of uranium and 1/4-inch end caps on each end. The end caps are Zircaloy-2 and are welded to the ends of the counter bored tubes by the vacuum electron-beam welding process. The tubes have supports welded to the outer surface to maintain the outer annulus required for the coolant stream. The top tube which has the rupture defect has an additional end fitting welded to it for attaching the hydraulically operated rupture mechanism. All of the components have been autoclaved for 48 hours in 752 F steam to insure the integrity of the jacket and the welded closures. Figure 6 is a cross section drawing of the defected tube in the basket. Figure 7 is a photograph of both tubes after autoclaving.

The element with the defect and hydraulic cylinder will be placed on top of the other element. The second or bottom element is used only for adding heat to the loop so desired coolant temperatures can be obtained.

IV. DESCRIPTION OF THE RUPTURE MECHANISM

Details of the rupturing mechanism are shown in Figures 8 and 9. It is the same design as used in previous rupture tests such as GEH-10; 28, 29, 30, 31. Loop pressure will again hold the chisel in the retracted position until it is required to advance. The Black Hawk pump is used to move the piston forward to shear the defect cap. Loop pressure returns the piston to the retracted position.

The same type unit has been operated successfully in-reactor on tubular type fuel elements. No malfunctioning of the mechanism was observed.

V. DESCRIPTION OF THE FUEL BASKET ASSEMBLY

The two fuel elements, basket and rupture mechanism will be shipped from Hanford unassembled. The basket will be assembled by Hanford representatives at the ETR site to insure that the unit will function properly. The basket is fabricated from 35 mil wall Zircaloy-2 tubing and is 54-1/4 inches long with an average inside diameter of 1.59 inches. An orifice welded in the downstream end is designed to permit 40 gpm through the basket past the fuel. One gpm will flow around the basket to cool the process tube and basket. The foregoing conditions apply to loop operation at an inlet temperature of 435 F.

This test will not use special ΔT thermocouples. The installed loop couples will be used to measure power generation. It is requested, though, that potentiometer checks be made to assure more accurate readings.

VI. THERMAL AND HYDRAULIC TABLE

The following table specifies the required thermal conditions for the reactor loop facility for the proposed test:

TABLE I

Reactor Power Level MW	-	175
Loop Flow Rate, lb/hr	-	17,200 (41 gpm at 435 F)
Loop Pressure, psia	-	2000
Saturation Temperature of Loop Water °F	-	636
Coolant Velocity Past Fuel, See Figure 5	-	
Heat Transfer Film Coefficient BTU/hr-ft ² -°F	-	
Annulus	-	7,900 (see footnote, page 1)
Bore	-	11,600
Inlet Water Temperature	-	See Figure 10
Outlet Water Temperature	-	See Figure 11
Pressure Drop Across Fuel at 41 gpm (cold), psi	-	8

VII. THERMAL CHARACTERISTICS OF THE EXPERIMENT

The following table shows the thermal characteristics of the experiment as detailed by the calculations submitted in Appendix I of this document.

TABLE II

Total Heat Output of the Fuel	-	See Figure 12
Maximum Surface Heat Flux and Burnout Flux		
Outer Surface	-	See Figure 13
Bore	-	See Figure 14
Maximum Cladding Surface Temperature		
Outer Surface	-	See Figure 2
Bore	-	See Figure 3
Maximum Fuel Core Temperature	-	See Figure 1
Average Burnup Rate	-	See Figure 15

VIII. HAZARDS

Several tubular elements have been irradiated successfully in Hanford reactors to high exposures in 445 F water. Past rupture tests performed in the ETR on unirradiated elements show that corrosion rates are slow enough that they will not endanger the reactor or facility tube. All parts of the GEH-R3A, P7 loop are fully instrumented.

For this test, the activity monitor will not be connected to the reactor scram circuit, but the activity level scram will be manually controlled. Provision has been made to record coolant activity, temperature, flow and pressure throughout the loop facility. Should any of these parameters reach conditions not within the operating limits of the loop, the reactor will be scrammed before any damage to the reactor or loop components could occur.

The intentional rupture of the green tubular element will not be hazardous to the facility or reactor. Previous tests of high exposure and green fuel ruptures have proved that the facility can handle the contamination and that the loop can be decontaminated.

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-4-

HW-65603

IX. SPECIAL INSTRUMENTATION

The instruments on the DN loop that measure gross gamma and delayed neutron activity will be used. A representative from Hanford will be present to consult with Phillips Petroleum Company personnel regarding operation of that equipment.

X. TECHNICAL BASES FOR THE PROPOSED TEST

Previous rupture tests in the ETR have been conducted using 7-rod clusters and KER tube and tube fuel elements. This test will be a prototype of the NFR inner tube and will meet our requirements for establishing base corrosion rates.

