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AUTHOR

R. L. Call and

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PROPOSAL FOR CHARGING THE SIXTH RUPTURE FUEL ELEMENT

EXPERIMENT: GEH-10:

46, 47

R. L. Call and D. C. Kaulitz

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PROPOSAL FOR CHARGING THE SIXTH RUPTURE FUEL
ELEMENT EXPERIMENT: GEH-10:
46, 47

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I. OBJECTIVE

The objective of this test is to verify further the corrosion rate of irradiated tubular fuel elements under conditions of high specific power and central core temperatures. The fuel will be the inner component of an NPR fuel element assembly.

Permission to charge two tubular fuel elements into the GEH-P7 Loop is requested. The upper fuel element of the two has been irradiated to 1660 MWD/T in a Hanford reactor. A hydraulic cylinder will be attached to the upper fuel element. The hydraulic cylinder will be used to move a chisel along the outer surface of the upper fuel element and shear off a cap which has been welded over a hole through the cladding. The lower fuel element of the two will be used to add heat to the loop so that the desired loop water temperatures can be maintained.

II. SUMMARY OF TEST CONDITIONS

Maximum fuel temperature	-	See Figure 1
Maximum cladding surface temperature		
Outer surface		See Figure 2
Bore surface		See Figure 3
Weight of uranium, g/in		394
Total weight of uranium, g		9460
Weight of basket and cladding, g		1100
Maximum specific power generation		See Figure 4
Number of fuel assemblies		2
Water flow rate		See Figures 5, 6, and 7

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Average water velocity	See Figures 8 and 9
Water film coefficients	See Figures 10 and 11
Maximum loop pressure, psia	2000

III. DESCRIPTION OF THE FUEL ELEMENTS

Two fuel tubes, each one the inner component of an NPR fuel assembly, are proposed for this test. The upper element will be GEH-10-46 and the lower one GEH-10-47. The fuel elements were fabricated from natural uranium coextruded in Zircaloy-2 cladding. The cladding is a nominal 18 mils thick. The O.D. of the fuel elements is 1.428 inches and the I.D. is 0.528 inches. The fuel tubes are each 12.5 inches long overall, and the uranium length is 12.0 inches. The 1/4 -inch thick Zircaloy-2 end caps were welded into the counterbored ends of the tubes by the vacuum electron-beam welding process. The fuel tubes have supports spot-welded to their outer surfaces to center the tubes inside the basket.

The upper fuel tube has a 0.037-inch hole drilled through the outer cladding near one end. This hole has been covered by a cap which was vacuum electron-beam welded over it. The upper end of this tube has a coupling welded to it to which the hydraulically operated rupture mechanism will be attached. Both fuel elements have been autoclaved for 48 hours in 752 F, 1500 psi steam and the defected fuel element has been irradiated in 480 F water to an exposure of 1660 MWD/T.

Figure 12 is a cross-section drawing of the defected fuel tube in the basket. Figure 13 shows fuel element GEH-10-46 before irradiation. Fuel element GEH-10-47 is identical to GEH-10-46, except that it does not have a coupling welded to one end. Figure 14 shows the defected fuel element and the sleeve in which the fuel element was placed before it was charged into the reactor at Hanford. The four supports at either end of the sleeve were welded into the sleeve to hold the fuel element in the center of the sleeve and to keep it from sliding out of the sleeve during irradiation or during subsequent handling.

IV. DESCRIPTION OF THE RUPTURING MECHANISM

Details of the rupturing mechanism are shown in Figures 15 and 16. The same design will be used in this test as has been used in previous rupture tests. Loop pressure will hold the chisel in the retracted position until it is driven forward by the pump. The pump is used only to move the chisel forward and shear off the defect. Loop pressure returns the chisel to the retracted position.

V. DESCRIPTION OF THE FUEL BASKET

The basket is a 54-inch long Zircaloy-2 tube with an I.D. of 1.590 inches and a wall thickness of 35 mils. It has been orificed so that 7.7 per cent of the total loop flow will flow through the annulus between the

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basket and the process tube to cool these two components. The rest of the flow will pass through the basket to cool the fuel elements.

Two chromel-alumel thermocouples are located in the basket assembly so that they read the inlet and outlet temperature of the water in the basket. One thermocouple, color-coded yellow, is located about four inches down from the top of the basket; the other, color-coded red, is located at the bottom of the fuel charge. These thermocouples are swaged thermocouples and have a 1/16-inch O.D. stainless steel sheath with magnesia for insulation. The thermocouples and the 1/8-inch hydraulic line attached to the rupturing mechanism are encased in a 3/8-inch stainless steel tube. This tube is sealed against the process water by a silver solder junction which will be above the process tube cap when the experiment is installed in the reactor. Therefore, the silver solder junction will not be in contact with the high temperature loop water. It is requested that the two thermocouples be connected to appropriate continuous recording devices during the test. Before the test assembly is charged into the reactor the continuity and resistance of the thermocouples must be checked to insure that they will operate properly during the test.

VI. THERMAL AND HYDRAULIC TABLE

The following table specifies the required thermal and hydraulic conditions for the GEH-P7 loop during the test.

TABLE I

Reactor power level, MW	175
Loop flow rate	See Figures 5, 6, and 7
Loop pressure, psia	2000
Saturation temperature of loop water, °F	- 636
Coolant velocity past the fuel	See Figures 8 and 9
Heat transfer film coefficient	See Figures 10 and 11
Inlet water temperature	See Figure 17
Outlet water temperature	See Figure 18
Pressure drop across the fuel (at 60.6 gpm, cold)	10 psi

VII. THERMAL CHARACTERISTICS OF THE EXPERIMENT

The following table shows the thermal characteristics of the experiment. A sample heat transfer calculation is given in the Appendix.

