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TITLE

PROPOSAL FOR CHARGING THE THIRD RUPTURE
FUEL ELEMENT EXPERIMENT, GEH-10: 24, 25,
26. GEH 12-16, 17, 18

AUTHOR

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PROPOSAL FOR CHARGING THE THIRD RUPTURE

FUEL ELEMENT EXPERIMENT, GEH-10: 24, 25, 26

R. L. Call and D. C. Kaulitz

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PROPOSAL FOR CHARGING THE THIRD RUPTURE
FUEL ELEMENT EXPERIMENT. GEH-10: 24, 25, 26

R. L. Call and D. C. Kaulitz

I. OBJECTIVE

The objective of this irradiation is to verify the corrosion rate of a cluster-type fuel element under conditions of high specific power and central core temperatures. The test will also be used in the development of rupture detection instrumentation and decontamination procedures as a necessary part in the development of the NPR. In this test a fuel element irradiated to 3200 MWD/T will be ruptured, the rate of rupture product released will be determined and the gamma spectrum from fission products released into the coolant will be observed.

Permission is requested for charging three 7-rod cluster fuel elements (one previously irradiated to 3200 MWD/T at the Hanford project, the other two unirradiated) into the GEH-R33P7 loop of the ETR. The irradiated element will have attached to it a hydraulic mechanism for opening a defect in one of its fuel rods. The other two elements are to serve as heaters to enable the loop to operate at desired temperatures.

II. SUMMARY OF TEST CONDITIONS

Weight of Uranium, gm/in of element	-	481.35
Total Weight of Uranium, gm.		
Enriched Element	-	4813.5
Natural Elements	-	8086.7
Total for all Elements	-	12900.2
Weight of Zircaloy-2 in Cladding, Supports & Basket in Active Zone, gm.	-	2740
Number of Fuel Elements	-	3
Number of Cycles	-	1
Maximum Loop Pressure, psia	-	2000

The following conditions apply to the initial three-day operating period, only.

Maximum Fuel Temperature, °F(°C)	-	1323(718)
Maximum Surface Temperature, °F(°C)	-	505(263)
Maximum Power Generation, kw/ft		
Enriched Element	-	178 (14,800 watt/in)
Natural Elements	-	100 (8,330 watt/in)
Water Flow Rate, gpm	-	55
Water Velocity ft/sec.	-	13.1
Film Coefficient, BTU/hr-ft ² -°F	-	5310

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The following conditions apply to the period of operation immediately prior to and following the shearing off of the defect cap.

Maximum Fuel Temperature, °F(°C)	-	1245 (675)
Maximum Surface Temperature, °F(°C)	-	689 (365)
Maximum Power Generation, kw/ft		
Enriched Element	-	124(10,300 watt/in)
Natural Elements	-	70(5,830 watt/in)
Water Flow Rate, gpm	-	70
Water Velocity, ft/sec	-	16.8
Film Coefficient, BTU/hr-ft ² -°F	-	8160

III. DESCRIPTION OF FUEL ELEMENTS

Three fuel elements are proposed for this irradiation. Two of them are natural uranium and the third is enriched to 1.6 per cent U-235. The fuel elements are 7-rod clusters fabricated from 0.593 inch OD rods coextruded in Zircaloy-2 jackets 30 mils thick. The enriched element has an over-all length of 11.6 inches (10.0 inches of uranium), not including the fittings for attaching the rupturing mechanism. See Figure 1. The two natural elements are each 10.0 inches in length over-all (8.4 inches of uranium). See Figure 2. The end caps are Zircaloy-2 and are welded to the ends of the rods by the vacuum electron-beam welding process. The caps are slotted to receive triangular end supports, (See Figures 1, 2, 3, 4, and 5) which are welded in place by an inert arc welding process. Inter-rod spacing is 0.055 inches. The element containing the defect has an additional end fitting welded to it for attaching the hydraulically operated rupturing mechanism. All three elements have been autoclaved for at least 72 hours in 752 F(400 C) steam and the enriched element was irradiated at HAPO to an exposure of 3200 MWD/T.