XI. TEST PROCEDURE

After the test assembly has been loaded in the GEH-3X3P7 process tube and the reactor is at full power, the loop inlet is to be adjusted to the value determined from Figure 10. It should be kept in mind that the conditions of Figure 10 are based on a total loop flow of 17,200 lb/hr (41 gpm at an inlet temperature of 435 F). This will give a surface temperature of 576 F at the defect which is the temperature desired. When the conditions of Figure 10 have been met, the fuel is to be ruptured. We are requesting, as in the previous two tests, seven full days of operation at full power after the fuel has been ruptured. However, if the element shows gross failure and a large release of fission products occurs prior to the end of the seven days, the reactor is to be scrammed and loop temperature dropped as quickly as possible. The criterion for gross failure of the fuel element and termination of the test will be an activity level in the loop piping high enough to cause a 40 R reading on the unshielded scintillator which is located in the corner of the GEH-3X3P7 cubicle. The scintillator should be calibrated prior to the test.

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APPENDIX I

HW-66608
8/27/60HEAT TRANSFER CALCULATIONSI. SYMBOLS

- A - Area (ft²)
 C_p - Specific heat (BTU/lb.-°F)
 D - Diameter (ft. or in.)
 D_e - Equivalent diameter (ft.)
 e - The base of natural logarithms
 E - Surface heat flux (BTU/hr-ft²)
 E_{BO} - Burnout heat flux (BTU/hr-ft²)
 G - Mass flow per unit area (lb/hr-ft²)
 h - Heat transfer coefficient (BTU/hr-ft²-°F)
 H - Enthalpy of the liquid (BTU/lb)
 k - Thermal conductivity (BTU/hr-ft-°F)
 L - Length of a fuel element or length of the fuel charge (ft)
 m - Mass (gm/gm-atom)
 N - Avogadro's number (6.02X10²³ atom/gm-atom)
 P - Total power generation (kw or BTU/hr)
 Q - Specific power (kw/ft)
 q - specific power (watt/cm³)
 r - radius (cm)
 T - Temperature (°F)
 V - Specific volume of the fuel (in³/ft. of tube)
 v - velocity (ft/sec)
 w - mass flow rate (lb/hr)
 μ - Viscosity (lb/hr)
 ρ - Density (lb/ft³ or gm/cm³)
 Σ_f - Macroscopic fission cross section (cm⁻¹)

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-6-

HW-66603
8/25/60

- σ_f - Microscopic fission cross section (cm^2)
 ϕ - Volume average neutron flux in the fuel (nv or neutrons/ $\text{cm}^2\text{-sec}$)

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

- a - average
- b - bore (or inner surface of the fuel tube)
- d - located at or related to the defect
- o - outer surface or annulus around the fuel tube
- p - peak or maximum value
- t - total

II. HOT SPOT AND HOT CHANNEL FACTORS

	<u>F_b</u>	<u>F_h</u>	<u>ΔT Add</u>
A Mechanical Tolerances			
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			

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-7-

HW-66603
8/25/60

	<u>F_b</u>	<u>F_h</u>	<u>T add</u>
1. Flow measurements			
2. Temperature measurement	1.03	1.03	1.4
3. Pressure measurements			
4. Temperature measurements (coolant)	1	1	4.3
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	1	
G. Neutron and Gamma Flux Uncertainty			
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	
H. Heat Flux Over Element Surface			
1. Fuel homogeneity	1	1.02	
2. Mechanical effects	1	1	
I. Effects Peculiar to Experiment	1	1	
Products (total ΔT add)	1.16	1.47	5.7

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-8-

HW-66603
8/25/60

III. CALCULATION OF FLUXES

$$\begin{aligned}\Sigma_f &= \sigma_f \frac{\rho}{m} N \\ &= \frac{4.2 \times 10^{-24} \frac{\text{cm}^2}{\text{nucleus}} \times 18.9 \frac{\text{gm}}{\text{cm}^3} \times 6.02 \times 10^{23} \frac{\text{nuclei}}{\text{gm-atom}}}{238 \frac{\text{gm}}{\text{gm-atom}}} \\ &= 0.200 \text{ cm}^{-1}\end{aligned}$$

The peak, axial, unperturbed flux in the GEH-R33P7 loop during Cycle 32 will be 1.47×10^{14} nv. ⁽¹⁾ The flux depression factor of 0.516 obtained from the previous rupture tests will be used here. Then the peak, axial, perturbed flux will be:

$$1.47 \times 10^{14} \text{ nv} \times 0.516 = 0.758 \times 10^{14} \text{ nv.}$$

The correction for horizontal peaking at the reactor midplane is 1.52. ⁽¹⁾

Therefore, the absolute, peak, perturbed flux will be:

$$\phi_p = 0.758 \times 10^{14} \text{ nv} \times 1.52 = 1.152 \times 10^{14} \text{ nv.}$$