TABLE II

Total heat output of the fuel	See Figure 19
Maximum surface heat flux and burnout flux	
Outer surface	See Figure 20
Bore surface	See Figure 21
Maximum cladding surface temperature	
Outer surface	See Figure 2
Bore surface	See Figure 3

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Maximum fuel core temperature
Average burnup rate

See Figure 1
See Figure 22

VIII. HAZARDS

Several tubular fuel elements have been irradiated successfully in Hanford reactors to high exposures in 480 F water. 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 of damage as a result of the test. Loop instrumentation is adequate to detect any potentially serious condition in time to shut down the reactor before damage occurs.

During this test the activity monitor will not be connected to the reactor scram circuit. The activity level scram will be manually controlled. Provision has been made to record coolant activity, temperature, flow, and pressure in the loop. Should any of these parameters reach 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 irradiated tubular fuel element will not be hazardous to either the reactor or the loop facility. Previous rupture tests performed on both irradiated and unirradiated fuel elements have demonstrated that the facility can safely handle the contamination and that the loop can be decontaminated.

IX. SPECIAL INSTRUMENTATION

The instruments on the D/N loop which measure gross gamma and delayed neutron activity will be used. The gamma scintillators which are located at various points through the loop will also be used.

X. TECHNICAL BASIS FOR THE PROPOSED TEST

The fifth rupture test of this series was performed to establish a base corrosion rate and failure delay time for NPR inner fuel components. This test will be performed to determine the effects of irradiation on these two properties.

XI. PROCEDURE FOR ASSEMBLING THE FUEL ELEMENTS AND BASKET

The fuel elements, basket, thermocouples, and hydraulic mechanism will be received at the ETR unassembled. Three irradiated, defected fuel elements will be shipped to the ETR in a cask. The unirradiated heater element(s) will be shipped separately. Two of the irradiated fuel elements were irradiated to 1660 MWD/T and one to 400 MWD/T. One of the 1660 MWD/T exposure fuel elements will be used in this test. The other will be used in this test if the first fuel element is damaged. The 400 MWD/T exposure fuel element will be used only if one of the other two is unsuitable for use for any reason.

Each irradiated fuel element is inside a Zircaloy-2 sleeve (See Figure 14).

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These sleeves will be placed inside aluminum shipping containers and these containers will be placed inside a shielded cask. The two 1660 MWD/T exposure fuel elements will be in one container and the third fuel element in the other. After the shipping containers have been removed from the cask they are to be opened and the Zircaloy-2 sleeves containing the fuel elements removed. One end is to be cut off one of the sleeves which contains a 1660 MWD/T exposure fuel element. Sawing by hand, rather than with a power saw, is recommended so that the supports under the end of the fuel element do not become jammed. When the end has been cut off the sleeve the fuel element can be pushed out. The fuel element will then be inspected for damage, and if acceptable will be inserted in the basket for charging into the reactor.

The basket is to be prepared for receiving the fuel elements by threading the downstream thermocouple (which will already be attached to the basket) through the quick disconnect fitting and through the 3/8-inch tube. The heater element will then be inserted into the basket and the basket will be placed in the canal on a platform approximately 11 feet under the water surface. The quick disconnect and the 3/8-inch tube will be kept on the edge of the canal.

The next step will be to place the defected fuel element in the assembly jig. Figure 23 shows the assembly jig and assembly tool used for the second and third rupture tests. Similar pieces of equipment, modified for use with the present fuel elements, will be provided. The rupturing mechanism with the hydraulic line attached will be locked into the assembly tool. The assembly tool will be engaged with the indexing post on the assembly jig and pushed down until the rupturing mechanism engages the coupling on the fuel element. The assembly tool will then be unlocked from the fuel element and the locking ring on the rupturing mechanism pushed into place.

The hydraulic line attached to the rupturing mechanism can be used as a handle for lifting the defected fuel element and rupturing mechanism from the assembly jig. The rupturing mechanism-fuel element assembly is to be inserted into the basket so that the defect cap is directly under the index mark scribed on the outside of the basket. The quick disconnect and the 3/8-inch tube will next be threaded over the hydraulic line, pushed down into place in the basket, and locked into place. A plug which contains slots for the thermocouples and hydraulic line is to be pushed into the end of the 3/8-inch tube and silver-soldered into place.

XII. 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. The test assembly will be transferred through the canal to the discharge chute and the thermocouples and hydraulic line threaded through the discharge chute into the reactor shell. The test assembly will then be lifted into the reactor shell and lowered into the process tube. The test assembly is seated properly when 6-1/2 inches of the 3/8-inch tube con-

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taining the hydraulic line and thermocouples remains above the process tube opening. The thermocouples and hydraulic line are to be threaded through the process tube cap and the cap seated on the tube. The Conax seal fittings are to be threaded onto the thermocouples and hydraulic line and the thermocouples and hydraulic line are to be threaded through the external thermocouple hole. The Gonax seals will be tightened and the thermocouples connected to the thermocouple bus. The hydraulic line is to be connected to the Black Hawk pump and valves, as shown in Figure 16.