The enriched element has an intentional defect on one of its rods as shown in Figures 1, 3, 4, and 5. The two natural elements are used only for adding heat to the loop so that the desired temperatures can be obtained. The element with the defect will be placed at the top of the reactor core. The other two elements will be placed directly below it.

IV. DESCRIPTION OF RUPTURE MECHANISM

Details of the rupturing mechanism are shown in Figures 3 and 6. A schematic drawing of the entire hydraulic system is shown in Figure 7. Friction and the loop pressure will hold the chisel in the retracted position until it is required. The Black Hawk jack is used to move the piston and chisel forward to shear off the defect cap. Loop pressure returns the piston to the retracted position.

This unit has been operated successfully in recirculating out-of-reactor loops at pressure and temperature comparable to those of this test. No malfunctioning of the mechanism was observed. The rupture mechanism operated

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successfully in the first rupture test of this series; however, the time in the reactor was not as long as it will be for this test. The ex-reactor loop tests were performed with a coextruded Zircaloy-2 clad fuel element containing a defect as described above. The element was operated for three days in 572 F(300 C) and 2000 psia water prior to rupturing it. After the test, the piston packing and other parts of the rupture mechanism were examined and no damage observed. The piston packing is Garlock Chevron ring No. 530, a combination of graphite and asbestos designed for high temperature application.

V. DESCRIPTION OF THE FUEL BASKET ASSEMBLY

The basket, thermocouples, fuel elements, etc. are shown assembled in Figures 3 and 10. The basket was fabricated from 30 mil wall Zircaloy-2 tubing and is 54-1/4 inches long with an average inside diameter of 2.010 inches. An orifice welded in the downstream end is designed to permit 65 gpm flow through the basket past the fuel. Five gpm will flow around the basket to cool the process tube and basket. The foregoing conditions apply to loop operation at high (527 F) inlet temperatures. Lesser values will apply at reduced inlet temperatures.

Immediately upstream from the orifice is a spacer two inches in length. The three fuel elements and the rupture mechanism are located above the spacer.

Two chromel-alumel thermocouples located so that they read the temperature difference across the fuel are included in the basket assembly.

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 insulation inside it. The thermocouples and the 1/8-inch hydraulic line attached to the rupture 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 joint will not be in contact with the 520 F loop water. It is requested that the two thermocouples be connected to appropriate continuous recording devices during the test. Potentiometer checks must also be made prior to charging the test assembly into the reactor to assure accurate readings. Continuity must also be checked by appropriate resistance measurements.

VI. EX-REACTOR DEFECT TESTS

A discussion of the ex-reactor defect tests is given in the proposal for charging the unirradiated defect test and will not be repeated here.(1)

VII. THERMAL AND HYDRAULIC TABLE

Table I, on page 4, specifies the required thermal and hydraulic conditions for the reactor loop facility for the initial three-day period of the test. Table II, on page 4, specifies the conditions for the time immediately prior to and following the shearing off of the defect cap.

(1) Kaulitz, D. C., Proposal for Charging Rupture Fuel Element Experiment
GEH 10-9, HW-61762, September 2, 1959, pp. 4-6.

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TABLE I

Reactor Power Level, MW	-	115
Loop Flow Rate, gpm	-	55
Loop Pressure, psia	-	2000
Saturation Temperature of Loop Water, °F(°C)	-	636(335)
Coolant Velocity Past Fuel, ft/sec	-	13.1
Heat Transfer Film Coefficient, BTU/hr-ft ² -°F	-	5310
Inlet Water Temperature, °F(°C)	-	175(79)
Outlet Water Temperature (applying F_b correction to bulk ΔT), °F(°C)	-	218(103)
Pressure Drop Across the Fuel at 55 gpm, psi	-	11