The ratio of peak axial flux to average axial flux for this test will be 1.14. ⁽²⁾

Therefore the average effective flux will be:

$$\phi_a = 0.758 \times 10^{14} \text{ nv} \times \frac{1}{1.14} = 0.665 \times 10^{14} \text{ nv.}$$

IV. POWER GENERATION

Let us assume that the total heat released per fission is 194 Mev. ⁽³⁾ Assume that 95 per cent of the heat is released within the fuel core. Then the heat generation in the core is:

$$194 \frac{\text{Mev}}{\text{fission}} \times 0.95 = 184 \frac{\text{Mev}}{\text{fission}}$$

(1) Personal Communication from J. H. Johnson.

(2) Calculated on the basis of the GEH-R33P7 loop flux map of Cycle 3.

(3) Based on the example for the swimming pool reactor on p. 642 of Principles of Nuclear Reactor Engineering, 1st Edition, S. Glasstone

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-9-

NW-66603
8/25/60

The peak specific power will be:

$$\begin{aligned}
 Q_p &= \text{heat/fission} \times \sum_f \phi_p V \\
 &= 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.200 \frac{\text{fission}}{\text{cm-neutron}} \\
 &\quad \times 1.152 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2 \text{ sec}} \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{\pi}{4} (1.390^2 - 0.560^2) \\
 &\quad \times \frac{12 \text{ in}^3}{\text{ft. of tube}} \\
 &= 172 \text{ kw/ft of tube} \\
 q_p &= 172,000 \frac{\text{watt}}{\text{ft}} \times 0.0610 \frac{\text{in}^3}{\text{cm}^3} \\
 &\quad \frac{\pi}{4} (1.390^2 - 0.560^2) \times 12 \frac{\text{in}^3}{\text{ft}} = 688 \frac{\text{watt}}{\text{cm}^3}
 \end{aligned}$$

The average specific power generation will be:

$$\begin{aligned}
 Q_a &= \frac{\text{heat}}{\text{fission}} \sum_f \phi_a V \\
 &= 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.200 \frac{\text{fission}}{\text{cm-neutron}} \\
 &\quad \times 0.665 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2 \text{ sec}} \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{\pi}{4} (1.390^2 - 0.560^2) \times \frac{12 \text{ in}^3}{\text{ft}} \\
 &= 99.3 \text{ kw/ft.}
 \end{aligned}$$

The ratio of flux at the defect to peak flux is 0.822. Also, the fraction of the total heat generated by the charge which is generated between the top of the first element and the defect is 9.85 per cent. (4) Therefore, the maximum specific power at the defect will be:

$$Q_d = 172 \frac{\text{kw}}{\text{ft}} \times 0.822 = 141 \text{ kw/ft.}$$

(4) See Footnote (2).

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-10-

HW-66603
8/25/60

The heat generated between the top of the first element and the defect (assume that the fuel element operates in a flux field equal at each point to the axial flux multiplied by the horizontal peaking factor) will be:

$$P_d = \frac{172 \text{ kw}}{1.14 \text{ ft}} \times 2 \text{ ft} \times 0.0985 = 29.7 \text{ kw.}$$

V. FLOW IN THE VARIOUS CHANNELS

$$D_e = \frac{4 \times \text{flow area}}{\text{wetted perimeter}}$$

$$\begin{aligned} \text{Annulus flow area} &= \frac{\pi}{4} (1.590^2 - 1.421^2) \\ &= 0.127 \pi \text{ in}^2 = 0.000882 \pi \text{ ft}^2 \\ &= 0.00277 \text{ ft}^2 \end{aligned}$$

$$\begin{aligned} \text{Bore flow area} &= \frac{\pi}{4} \times 0.526^2 = 0.0692 \pi \text{ in}^2 = 0.000480 \pi \text{ ft}^2 \\ &= 0.00151 \text{ ft}^2 \end{aligned}$$

$$\begin{aligned} \text{wetted perimeter of annulus} &= \pi (1.590 + 1.421) \\ &= 3.011 \pi \text{ in} = 0.251 \pi \text{ ft} \end{aligned}$$

$$D_{e_a} = \frac{4 \times 0.000882 \pi \text{ ft}^2}{0.251 \pi \text{ ft}} = 0.0140 \text{ ft}$$

$$D_{e_b} = 0.526 \text{ in} = 0.0438 \text{ ft}$$

Assume a total flow of 41 gpm at an inlet temperature of 435 F. Assume

40 gpm to flow through the basket and 1 gpm around it. Total mass flow

$$\begin{aligned} &= 41 \frac{\text{gal}}{\text{min}} \times \frac{0.134 \text{ ft}^3}{\text{gal}} \times \frac{60 \text{ min}}{\text{hr}} \times 52.2 \frac{\text{lb}}{\text{ft}^3} \\ &= 17,200 \text{ lb/hr.} \end{aligned}$$

$$\begin{aligned} \text{Flow through the basket} &= 40 \frac{\text{gal}}{\text{min}} \times 0.134 \frac{\text{ft}^3}{\text{gal}} \\ &\times 60 \frac{\text{min}}{\text{hr}} \times 52.2 \frac{\text{lb}}{\text{ft}^3} = 16,800 \frac{\text{lb}}{\text{hr}} \end{aligned}$$

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HW-66603
8/25/66

The above flows (lb/hr) should remain constant throughout the test.