XIII. TEST PROCEDURE

As soon as the reactor has reached full power, the loop outlet temperature is to be adjusted to the value determined from Figure 18. Figure 18 is based on a flow controller setting of 80 percent of full flow. The outlet temperatures of Figure 18 were calculated on the basis of maintaining a surface temperature of 556 F at the defect. After the reactor has been at full power for three hours, the defected fuel element is to be ruptured. The procedure below will be followed:

- 1) Check to see that valves 1, 2, and 3, shown in Figure 18, are closed.
- 2) Open valves 2 and 3 until a steady dripping of water comes from the drain line. This shows the hydraulic line is full of water.
- 3) Close Valve 2 and open Valve 1.
- 4) Operate the Black Hawk pump until water comes from the drain line.
- 5) Close Valve 3.
- 6) Operate the pump until the pressure as shown on the pressure gage builds up to 2200 psi.
- 7) Open Valve 2 and operate the pump rapidly until the pressure builds up to approximately 4000 lbs.
- 8) Close Valve 1 and open Valve 3. About 1/2 cup of water should run from the drain line to indicate that the chisel has been retracted.
- 9) Close Valves 2 and 3.

We request, as we have in the past, seven full days of reactor operation at full power after the fuel element has been ruptured. However, if 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

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loop piping high enough to cause a 40 R reading on the unshielded scintillator (scintillator No. 6) which is located in the corner of the GEH-P7 cubicle. The scintillator is to be calibrated prior to the test.

XIV. REMOVAL OF THE TEST ASSEMBLY

After the reactor top shielding has been removed, the thermocouples and hydraulic line are to be cut just above the end of the 3/8-inch tubing. The process tube, the 3/8-inch tube and the test assembly are to be removed from the process tube as a unit. The basket is to be inserted in a stainless steel shipping container, the 3/8-inch tube cut off just above the quick disconnect fitting, and the cap installed on the shipping container. The shipping container will then be removed from the reactor through the discharge chute.

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APPENDIX

SAMPLE HEAT TRANSFER CALCULATIONS

I. NOMENCLATURE

- A - Area (ft^2).
C_p - Specific heat (BTU/lb-°F).
D_e - Equivalent diameter (ft.).
D_o - Outer diameter of either the inner or outer fuel cladding (in).
D_i - Inner diameter of either the inner or outer fuel cladding (in).
e - The base of natural logarithms.
E - Surface heat flux (BTU/hr-ft²).
E_{BO} - Burnout heat flux (BTU/hr-ft²).
F_h - Hot spot factor.
F_b - Hot channel factor.
G - Mass velocity (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 (g/g-atom).
N - Avogadro's number (6.02×10^{23} atom/g-atom).
Q - Total power generation (kw or BTU/hr).
q - Specific power (kw/ft or 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 g/cm³).
 Σ_f - Macroscopic fission cross section (cm⁻¹).
 σ_f - Microscopic fission cross section (cm²).
 ϕ - Volume average neutron flux in the fuel (nv or neutrons/cm²- sec).
F - Volume flow (ft³/sec).

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

- a - Pertaining to the annulus between the basket and the fuel elements.
b - Pertaining to the bore of the fuel elements.
d - Located at or pertaining to the defect.
o - Pertaining to the annulus between the basket and the loop process tube.
p - Peak or maximum value.
t - Total.

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II. HOT SPOT AND HOT CHANNEL FACTORS⁽¹⁾

	<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			
1. Flow measurements)			
2. Temperature measurement)	1.03	1.03	1.4
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 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 (of F _b and F _h columns) or sum (of ΔT add column).	1.16	1.47	5.7

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

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III. CALCULATION OF FLUXES

$$\Sigma_f = \frac{\sigma_f \rho_N}{m} = \frac{4.2 \times 10^{-24} \text{ cm}^2/\text{nucleus} \times 18.9 \text{ g/cm}^3 \times 6.02 \times 10^{23} \text{ nuclei/g-atom}}{238 \text{ g/g-atom}} \\ = 0.200 \text{ cm}^{-1}.$$

The peak, axial, unperturbed flux in the GEH-P7 loop during Cycle 36 will be 1.47×10^{14} nv. ⁽²⁾ A flux depression factor of 0.279 has been calculated from the results of the Fifth Rupture Test which was run during Cycle 33. Because the fuel elements for the current test are very nearly identical to those of the fifth test, the above flux depression factor will be used in the following calculations. Therefore, the peak, axial, perturbed flux will be:

$$1.47 \times 10^{14} \text{ nv} \times 0.279 = 0.410 \times 10^{14} \text{ nv}.$$

The correction for horizontal peaking at the reactor midplane is 1.52. ⁽²⁾ Therefore, the absolute peak, perturbed flux will be:

$$\phi_p = 0.410 \times 10^{14} \text{ nv} \times 1.52 = 0.623 \times 10^{14} \text{ nv}.$$

The ratio of peak axial flux to average axial flux for this test will be 1.14. ⁽³⁾ Thus the average effective flux will be:

$$\phi(\text{ave}) = 0.410 \times 10^{14} \text{ nv} \times \frac{1}{1.14} = 0.360 \times 10^{14} \text{ nv}.$$

IV. POWER GENERATION

Let us assume that the total heat released per fission is 194 Mev. ⁽⁴⁾ Assume that 95 per cent of the heat is released within the fuel core. Then the heat generation in the core is:

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

The peak specific power will be:

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

(3) Calculated on the basis of a GEH-P7 Loop flux map of Cycle 3.

(4) Based on the example for a swimming pool reactor on p.642 of Principles of Nuclear Reactor Engineering, 1st Edition, D. Van Nostrand Co., Inc., Princeton, N. Jersey, by S. Glasstone.