TABLE II

Reactor Power Level, MW	-	80
Loop Flow Rate, gpm	-	70
Loop Pressure, psia	-	2000
Saturation Temperature of Loop Water, °F(°C)	-	636(335)
Coolant Velocity Past Fuel, ft/sec	-	16.8
Heat Transfer Film Coefficient, BTU/hr-ft ² -°F	-	8160
Inlet Water Temperature, °F(°C)	-	533(278)
Outlet Water Temperature (applying F_b Correction to bulk ΔT), °F(°C)	-	556(291)
Pressure Drop Across the Fuel at 70 gpm, psi	-	11

VIII. THERMAL CHARACTERISTICS OF THE EXPERIMENTS

The following tables show the thermal characteristics of the experiment as detailed by the calculations in Appendix I of this document. Table III applies to the initial three-day operating period and Table IV to the period immediately prior to and following the shearing off of the defect cap.

TABLE III

Heat Output of the Fuel, BTU/hr	-	772,000
Maximum Surface Heat Flux, BTU/hr-ft ²	-	851,000
Maximum Cladding Surface Temperature, °F(°C)	-	505(263)
Maximum Fuel Core Temperature, °F(°C)	-	1323(718)
Average Burnup Rate, MW/T		
Enriched Element	-	21.9
Natural Elements	-	12.4

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TABLE IV

Heat Output of the Fuel, BTU/hr	-	538,000
Maximum Surface Heat Flux, BTU/hr-ft ²	-	594,000
Maximum Cladding Surface Temperature, °F(°C)	-	689(365)
Maximum Fuel Core Temperature, °F(°C)	-	1245(675)
Average Burnup Rate, MW/T		
Enriched Element	-	15.3
Natural Elements	-	8.6

IX. HAZARDS

Several 7-rod cluster fuel elements have been irradiated successfully in Hanford reactors to exposures of 3,000 MWD/T in 455 F(235 C) water. Elements of the same type have been irradiated in the GEH R3X3P7 facility. The ex-reactor corrosion testing of defected coextruded rods has shown that the corrosion rate of the coextruded material is markedly less than that of unbonded rods. The type of failure proposed herein should not reduce the coolant flow through the loop appreciably.

All parts of the GEH R3X3P7 loop are fully instrumented and connected to the reactor scram circuit. For this test, the activity monitor will not be connected to the 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 scrambled before any damage to the reactor or loop components could occur.

The intentional rupture of a 7-rod cluster fuel element should not be hazardous to the facility or to the reactor. The loop has been modified to handle such a rupture. The appropriate loop valves have been modified for remote operation and provisions have been made for decontaminating the system after the ruptured element has been removed. The decontamination procedure, which is included as Appendix IV of this document, has been used with excellent results for decontaminating heavily contaminated out-of-reactor loops.

Calculations have been made as to the activity levels which might exist in the loop control cubicle after the fuel element has been ruptured. For these calculations it was assumed that fission products with an infinite irradiation exposure were in the loop piping so as to cause a reading of 40 R/hr on the eight-inch, 240 lb/ft³ concrete wall which has four inches of lead shielding outside it. With these assumptions as a basis, the level at the surface of the lead shielding was calculated to be from one to 10 mR/hr.(2)

A further calculation has been made regarding the Sr-90 activity of the loop water. In this calculation it was assumed that 50 gm of uranium was dispersed in the loop, all of which ended up in hot waste. It was further assumed that the exposure of the fuel element was 3000 MWD/T. On the basis of these assumptions, the Sr-90 activity level was calculated to be 0.53 curies.(3)

- (2) Personal Communication from J. L. Weeks, Jr.
(3) Personal Communication from R. B. Hall.

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Exhaustive tests have demonstrated that sudden contact between 570 F(300 C) water and heated uranium does not give rise to increased corrosion rates beyond what might be expected with water at the above temperature and relatively cool uranium. The tests have also demonstrated that the protection provided by Zircaloy-2 and the associated Zircaloy-2-to-uranium bond which is produced in the coextrusion process localizes corrosion effectively and actually prevents dispersion of uranium oxide during a considerable induction period following a rupture. The ability of the GEH-R33P7 loop to sustain massive damage to an irradiated specimen without damage to the loop or difficulty in specimen removal has already been demonstrated. We are, therefore, confident that this test can be carried out successfully.