The relative flows through the bore and annulus can be calculated by means of the following equations: (5)

$$v_t A_t = v_o A_o + v_b A_b = 40 \text{ gal/min}$$

$$\frac{v_o^{1.8}}{D_{e_o}^{1.2}} = \frac{v_b^{1.8}}{D_{e_b}^{1.2}}$$

$$\text{Then } v_o^{1.8} \left(\frac{D_{e_o}}{D_{e_b}} \right)^{1.2} = v_b^{1.8}$$

$$\text{and } v_o = v_b \left(\frac{D_{e_o}}{D_{e_b}} \right)^{\frac{1.2}{1.8}} = v_b \left(\frac{0.0140}{0.0438} \right)^{2/3}$$

$$= 0.467 v_b$$

$$\text{Then } 0.467 v_b A_o + v_b A_b = 40 \text{ gal/min.}$$

$$v_b = \frac{40 \text{ gal/min}}{0.467 A_o + A_b} = \frac{40 \text{ gal/min}}{0.467 \times 0.00277 \text{ ft}^2 + 0.00151 \text{ ft}^2}$$

$$= 14280 \frac{\text{gal}}{\text{min-ft}^2} \times 0.134 \frac{\text{ft}^3}{\text{gal}} \times \frac{\text{min}}{60 \text{ sec}} = 31.9 \text{ ft/sec}$$

$$v_o = 0.467 v_b = 0.467 \times 31.9 = 14.9 \text{ ft/sec}$$

$$w_o = v_o A_o = 14.9 \text{ ft/sec} \times 0.00277 \text{ ft}^2 \times \frac{3600 \text{ sec}}{\text{hr}} \times 52.2 \frac{\text{lb}}{\text{ft}^3}$$

$$= 7,750 \text{ lb/hr}$$

(5) Personal Communication from E. D. Waters.

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8/25/60

$$w_b = V_b A_b = 31.0 \frac{\text{ft}}{\text{sec}} \times 0.00151 \text{ ft}^2 \times \frac{3600 \text{ sec}}{\text{hr}} \times 52.2 \frac{\text{lb}}{\text{ft}^3}$$

$$= 9050 \text{ lb/hr}$$

$$G_o = \frac{w_o}{A_o} = \frac{7750 \text{ lb/hr}}{0.00277 \text{ ft}^2} = 2.80 \times 10^6 \text{ lb/hr-ft}^2$$

$$G_b = \frac{w_b}{A_b} = \frac{9050 \text{ lb/hr}}{0.00151 \text{ ft}^2} = 5.99 \times 10^6 \text{ lb/hr-ft}^2$$

VI. HEAT SPLIT

The method developed by Geering⁽⁶⁾ was used to calculate the heat split. The figures thus obtained were adjusted by trial and error so that the core temperature obtained by calculating from the outer surface inward was within a few degrees of the value obtained by calculating from the bore inward. The values chosen were 70 per cent to the outer surface and 30 per cent to the inner surface.

VII. MAXIMUM COOLANT ΔT 's

Top of First Fuel Element to Defect

Assume an inlet temperature of 435 F, assume the average water temperature along the line of maximum power generation between the top and the defect is 439 F. Then C_p at 439 F and 2000 psia is 1.09 BTU/lb-°F.

$$\Delta T_d = \frac{0.700 P_d}{w_o C_p} = \frac{0.700 \times 29.7 \text{ kw} \times 3413 \text{ BTU/kw-hr}}{7750 \text{ lb/hr} \times 1.09 \text{ BTU/lb-°F}}$$

$$= 7.7 \text{ °F.}$$

Top of the First Fuel Element to Reactor Mid-Plane

Assume the average water temperature along the maximum power generation line between the top and the mid-plane in the annulus is 454 F. Then $C_p = 1.108 \text{ BTU/lb-°F}$

$$\Delta T_o = \frac{0.700 \times \frac{172}{1.14} \text{ kw/ft} \times 1 \text{ ft} \times 3413 \text{ BTU/kw-hr}}{7750 \text{ lb/hr} \times 1.108 \text{ BTU/lb-°F}}$$

$$= 42.0 \text{ F.}$$

(6) Geering, G. T., Tubular Fuel Elements, Part One, Operation of Prototype Elements in the ETR Hanford Loop, HW-59940, April 9, 1959, Confidential

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-13-

HW-66603
8/25/60

With F_b correction $\Delta T_o = 42.0 \times 1.16 = 48.7$ F

Assume that in the bore the average water temperature will be 442 F.