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$$q_p = \text{heat/fission} \times \sum \phi_p V = 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.200 \frac{\text{fission}}{\text{cm-neutron}}$$

$$\times 0.623 \times 10^{14} \text{ neutron/cm}^2\text{-sec} = 0.368 \text{ watt/cm}^3.$$

$$0.368 \frac{\text{watt}}{\text{cm}^3} \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{\pi}{4} (1.392^2 - 0.564^2) \times 12 \frac{\text{in}^3}{\text{ft. of tube}} = 91.9 \frac{\text{kw}}{\text{ft. of tube}}.$$

The average specific power generation will be:

$$q(\text{ave}) = 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.200 \frac{\text{fission}}{\text{cm-neutron}} \times 0.360 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2\text{-sec}}$$

$$\times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{\pi}{4} (1.392^2 - 0.564^2) \times 12 \frac{\text{in}^3}{\text{ft. of tube}} = 53.0 \frac{\text{kw}}{\text{ft. of tube}}.$$

The ratio of flux at the defect to peak flux is 0.822⁽⁵⁾. 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.⁽⁵⁾ Therefore, the maximum specific power at the defect will be:

$$q_d = 91.9 \text{ kw/ft} \times 0.822 = 75.5 \text{ kw/ft.}$$

Let us assume that there is no mixing of the coolant as it flows past the fuel. Then, the equivalent peak total heat generation from the top of the fuel element to the defect along the maximum heat generation line will be:

$$Q_d = \frac{91.9}{1.14} \text{ kw/ft} \times 2 \text{ ft} \times 0.0985 = 15.9 \text{ kw.}$$

The mass of the process tube in the active zone

(5) Calculated on the basis of a GEH-P7 Loop flux map of Cycle 3.

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

The mass of the Zircaloy-2 basket in the active zone

$$= (1.654^2 - 1.590^2) \frac{\pi}{4} \times 36 \text{ in}^3 \times 0.237 \frac{\text{lb}}{\text{in}^3} \times \frac{1000 \text{ g}}{2.2 \text{ lb}} = 670 \text{ g.}$$

The mass of the Zircaloy-2 fuel element cladding

$$= (1.428^2 - 1.392^2 + 0.564^2 - 0.528^2) \frac{\pi}{4} \times 24 \text{ in}^3 \times 0.237 \frac{\text{lb}}{\text{in}^3} \times \frac{1000 \text{ g}}{2.2 \text{ lb}} = 280 \text{ g.}$$

The mass of the Zircaloy-2 end caps

$$= 4 \times \frac{\pi}{4} \times (1.428^2 - 0.528^2) \times 0.25 \text{ in}^3 \times 0.237 \frac{\text{lb}}{\text{in}^3} \times \frac{1000 \text{ g}}{2.2 \text{ lb}} = 150 \text{ g}$$

Total	8520 g
-------	--------

Assume a gamma heat generation of 1.1 watts per gram. Then the gamma heat generation in the loop will be:

$$8520 \text{ g} \times 1.1 \text{ watt/g} \times \text{kw}/1000 \text{ watt} = 9.4 \text{ kw,}$$

and the total heat generation in the loop (gamma plus fission will be:

$$Q_t = 530 \text{ kw/ft} \times 2 \text{ ft} + 9.4 \text{ kw} = 115.4 \text{ kw} = 394,000 \text{ BTU/hr.}$$

Gamma heat from the process tube and basket only will be

$$Q(\text{tube and basket}) = 8090 \text{ g} \times 1.1 \text{ watt/g} \times \text{kw}/1000 \text{ watt} = 8.9 \text{ kw.}$$

V. FLOW IN THE VARIOUS CHANNELS

Assume an inlet temperature of 504 F. Assume further that the flow controller is set at 80 per cent of full flow. At this temperature and setting the total flow through the loop will be 61.2 gpm. For water

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at 504 F and 2000 psia, $\rho = 49.2 \text{ lb/ft}^3$.⁽⁶⁾

$$w_t = 61.2 \text{ gal/min} \times 60 \text{ min/hr} \times 0.1337 \text{ ft}^3/\text{gal} \times 49.2 \text{ lb/ft}^3 = 24,100 \text{ lb/hr.}$$

Assume a bulk ΔT of 14 F. The average bulk temperature will then be 511 F. For water at 511 F and 2000 psia, $C_p = 1.173 \text{ BTU/lb-}^\circ\text{F}$, and

$$\Delta T(\text{bulk}) = \frac{Q_t}{w_t C_p} = \frac{115.4 \text{ kw} \times 3413 \text{ BTU/kw-hr}}{24,100 \text{ lb/hr} \times 1.173 \text{ BTU/lb-}^\circ\text{F}} = 13.9^\circ\text{F.}$$

In order to maintain a ΔT of 13.9 F for the flow around the outside of the basket with an average bulk temperature of 511 F, the required flow will be (assume that all the gamma heat from the process tube and basket goes to heat the water which flows around the basket):

$$w_o = \frac{Q(\text{tube and basket})}{C_p \Delta T} = \frac{8.9 \text{ kw} \times 3413 \text{ BTU/kw-hr}}{1.173 \text{ BTU/lb-}^\circ\text{F} \times 13.9^\circ\text{F}} = 1860 \text{ lb/hr.}$$

This gives us 7.7 per cent of the total flow around the basket and 92.3 per cent through it. We will assume that these percentages remain constant regardless of changes in flow and temperature. The flow through the basket will be: $24,100 \text{ lb/hr} - 1860 \text{ lb/hr} = 22,240 \text{ lb/hr}$.