X. SPECIAL INSTRUMENTATION

One of the most important features of the loop is the ability of the special instruments to gather information pertaining to the types and magnitude of radioactivity that contaminate a coolant stream in which a fuel element is ruptured. These instruments will measure gross gamma and delayed neutron activity and will also analyze the gamma energy spectrum. Information from these instruments will help in developing a better fuel element failure detector. The instrument heads or detectors for these devices are located on the secondary loop which is installed between the 3X3 and 6X9 loop cubicles. The secondary loop will give the holdup time of 45 to 60 seconds which is required for the activity detectors.

The primary loop also has monitors installed at six points for measuring gamma activity. These instruments are sensitive enough to give accurate measurements of activities ranging from 100 mr/hr to 100 R/hr.

R. Gilmore and J. W. Green of the Instrument Development Operation, Irradiation Processing Department, HAPD, will be available to consult with Phillips Petroleum Company personnel before the test. They will be able to answer questions that may arise concerning the operation of these instruments.

XI. TEST PROCEDURE

The procedures for assembling the experiment and for inserting it into and removing it from the reactor are given in Appendices II and III respectively.

In the cycle during which the test assembly is in the reactor, the reactor is to operate at the power level indicated in Table I for three days. Loop inlet temperature will also be as indicated in Table I. Then just prior to the time the fuel element is to be ruptured, the reactor power level is to be reduced to the level shown in Table II and loop inlet temperature is to be raised to the value shown in Table II. When the conditions of Table II have been met, the fuel element may be ruptured by following the procedure given below:

- 1) Check to see that valves 1, 2, and 3 shown in Figure 7 are closed.
- 2) Operate the Blackhawk pump until the pressure gage indicates a

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pressure equal to loop pressure.

3. Open valves 1 and 2.
4. Operate the pump again to move the piston in the rupture device. When the pressure gage shows a sudden, sharp increase in pressure, the piston has traveled to its limit and the rupture cap has been sheared off.
5. Close valve 1 and open valve 3 to bleed off pump pressure.
6. Close valve 2 and open valve 1 slightly to bleed off pressure and retract the rupture mechanism.
7. Close valves 1 and 3.

After the fuel element has been ruptured, the reactor is to continue operation for not more than three hours at the power level and loop inlet temperature specified in Table II. The reactor is then to be shut down and the test assembly discharged. However, if prior to the expiration of the above three-hour period the activity level as read by the unshielded scintillator located in the corner of the 3X3 loop cubicle exceeds 40 R/hr, the reactor is to be shut down immediately. In the event the 40R limit is not reached, the reactor should be scrammed three hours after the fuel element is ruptured. The loop will then be cooled and depressurized and the fuel elements removed.

The scintillator head for the unshielded scintillator is calibrated to the instrument on the 3X3 loop panel at 4.4 R for two minor divisions or 44 R for 20 minor divisions. This calibration should be checked prior to the test.

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APPENDIX I

HEAT TRANSFER CALCULATIONS

Nomenclature

- A = area (ft²).
cp = specific heat (BTU/lb-°F).
D_e = equivalent diameter (ft.).
D_i = inner diameter (in.).
D_o = outer diameter (in.).
G = mass velocity (lb/hr-ft²).
h = heat transfer coefficient (BTU/hr-ft²-°F).
k = thermal conductivity (BTU/hr-ft-°F).
m = mass (gm/gm-atom).
N = Avogadro's number (6.02 x 10²³ atoms/gm-atom).
q = total heat generation in the fuel (kw or BTU/hr).
Q = specific power (kw/ft of element).
Q' = specific power (kw/ft of rod).
T = temperature (°F).
V = specific volume of fuel rod (in³ /ft of rod).
w = mass flow rate (lb/hr).
μ = viscosity (lb/hr-ft).
ρ = density (lb/ft³ or gm/cm³)
Σ_f = macroscopic cross section (cm⁻¹)
σ_f = microscopic cross section (cm²)
φ = volume average flux in fuel (nv or neutrons/cm²-sec)
φ_p = peak flux in fuel (nv or neutrons/cm²-sec)