Then $C_p = 1.095$ BTU/lb-°F

$$\Delta T_b = 0.300 \times \frac{172}{1.14} \text{ kw/ft} \times 1 \text{ ft} \times 3413 \text{ BTU/kw-hr}$$

$$\frac{9050 \frac{\text{lb}}{\text{hr}} \times 1.095 \text{ BTU/lb-°F}}$$

$$= 15.6^\circ\text{F.}$$

With F_b correction $\Delta T_b = 15.6 \times 1.16 = 18.1$ °F.

VIII. WATER FILM COEFFICIENTS

The Modified Colburn Equation⁽⁷⁾ is:

$$\frac{h D_e}{(k)_f} = 0.023 \left(\frac{D_e G}{\mu}_f \right)^{0.8} \left(\frac{C_p \mu}{k}_f \right)^{0.3}$$

where $()_f$ indicates that all physical properties within the parentheses are evaluated at the film temperature.⁽⁸⁾ The equation can be rewritten as follows:

$$h = \frac{0.023(k)_f}{D_e} \left(\frac{D_e G}{\mu}_f \right)^{0.8} \left(\frac{C_p \mu}{k}_f \right)^{0.3}$$

Assume a film temperature at the defect of 500 F. Then:

$$C_p = 1.16 \text{ BTU/lb-°F}$$

$$k = 0.348 \text{ BTU/hr-ft-°F}$$

$$\rho = 48.8 \text{ lb/ft}^3$$

$$\mu = 0.24 \text{ lb/hr-ft}$$

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

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

DECLASSIFIED

-14-

HW-66603

8/25/60

$$h = \frac{0.023 \times 0.348 \text{ BTU/hr-ft-}^\circ\text{F}}{0.0140 \text{ ft}} \left(\frac{0.0140 \text{ ft} \times 2.80 \times 10^6 \text{ lb/hr-ft}^2}{0.24 \text{ lb/hr-ft}} \right)^{0.8} \\ \times \left(\frac{1.16 \text{ BTU/lb-}^\circ\text{F} \times 0.24 \text{ lb/hr-ft}}{0.348 \text{ BTU/hr-ft-}^\circ\text{F}} \right)^{0.3} \\ = 7900 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}.$$

Assume a film temperature at the reactor mid-plane in the annulus of 547 F.

Then:

$$C_p = 1.25 \text{ BTU/lb-}^\circ\text{F}$$

$$k = 0.325 \text{ BTU/hr-ft-}^\circ\text{F}$$

$$\rho = 46.2 \text{ lb/ft}^3$$

$$\mu = 0.22 \text{ lb/hr-ft}$$

$$h_o = \frac{0.023 \times 0.325}{0.0140} \left(\frac{0.0140 \times 2.80 \times 10^6}{0.22} \right)^{0.8} \left(\frac{1.25 \times 0.22}{0.325} \right)^{0.3} \\ = 7910 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$$

Assume a film temperature in the bore at the reactor mid-plane of 504 F. Then:

$$C_p = 1.16 \text{ BTU/lb-}^\circ\text{F}$$

$$k = 0.348 \text{ BTU/hr-ft-}^\circ\text{F}$$

$$\rho = 48.8 \text{ lb/ft}^3$$

$$\mu = 0.24 \text{ lb/hr-ft}$$

$$h = \frac{0.023 \times 0.348}{0.0438} \left(\frac{0.0438 \times 5.99 \times 10^6}{0.24} \right)^{0.8} \left(\frac{1.16 \times 0.24}{0.348} \right)^{0.3} \\ = 11,600 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$$

DECLASSIFIED

-15-

HW-66603
8/25/60

IX. ΔT 's ACROSS THE COOLANT FILM

$$\Delta T_d = \frac{0.700 Q_d}{h A_o} = \frac{0.700 \times 141 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr}}{7900 \text{ BTU/hr-ft}^2 \cdot ^\circ\text{F} \times \frac{1.421 \pi \times 12 \text{ ft}^2}{144}} = 114 \text{ F}$$

$$\Delta T_o = \frac{0.700 \times 172 \times 3413}{7910 \times \frac{1.421 \pi \times 12}{144}} = 137 \text{ F}$$

with F_h correction $\Delta T_o = 137 \times 1.47 = 202 \text{ F}$

$$\Delta T_b = \frac{0.300 \times 172 \times 3413}{11,600 \times \frac{0.526 \pi \times 12}{144}} = 108 \text{ F}$$

With F_h correction $\Delta T_b = 108 \times 1.47 = 159 \text{ F}$.