The relative flows through the annulus and the bore can be calculated from the following equations:⁽⁷⁾

$$v_t A_t = F_t = v_a A_a + v_b A_b = F_a + F_b,$$

$$\frac{v_a^{1.8}}{D_{e_a}^{1.2}} = \frac{v_b^{1.8}}{D_{e_b}^{1.2}}.$$

(6) The values of C_p , k , u , and ρ used in these calculations were obtained from Tables 1.3.3, 1.3.5, and 1.3.6, and Figure 1.3.9 of The Reactor Handbook, Vol. 2, Engineering, 1st Edition, The Atomic Energy Commission, 1955.

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

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The annulus flow area = $\frac{\pi}{4} (1.590^2 - 1.428^2) \text{ in}^2 = 0.002645 \text{ ft}^2$.

The annulus wetted perimeter = $\pi(1.590 + 1.428) \text{ in} = 0.791 \text{ ft}$.

$$D_{e_a} = \frac{4 \times \text{flow area}}{\text{wetted perimeter}} = \frac{4 \times 0.002645 \text{ ft}^2}{0.791 \text{ ft}} = 0.01339 \text{ ft}.$$

The bore flow area = $\frac{\pi}{4} (0.525)^2 \text{ in}^2 = 0.001522 \text{ ft}^2$.

$$D_{e_b} = 0.528 \text{ in} = 0.0440 \text{ ft}.$$

If we rearrange the second of the above two flow equations we get:

$$v_a = v_b \left(\frac{D_{e_a}}{D_{e_b}} \right)^{2/3}.$$

Substituting the above values of D_e , we get:

$$v_a = v_b \left(\frac{0.01339}{0.0440} \right)^{2/3} = 0.452 v_b.$$

If we substitute this into the first of the two flow equations above, we get:

$$F_t = 0.452 v_b A_a + v_b A_b = 0.452 v_b \cdot 0.002645 + 0.001522 v_b.$$

$$v_b = 368.0 F_t.$$

$$v_a = 0.452 v_b = 0.452 \times 368.0 F_t = 166.5 F_t.$$

$$F_a = v_a A_a = 166.5 F_t \times 0.002645 = 0.440 F_t.$$

$$F_b = v_b A_b = 368.0 F_t \times 0.001522 = 0.560 F_t.$$

then

$$w_a = 22,240 \text{ lb/hr} \times 0.440 = 9,790 \text{ lb/hr},$$

$$w_b = 22,240 \text{ lb/hr} \times 0.560 = 12,450 \text{ lb/hr}.$$

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$$G_a = \frac{w_a}{A_a} = \frac{9790 \text{ lb/hr}}{0.002645 \text{ ft}^2} - 3.70 \times 10^6 \text{ lb/hr-ft}^2, \text{ and}$$

$$G_b = \frac{w_b}{A_b} = \frac{12,450 \text{ lb/hr}}{0.001552 \text{ ft}^2} - 8.03 \times 10^6 \text{ lb/hr-ft}^2.$$

VI. HEAT SPLIT

The method developed by Geering⁽⁸⁾ was used to calculate the heat split. The values 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 outward. The values chosen were 70 per cent to the outer surface and 30 per cent to the inner surface.

VII. MAXIMUM COOLANT ΔT 's

Assume an inlet temperature of 504 F. Assume further that no mixing of the water occurs as it flows along the fuel elements. Assume a coolant

ΔT of 3 F along the maximum heat generating line from the top of the first fuel element to the defect. The average coolant temperature along that line will be 506 F and $C_p = 1.165 \text{ BTU/lb-}^\circ\text{F}$.

$$\Delta T(\text{top to defect}) = \frac{0.7 Q_d}{w_a C_p} = \frac{0.7 \times 11.1 \text{ kw} \times 3413 \text{ BTU/kw-hr}}{9790 \text{ lb/hr} \times 1.165 \text{ BTU/lb-}^\circ\text{F}} = 3.3 \text{ }^\circ\text{F}.$$

Assume the average water temperature in the annulus along the maximum heat generation line between the top of the first fuel element and the reactor midplane is 512 F.

Then $C_p = 1.175 \text{ BTU/lb-}^\circ\text{F}$, and

$$\begin{aligned} \Delta T_a(\text{top to midplane}) &= \frac{0.7 \times 1.52 \times q(\text{ave}) \times L}{w_a C_p} = \frac{0.7 \times 1.52 \times 53.0 \text{ ft} \times 1 \text{ ft} \times 3413 \text{ kw-hr}}{9790 \text{ lb/hr} \times 1.175 \text{ BTU/lb-}^\circ\text{F}} \\ &= 16.7 \text{ }^\circ\text{F}. \end{aligned}$$

With F_b correction, $\Delta T_a(\text{top to midplane}) = 16.7^\circ\text{F} \times 1.16 = 19.4^\circ\text{F}$.

Assume the average water temperature in the bore of the fuel elements along the maximum heat generation line is 507 F. Then $C_p = 1.167 \text{ BTU/lb-}^\circ\text{F}$, and

$$\Delta T_b(\text{top to midplane}) = \frac{0.3 \times 1.52 \times q(\text{ave}) \times L}{w_b C_p} = \frac{0.3 \times 1.52 \times 53.0 \text{ ft} \times 3413 \text{ kw-hr}}{12450 \text{ lb/hr} \times 1.167 \text{ BTU/lb-}^\circ\text{F}}$$

(Footnote 8, see page 16) = 5.7 $^\circ\text{F}$.