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I. Calculation of Heat Generation in the Fuel

Macroscopic Cross Section:

For Natural Uranium:

$$\begin{aligned}\Sigma_f &= \frac{C_f \cdot \rho_N}{m} \\ &= \frac{4.2 \times 10^{-24} \frac{\text{cm}^2}{\text{nucleus}} \times 18.8 \frac{\text{gm}}{\text{cm}^3} \times 6.02 \times 10^{23} \frac{\text{nuclei}}{\text{gm-atom}}}{\frac{238 \text{ gm}}{\text{gm-atom}}} \\ &= 0.200 \text{ cm}^{-1}\end{aligned}$$

For 1.6% enriched uranium:

$$\Sigma_f = 0.200 \text{ cm}^{-1} \times \frac{1.6}{0.72} = 0.444 \text{ cm}^{-1}$$

Flux Calculations:

The axial peak unperturbed flux in the GEH-R33P7 loop at full reactor power is $1.63 \times 10^{14} \text{ nv.}$ ⁽¹⁾ The horizontal correction factor to be applied to obtain the absolute peak flux is 1.52.⁽²⁾ Therefore, the peak unperturbed flux is:

$$1.63 \times 10^{14} \times 1.52 = 2.48 \times 10^{14} \text{ nv.}$$

The flux depression factor obtained from the first rupture test, GEH-12-9,10,11, was 0.516.⁽¹⁾ The peak perturbed flux will be:

$$= 2.48 \times 10^{14} \times 0.516 = 1.279 \times 10^{14} \text{ nv.}$$

Heat Generation:

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

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

The peak specific power generated in the natural uranium will be:

- (1) Personal Communication from J. H. Johnson.
- (2) Personal Communication from R. Neidner.
- (3) Based on the example for the swimming pool reactor on p. 642 of Principles of Nuclear Reactor Engineering, 1st Edition, by S. Glasstone.

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$$\begin{aligned}
 Q' &= \text{heat/fission} \times \Sigma_f \phi V \\
 &= 184 \text{ Mev/fission} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.200 \frac{\text{fission}}{\text{cm-neutron}} \\
 &\quad \times 1.279 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2\text{-sec}} \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{(.533)^2 \pi \times 12 \text{ in}^3}{4 \text{ ft-rod}} \\
 &= 33.1 \text{ kw/ft-rod.}
 \end{aligned}$$

An analysis of heat generation data for 7-rod cluster elements irradiated in the KER loops at the 105 KE reactor indicates that 1.6 percent enriched 7-rod cluster fuel elements will generate 1.5 to 1.7 times the heat generated by natural 7-rod cluster fuel elements under the same flux conditions. If an additional flux depression factor of 0.8 (in addition to the one used above) is applied to the heat generation calculations for the 1.6 percent enriched element of this test, the ratio of specific power in the natural clusters to that in the enriched is 1.78. This appears to be a conservative estimate and will be used in the following calculations. The peak specific power in the 1.6 percent elements will then be:

$$\begin{aligned}
 Q' &= 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.444 \frac{\text{fission}}{\text{cm-neutron}} \\
 &\quad \times 1.279 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2\text{-sec}} \times 0.8 \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{(.533)^2 \pi \times 12 \text{ in}^3}{4 \text{ ft-rod}} \\
 &= 58.7 \frac{\text{kw}}{\text{ft-rod}}.
 \end{aligned}$$