X. ΔT 's ACROSS THE JACKETS

Assume that at the reactor mid-plane the average temperature of the outer jacket at the peak power point is 704 F. Then:

$$\begin{aligned} k &= 8.18 \text{ BTU/hr-ft-}^\circ\text{F} \\ \Delta T_o &= \frac{0.700 \times Q_p \times \ln \frac{D_o}{D_b}}{2 \pi k} \\ &= \frac{0.700 \times 172 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr} \times \ln \frac{1.421}{1.390}}{2 \pi \times 8.18 \text{ BTU/hr-ft-}^\circ\text{F}} \\ &= 174 \text{ F.} \end{aligned}$$

Assume that at the bore the average temperature is 663 F. Then $k = 8.18$ BTU/hr-ft- $^\circ\text{F}$.

$$\begin{aligned} \Delta T_b &= \frac{0.300 \times 172}{2 \pi \times 8.18} \times 3413 \times \ln \frac{0.560}{0.526} \\ &= 212 \text{ F.} \end{aligned}$$

DECLASSIFIED

-16-

HW-66603
8/25/60

XI. BOND LAYER ΔT 's

Assume $h = 30,000$ BTU/hr-ft²-°F for the bond layer.

$$\Delta T_o = \frac{0.700 \times q_p}{h A_o} = \frac{0.700 \times 172 \text{ kw/ft} \times 1 \text{ ft} \times 3413 \text{ BTU/kw-hr}}{30,000 \text{ BTU/hr-ft}^2\text{-°F} \times \frac{1.390 \pi \times 12 \text{ ft}^2}{144 \text{ ft}^2}}$$

$$= 37 \text{ F}$$

$$\Delta T_b = \frac{0.300 \times 172 \times 3413}{30,000 \times \frac{0.560 \pi \times 12}{144}} = 40 \text{ F}$$

XII. ΔT ACROSS THE CORE

Geering (9) has developed the following equation for calculating the temperature drop through a tubular fuel element:

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

Assume an average core temperature of 1050 F. Then $k = 21.9 \frac{\text{BTU}}{\text{hr-ft-°F}} = 0.201 \frac{\text{watt}}{\text{cm-°F}}$

$$r_o = \frac{1.390}{2} \text{ in} \times 2.54 \frac{\text{cm}}{\text{in}} = 1.763 \text{ cm}$$

$$r_b = \frac{0.560}{2} \text{ in} \times 2.54 \frac{\text{cm}}{\text{in}} = 0.711 \text{ cm}$$

$$\Delta T = \frac{688 \frac{\text{watt}}{\text{cm}^2}}{4 \times 0.201 \frac{\text{watt}}{\text{cm-°F}}} \left\{ \frac{(1.763)^2 - (0.711)^2}{\ln \frac{1.763}{0.711}} \ln \left(\frac{\sqrt{\frac{(1.763)^2 - (0.711)^2}{2 \ln \frac{1.763}{0.711}}}}{1.763} \right) + (1.763)^2 - \frac{(1.763)^2 - (0.711)^2}{2 \ln \frac{1.763}{0.711}} \right\}$$

$$= 485 \text{ F.}$$

(9) Geering, G. T., Op. Cit.

DECLASSIFIED

-17-

HW-66603
8/25/60

XIII. TEMPERATURE DISTRIBUTIONS

At Defect

	<u>Uncorrected Temperature</u>
Inlet Temperature	435 F
Maximum Coolant ΔT to Defect	<u>8</u>
Maximum local water temperature	443 F
Film ΔT	<u>114</u>
Cladding Surface Temperature	557 F

At Reactor Mid-Plane - From Outside

	<u>Uncorrected Temperatures</u>	<u>Corrected Temperature</u>
Inlet temperature	435 F	435 F
Maximum ΔT to reactor mid-plane	<u>42</u>	<u>49</u>
Maximum local water temperature	477 F	484 F
Film ΔT	137	202
ΔT Add	<u> </u>	<u>6</u>
Cladding Surface Temperature	614 F	692 F
Jacket ΔT	<u>174</u>	<u>174</u>
Bond layer surface temperature	788 F	865 F
Bond ΔT	<u>37</u>	<u>37</u>
Uranium surface temperature	825 F	903 F
Core ΔT	<u>485</u>	<u>485</u>
Core temperature	1310 F	1388 F