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With F_b correction, ΔT_b (top to midplane) = $5.7^\circ\text{F} \times 1.16 = 6.6^\circ\text{F}$.

VIII. WATER FILM COEFFICIENTS

The Modified Colburn Equation⁽⁹⁾ is:

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

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

Assume a film temperature at the defect of 532 F. At this temperature and 2000 psia.

$$C_P = 1.210 \text{ BTU/lb-}^\circ\text{F},$$

$$k = 0.3414 \text{ BTU/hr-ft-}^\circ\text{F},$$

$$\mu = 0.243 \text{ lb/hr-ft}.$$

Then,

$$h_d = \frac{0.023 \times 0.3414 \frac{\text{BTU}}{\text{hr-ft-}^\circ\text{F}}}{0.01339 \text{ ft}} \left(\frac{0.01339 \text{ ft} \times 3.70 \times 10^6 \text{ lb/hr-ft}^2}{0.243 \text{ lb/hr-ft}} \right)^{0.8} \times \left(\frac{1.210 \text{ BTU/lb-}^\circ\text{F} \times 0.243 \text{ lb/hr-ft}}{0.3414 \text{ BTU/hr-ft-}^\circ\text{F}} \right)^{0.3} = 9910 \text{ BTU/hr-ft}^2 \text{ }^\circ\text{F}.$$

Assume a film temperature in the annulus at the reactor midplane of 550 F. Then,

$$C_P = 1.252 \text{ BTU/lb-}^\circ\text{F},$$

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

(9) Nertney, Op. Cit.

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

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$$k = 0.3318 \text{ BTU/hr-ft-}^{\circ}\text{F},$$

$$\mu = 0.235 \text{ lb/hr-ft, and}$$

$$h_a = \frac{0.023 \times 0.3318}{0.01339} \left(\frac{0.01339 \times 3.70 \times 10^6}{0.235} \right)^{0.8} \left(\frac{1.252 \times 0.235}{0.3318} \right)^{0.3}$$
$$= 9990 \text{ BTU/hr-ft}^2 - ^{\circ}\text{F}.$$

Assume a film temperature in the bore at the reactor midplane of 534°F . Then,

$$C_p = 1.213 \text{ BTU/lb } ^{\circ}\text{F},$$

$$k = 0.3403 \text{ BTU/hr-ft } ^{\circ}\text{F},$$

$$\mu = 0.242 \text{ lb/hr-ft, and}$$

$$h_b = \frac{0.023 \times 0.3403}{0.0440} \left(\frac{0.0440 \times 8.03 \times 10^6}{0.242} \right)^{0.8} \left(\frac{1.213 \times 0.242}{0.3403} \right)^{0.3}$$
$$= 14,530 \text{ BTU/hr-ft}^2 - ^{\circ}\text{F}.$$

IX ΔT 's ACROSS THE COOLANT FILM

$$\Delta T_d(\text{film}) = \frac{0.7 q_d}{h_d A_a} = \frac{0.7 \times 75.5 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr}}{9910 \text{ BTU/hr-ft}^2 - ^{\circ}\text{F} \times \frac{1.428 \pi \times 12 \text{ in}^2}{\text{ft}} \times \frac{\text{ft}^2}{144 \text{ in}^2}}$$
$$= 48.7 ^{\circ}\text{F}.$$

$$\Delta T_a(\text{film}) = \frac{0.7 q_p}{h_a A_a} = \frac{0.7 \times 91.9 \times 3413}{9990 \times \frac{1.428 \pi \times 12}{144}} = 58.9 ^{\circ}\text{F}.$$

With F_h correction, $\Delta T_a(\text{film}) = 58.9 ^{\circ}\text{F} \times 1.47 = 86.3 ^{\circ}\text{F}.$

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$$\Delta T_b(\text{film}) = \frac{0.3 q_p}{h_b A_b} = \frac{0.3 \times 91.9 \times 3413}{14,530 \times 0.528 \pi \times 12} = 46.8 \text{ } ^\circ\text{F}.$$

With F_h correction, $\Delta T_b(\text{film}) = 46.8 \text{ } ^\circ\text{F} \times 1.47 = 68.8 \text{ } ^\circ\text{F}.$

X. ΔT 's ACROSS THE CLADDING

Assume that at the reactor midplane the average temperature of the outer cladding at the peak power point is 635°F . At this temperature

$k = 8.17 \text{ BTU/hr-ft-}^\circ\text{F}$ for Zircaloy-2. ⁽¹¹⁾ Then,

$$\Delta T_a(\text{cladding}) = \frac{0.7 q_p \ln \frac{D_o}{D_i}}{2 \pi k} = \frac{0.7 \times 91.9 \frac{\text{kw}}{\text{ft}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}} \times \ln \frac{1.428}{1.392}}{2 \pi \times 8.17 \text{ BTU/hr-ft-}^\circ\text{F}}$$

$$= 113.7 \text{ } ^\circ\text{F}.$$

Assume that in the bore at the reactor midplane the average temperature of the inner cladding is 620°F . At this temperature $k = 8.17 \text{ BTU/hr-ft-}^\circ\text{F}$, and

$$\Delta T_b(\text{cladding}) = \frac{0.3 \times 91.9 \times 3413 \times \ln \frac{0.564}{0.528}}{2 \pi \times 8.17} = 120.4 \text{ } ^\circ\text{F}.$$

XI. ΔT 's ACROSS THE BOND LAYER

Assume a heat transfer coefficient for the cladding-core bond layers of $30,000 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$ ⁽¹²⁾.