DECLASSIFIED

-18-

HW-66603
8/25/60

At Reactor Mid-Plane - From Bore

	<u>Uncorrected Temperatures</u>	<u>Corrected Temperatures</u>
Inlet temperature	435 F	435 F
Maximum ΔT to reactor mid-plane	<u>16</u>	<u>18</u>
Maximum local water temperature	451 F	453 F
Film ΔT	108	159
ΔT Add	<u> </u>	<u>6</u>
Cladding surface temperature	559 F	618 F
Jacket ΔT	<u>212</u>	<u>212</u>
Bond layer surface temperature	771 F	830 F
Bond ΔT	<u>40</u>	<u>40</u>
Uranium surface temperature	811 F	870 F
Core ΔT	<u>485</u>	<u>485</u>
Core temperature	1296 F	1355 F

XIV. TOTAL HEAT OUTPUT

$$P_t = Q_a \quad L = 99.3 \text{ kw/ft} \times 2 \text{ ft} = 198.6 \text{ kw} \times 3413 \text{ BTU/kw-hr}$$

$$= 678,000 \text{ BTU/hr.}$$

XV. BULK COOLANT ΔT

Assume an average bulk temperature of $435 + 18 = 453 \text{ F}$. Then $C_p = 1.108 \frac{\text{BTU}}{\text{lb-}^\circ\text{F}}$

$$\Delta T = \frac{P_t}{w_t C_p} = \frac{678,000 \text{ BTU/hr}}{16,800 \text{ lb/hr} \times 1.108 \text{ BTU/lb-}^\circ\text{F}} = 36.4 \text{ F}$$

Using the F_b correction $\Delta T = 36.4 \times 1.16 = 42.3 \text{ F}$.

XVI MAXIMUM SURFACE HEAT FLUXES

$$E = \frac{Q}{A}$$

$$E_o = \frac{0.700 Q_p}{A_o} = \frac{0.700 \times 172 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr}}{1.421 \text{ ft} \times 12 \frac{\text{ft}^2}{\text{ft}}}$$

$$= 1,100,000 \text{ BTU/hr-ft}^2$$

DECLASSIFIED

-19

104-66603
8/23/60

$$E_b = \frac{0.300 \times 1.72 \times 10^{13}}{\frac{0.526 \pi \times 12}{144}} = 1,270,000 \text{ BTU/hr-ft}^2$$

XVII. BURNOUT HEAT FLUXES

De Bartoli, et.al. (10) give the following equation for calculating burnout heat fluxes:

$$\frac{E_{BO}}{10^6} = 0.182 \left(\frac{H}{10^3} \right)^{-2.5} \left(1 + \frac{G}{10^7} \right)^2 e^{-0.0012 L/D_e}$$

For the annulus:

$$\left(\frac{H}{10^3} \right)^{-2.5} = \left(\frac{525}{10^3} \right)^{-2.5} = \frac{1}{0.200}$$

$$\left(1 + \frac{G}{10^7} \right)^2 = \left(1 + \frac{2.80 \times 10^6}{10^7} \right)^2 = 1.635$$

$$e^{-0.0012 \frac{L}{D_e}} = e^{-0.0012 \times \frac{2}{0.0140}} = \frac{1}{1.187}$$

$$E_{BO} = 0.182 \times \frac{1}{0.200} \times 1.635 \times \frac{1}{1.187} \times 10^6 = 1,260,000 \text{ BTU/hr-ft}^2$$

For the bore:

$$\left(\frac{H}{10^3} \right)^{-2.5} = \left(\frac{412}{10^3} \right)^{-2.5} = \frac{1}{0.109}$$

$$\left(1 + \frac{G}{10^7} \right)^2 = \left(1 + \frac{5.99 \times 10^6}{10^7} \right)^2 = 2.57$$

$$e^{-0.0012 \frac{L}{D_e}} = e^{-0.0012 \times \frac{2}{0.0438}} = \frac{1}{1.056}$$

$$E_{BO} = 0.182 \times \frac{1}{0.109} \times 2.57 \times \frac{1}{1.056} \times 10^6 = 4,060,000 \text{ BTU/hr-ft}^2.$$

(10) De Bartoli, R. A., et. al., Forced-Convection Heat Transfer Burnout Studies for Water in Rectangular Channels and Round Tubes at Pressures Above 500 psia, WAFD-188, October, 1958.

DECLASSIFIED

-20-

HW-66603
8/25/60

XVIII. FUEL WEIGHT

$$\text{Fuel weight per inch} = \frac{\pi}{4} \left[(1.390)^2 - (0.560)^2 \right] \frac{\text{in}^3}{\text{in}}$$
$$\times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{18.9 \text{ gm}}{\text{cm}^3} = 394 \text{ gm/in}$$

$$\text{Total weight} = 394 \text{ gm/in} \times 24 \text{ in} = 9,450 \text{ gm}$$

XIX. AVERAGE BURNUP RATE

$$\text{Average burnup rate} = \frac{P_t}{\text{wt. of fuel}}$$
$$= \frac{0.1986 \text{ MW}}{9.45 \text{ kg} \times \frac{2.2 \text{ lb}}{\text{kg}} \times \frac{\text{ton}}{2000 \text{ lb}}}$$
$$= 19.1 \frac{\text{MW}}{\text{ton}}$$

XX. FURTHER CALCULATIONS

The foregoing calculations were repeated at fuel element powers 60, 80, 120 and 140 per cent of the powers used above. The data thus obtained were used to plot the curves of Figure 1-5 and 10-15 inclusive. These calculations will not be repeated here. The pertinent data from which the curves were plotted is reproduced in the following tables. At powers 60 per cent and 80 per cent of those used above it was necessary to depress the inlet temperature a few degrees below the inlet temperature required to give a surface temperature at the defect of 556 F. This was done to keep the maximum surface heat flux below the burnout flux.

TABLE AI

ΔT °F	Inlet Temperature °F	Maximum Fuel Temperatures °F		Maximum Cladding Surface Temperature, °F			
		Uncorrected	Corrected for F_b and F_h	Uncorrected	Corrected for F_b and F_h	Uncorrected	Corrected for F_b and F_h
21.1	476	1010	1058	582	630	552	595
28.6	454	1150	1213	597	660	555	604
36.4	435	1310	1388	614	692	559	618
44.4	410	1416	1509	631	724	559	629
52.9	385	1520	1628	643	751	560	642

TABLE AII

ΔT °F	Maximum Specific Power Generation kw/ft	Water Velocity at Top of Fuel, kw/ft		Film Coefficients BTU/hr-ft ² -°F		Outlet Temperature °F
		Annulus	Bore	Annulus	Bore	
21.1	103	15.6	33.2	7960	11,340	497
28.6	138	15.3	32.5	7940	11,510	483
36.4	172	14.9	31.9	7910	11,600	471
44.4	207	14.6	31.2	7840	11,640	454
52.9	241	14.4	30.7	7780	11,670	438

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TABLE AIII

HW-66603

<u>ΔT</u> <u>°F</u>	<u>Maximum Surface Heat</u> <u>Flux - BTU/hr-ft²</u>		<u>Burnout Heat Flux</u> <u>BTU/hr-ft²</u>		<u>Average Burnup</u>
	<u>Annulus</u>	<u>Bore</u>	<u>Annulus</u>	<u>Bore</u>	<u>MW/ton</u>
21.1	655,000	762,000	656,000	1,590,000	11.5
28.6	875,000	1,020,000	882,000	2,500,000	15.3
36.4	1,100,000	1,270,000	1,260,000	4,060,000	19.1
44.4	1,320,000	1,520,000	1,820,000	6,520,000	22.9
52.9	1,540,000	1,780,000	2,330,000	10,900,000	26.8

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-23-

HW-66603

FIGURE 1

MAXIMUM CORE TEMPERATURE AS A FUNCTION OF ΔT ACROSS THE FUEL ELEMENTS

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FIGURE 2

MAXIMUM CLADDING SURFACE TEMPERATURE ON THE OUTER SURFACE OF THE FUEL ELEMENTS AS A FUNCTION OF ΔT ACROSS THE FUEL ELEMENTS

DECLASSIFIED

-25-

HW-66603

FIGURE 3

MAXIMUM CLADDING SURFACE TEMPERATURE IN THE BORE OF THE FUEL ELEMENTS AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

DECLASSIFIED

-26-

HW--66603

FIGURE 4

MAXIMUM SPECIFIC POWER GENERATION AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

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-27-

HW-66603

FIGURE 5

WATER VELOCITIES AT THE TOP OF THE FUEL ELEMENTS AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

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TUBE
O.D. 1.421
I.D. .526
CLADDING .020
LENGTH 12.5
END CAP $\frac{1}{4}$

FIG. 6
FUEL ASSEMBLY

FIGURE 7
The Fuel Elements for the Fifth Rupture Test Just After Autoclaving

FIG. 8
RUPTURE MECHANISM ASSEMBLY
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FIG. 9
RUPTURE EXPERIMENT
PORTABLE HYDRAULIC SYSTEM
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FIGURE 10

INLET TEMPERATURE (AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS)
REQUIRED TO MAINTAIN A SURFACE TEMPERATURE AT THE DEFECT OF 556 F

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FIGURE 11

OUTLET TEMPERATURE AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

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HW-66603

FIGURE 12

TOTAL HEAT OUTPUT OF THE FUEL AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

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FIGURE 13

MAXIMUM SURFACE HEAT FLUX AND BURNOUT FLUX IN THE ANNULUS AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

MAXIMUM SURFACE HEAT FLUX AND BURNOUT FLUX IN THE BORE AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

FIGURE 14

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FIGURE 15

AVERAGE BURNUP RATE AS A FUNCTION OF THE ΔT ACROSS THE FUEL ELEMENTS

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8/2/94

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