Then,

$$\Delta T_a(\text{bond layer}) = \frac{0.7 q_p}{h A_a} = \frac{0.7 \times 91.9 \frac{\text{kw}}{\text{ft}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}}}{30,000 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F} \times \frac{1.392 \pi \times 12 \text{ in}^2 \times \text{ft}^2}{144 \text{ in}^2}}$$

$$= 20.1 \text{ } ^\circ\text{F}, \text{ and}$$

$$\Delta T_b(\text{bond layer}) = \frac{0.3 \times 91.9 \times 3413}{30,000 \times 0.564 \pi \times 12} = 21.2 \text{ } ^\circ\text{F}.$$

(11) Merckx, K. R., (Ed) Fuel Element Design Handbook, HW-51000, September, 1958, p. 302. IT.

(12) Ibid., p. A100-5.

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XII. ΔT ACROSS THE CORE

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

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

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

Assume an average core temperature at the peak heat generating point of 845 F. Then $k = 19.9 \text{ BTU/hr-ft-}^\circ\text{F}$ ⁽¹⁴⁾

$$= 0.1915 \text{ watt/cm-}^\circ\text{F.}$$

$$r_a = \frac{1.392}{2} \text{ in} = 1.77 \text{ cm.}$$

$$r_b = \frac{0.564}{2} \text{ in} = 0.716 \text{ cm.}$$

$$\Delta T(\text{core}) = \frac{368 \text{ watt/cm}^3}{4 \times 0.1915 \frac{\text{watt}}{\text{cm-}^\circ\text{F}}} \left[\frac{1.77^2 - 0.716^2}{\ln \frac{1.77}{0.716}} \times \ln \left(\frac{\frac{1.77^2 - 0.716^2}{2 \ln \frac{1.77}{0.716}}}{1.77} \right) + 1.77^2 - \frac{1.77^2 - 0.716^2}{2 \ln \frac{1.77}{0.716}} \right] \text{ cm}^2 = 274^\circ\text{F.}$$

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

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

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XIII. TEMPERATURE DISTRIBUTIONS

At the Defect

	<u>Uncorrected Temperature</u>
Inlet temperature	504 F
ΔT (top to defect)	<u>3</u>
Maximum local water temperature	507 F
ΔT_d (film)	<u>49</u>
Cladding surface temperature	556 F

At the Reactor Midplane - From the Annulus Inward

	<u>Uncorrected Temperature</u>	<u>Corrected Temperature</u>
Inlet temperature	504 F	504 F
ΔT_a (top to midplane)	<u>17</u>	<u>19</u>
Maximum local water temperature	521 F	523 F
ΔT_a (film)	59	86
ΔT add	<u> </u>	<u>6</u>
Cladding surface temperature	580 F	615 F
ΔT_a (cladding)	<u>114</u>	<u>114</u>
Bond layer temperature	694 F	729 F
ΔT_a (bond)	<u>20</u>	<u>20</u>
Uranium surface temperature	714 F	749 F
ΔT (core)	<u>274</u>	<u>274</u>
Core temperature	988 F	1023 F

At the Reactor Midplane - From the Bore Outward

	<u>Uncorrected Temperature</u>	<u>Corrected Temperature</u>
Inlet temperature	504 F	504 F

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Table (Cont'd.)	Uncorrected Temperature	Corrected Temperature
ΔT_b (top to midplane)	<u>6</u>	<u>7</u>
Maximum local water temperature	510 F	511 F
ΔT (film)	47	69
ΔT add	<u> </u>	<u>6</u>
Cladding surface temperature	557 F	586 F
ΔT_b (cladding)	<u>120</u>	<u>120</u>
Bond layer temperature	677 F	706 F
ΔT_b (bond)	<u>21</u>	<u>21</u>
Uranium surface temperature	698 F	727 F
ΔT (core)	<u>274</u>	<u>274</u>
Core temperature	972 F	1001 F

XIV. MAXIMUM SURFACE HEAT FLUX

$$E_a = \frac{0.7 q_p}{A_a} = \frac{0.7 \times 91.9 \text{ kw/ft} \times 3413 \text{ BTU/kw-hr}}{\frac{1.428 \pi \times 12 \text{ in}^2}{\text{ft}} \times \frac{\text{ft}^2}{144 \text{ in}^2}} = 587,000 \text{ BTU/hr-ft}^2.$$

$$E_b = \frac{0.3 q_p}{A_b} = \frac{0.7 \times 91.9 \times 3413}{\frac{0.528 \pi \times 12}{144}} = 681,000 \text{ BTU/hr-ft}^2.$$

XV. BURNOUT HEAT FLUX

De Bortoli, et.al.⁽¹⁵⁾ give the following equation for calculating burnout heat fluxes:

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

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

For the annulus:

$$\left(\frac{H}{10^3} \right)^{-2.5} = \left(\frac{513}{10^3} \right)^{-2.5} = \frac{1}{0.189},$$

$$\left(1 + \frac{G}{10^7} \right)^2 = \left(1 + \frac{3.70 \times 10^6}{10^7} \right)^2 = 1.877,$$

$$e^{-0.0012 L/D_e} = e^{-0.0012 \times 2 / 0.01339} = 0.836, \text{ and}$$

$$E_{BO} = 0.182 \times \frac{1}{0.189} \times 1.877 \times 0.836 \times 10^6 = 1.51 \times 10^6 \text{ BTU/hr-ft}^2.$$

For the bore:

$$\left(\frac{H}{10^3} \right)^{-2.5} = \left(\frac{500}{10^3} \right)^{-2.5} = \frac{1}{0.177},$$

$$\left(1 + \frac{G}{10^7} \right)^2 = \left(1 + \frac{7.93 \times 10^6}{10^7} \right)^2 = 3.22,$$

$$e^{-0.0012 L/D_e} = e^{-0.0012 \times 2 / 0.0440} = 0.947, \text{ and}$$

$$E_{BO} = 0.182 \times \frac{1}{0.177} \times 3.22 \times 0.947 \times 10^6 = 3.14 \times 10^6.$$

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XVI. FUEL WEIGHT

$$\text{Fuel weight} = \frac{\pi}{4} (1.392^2 - 0.564^2) \text{ in}^3 / \text{in} \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{18.9 \text{ g}}{\text{cm}^3}$$
$$= 394 \text{ g/in.}$$

$$\text{Total weight} = 394 \text{ g/in} \times 24 \text{ in} = 9,460 \text{ g.}$$

XVII. AVERAGE BURNUP RATE

$$\text{Average burnup rate} = \frac{q(\text{ave})L}{\text{weight of fuel}} = \frac{53.0 \frac{\text{kw}}{\text{ft}} \times 2 \text{ ft} \times \frac{\text{MW}}{1000 \text{ kw}}}{9460 \text{ g} \times \frac{\text{kg}}{1000 \text{ g}} \times 2.2 \frac{\text{lb}}{\text{kg}} \times \frac{\text{ton}}{2000 \text{ lb}}}$$
$$= 10.2 \text{ MW/ton.}$$

XVIII. FURTHER CALCULATIONS

The foregoing calculations were repeated at fuel element powers 60, 80, 120, 140, and 160 per cent of the power used above. The data thus obtained were used to plot the curves of Figures 1 to 11 (inclusive) and Figures 17 to 22 (inclusive). 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 at the defect of 556 F.

The equation used for calculating burnout heat flux is valid for values of enthalpy at the burnout point between 500 and 1000 BTU/lb. The enthalpy in the annulus at powers of 140 and 160 per cent of that used above and in the bore at powers 120, 140, and 160 per cent of that used above was less than 500--which is outside the range for which the equation is valid. However, the calculated burnout heat flux is so far above the maximum surface heat flux that there is obviously no danger that the surface heat flux is approaching the burnout point.

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FIGURE 1

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

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FIGURE 2

MAXIMUM CLADDING SURFACE TEMPERATURE ON THE OUTER SURFACE OF THE FUEL ELEMENTS AS A FUNCTION OF THE BULK WATER TEMPERATURE RISE IN THE LOOP

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FIGURE 3

MAXIMUM CLADDING SURFACE TEMPERATURE IN THE BORE OF THE FUEL ELEMENTS
AS A FUNCTION OF THE BULK WATER TEMPERATURE RISE IN THE LOOP

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FIGURE 4

MAXIMUM SPECIFIC POWER GENERATION AS A FUNCTION OF THE BULK WATER TEMPERATURE RISE IN THE LOOP

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FIGURE 5

TOTAL LOOP FLOW AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 50)

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FIGURE 6

FLOW IN THE ANNULUS BETWEEN THE BASKET AND THE FUEL ELEMENTS AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 80%)

FIGURE 7

FLOW IN THE BORE OF THE FUEL ELEMENTS AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 80%)

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FIGURE 8

AVERAGE WATER VELOCITY IN THE ANNULUS BETWEEN THE BASKET AND THE FUEL ELEMENTS AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 80%)

FIGURE 9

AVERAGE WATER VELOCITY IN THE BORE OF THE FUEL ELEMENTS AS A FUNCTION OF INLET TEMPERATURE (FLOW CONTROLLER SET AT 80%)

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

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

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FIG. 12
FUEL ASSEMBLY

FIGURE 13
Fuel Element GEH-10-46 before Irradiation

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FIGURE 14

Fuel Element GEH-10-46 and the Zr-2 Sleeve in which the Fuel Element was Irradiated

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FIG. 15
RUPTURE MECHANISM ASSEMBLY

FIG 16
RUPTURE EXPERIMENT
PORTABLE HYDRAULIC SYSTEM
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BULK WATER TEMPERATURE RISE - °F

FIGURE 17

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

BULK WATER TEMPERATURE RISE - °F

FIGURE 18

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

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FIGURE 19

TOTAL HEAT OUTPUT AS A FUNCTION OF THE BULK WATER TEMPERATURE RISE IN THE LOOP

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FIGURE 20

MAXIMUM SURFACE HEAT FLUX AND BURNOUT FLUX IN THE ANNULUS BETWEEN THE BASKET AND THE FUEL ELEMENTS AS A FUNCTION OF INLET TEMPERATURE

FIGURE 21

MAXIMUM SURFACE HEAT FLUX AND BURNOUT HEAT FLUX IN THE BORE OF THE FUEL ELEMENTS AS A FUNCTION OF INLET TEMPERATURE

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FIGURE 22

AVERAGE BURNUP RATE AS A FUNCTION OF THE BULK WATER TEMPERATURE RISE IN THE LCCP

ASSEMBLY TOOL

RUPTURE EXPERIMENT

FIGURE 23

The Assembly Tool and Jig Used in Previous Rupture Tests

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