It can be shown by calculations which will not be reproduced here, that at full reactor power the 1.6 percent enriched fuel element will exceed the beta phase transformation temperature of uranium which is 1220 F (660 C). Operation above this temperature is undesirable, particularly because of the decrease in density and increase in volume which uranium undergoes when it changes from alpha to beta phase. Such an increase in volume could overstrain the Zircaloy-2 cladding and cause the fuel element to rupture ahead of schedule. In order to provide a safe margin below the beta phase transformation temperature the maximum calculated core temperature should not exceed 1170 F (632 C). It can be shown that if the loop inlet temperature is maintained at 175 C it will be necessary to reduce reactor power to 66 percent of full power level. During the initial three day period of operation, a loop inlet temperature of 175 F will be acceptable. It can also be shown that the enriched fuel element will be the limiting fuel element. Therefore, further calculations will be made for the enriched element only. The adjusted value for peak specific power for the enriched fuel element is then:

$$Q' = 58.7 \text{ kw/ft-rod} \times 0.66 = 38.7 \frac{\text{kw}}{\text{ft-rod}}$$

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II. Average Water Film Coefficient

The Modified Colburn Equation⁽⁴⁾ is:

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

where $()_f$ indicates that all physical properties within the parentheses are evaluated at the film temperature.⁽⁵⁾ The equation can be rewritten as follows:

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

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

$$\begin{aligned} \text{flow area} &= \frac{\pi (D \text{ tube})^2}{4} - \frac{7\pi (D \text{ rod})^2}{4} \\ &= \frac{\pi (2.010)^2}{4} - \frac{7\pi (0.593)^2}{4} \\ &= 0.3946 \pi \text{ in}^2 = 0.00274 \pi \text{ ft}^2 \end{aligned}$$

$$\begin{aligned} \text{wetted perimeter} &= \pi D \text{ tube} + 7 \pi D \text{ rod} \\ &= \pi (2.010) + 7 \pi (0.593) \\ &= 6.161 \pi \text{ in} = 0.513 \pi \text{ ft} \end{aligned}$$

$$De = \frac{4 (0.00274 \pi \text{ ft}^2)}{0.513 \pi \text{ ft}} = 0.0214 \text{ ft}$$

During the rupture period the bulk inlet temperature will be 533 F. The density of water at 533 F is 47.3 lb/ft³. We will assume that at this temperature 70 gpm total will flow through the loop of which 65 gpm will flow through the basket and 5 gpm around it. We will also assume that the mass flow rate, W, remains constant regardless of inlet temperature changes. Then

$$W = 65 \frac{\text{gal}}{\text{min}} \times \frac{60 \text{ min}}{\text{hr}} \times \frac{0.1337 \text{ ft}^3}{\text{gal}} \times 47.3 \frac{\text{lb}}{\text{ft}^3} = 24,600 \frac{\text{lb}}{\text{hr}}$$

and

$$G = \frac{W}{A} = \frac{24,600 \text{ lb/hr}}{0.00274 \pi \text{ ft}^2} = 2.86 \times 10^6 \frac{\text{lb}}{\text{hr-ft}^2}$$

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

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

During the initial three-day operating period the loop inlet temperature will be 175 F which, as can be seen from later calculations, will give a value of 275 F for the film temperature at the location of the hottest part of the fuel. The properties of water at 275 F and 2000 psia are:

$$\mu = 0.51 \text{ lb/hr-ft.}$$

$$\rho = 58.2 \text{ lb/ft}^3.$$

$$K = 0.400 \text{ Btu/hr-ft-}^\circ\text{F}$$

$$C_p = 1.010 \text{ Btu/lb-}^\circ\text{F.}$$

Then

$$h = \frac{0.023 \times 0.400 \text{ Btu/hr-ft-}^\circ\text{F}}{0.0214 \text{ ft}} \left(\frac{0.0214 \text{ ft} \times 2.86 \times 10^6 \frac{\text{lb}}{\text{hr-ft}^2}}{0.51 \frac{\text{lb}}{\text{hr-ft}}} \right)^{0.8} \left(\frac{1.010 \frac{\text{Btu}}{\text{lb-}^\circ\text{F}} \times 0.51 \frac{\text{lb}}{\text{hr-ft}}}{0.400 \frac{\text{Btu}}{\text{hr-ft-}^\circ\text{F}}} \right)^{0.3}$$

$$= 5310 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$$

III. Maximum ΔT Across Coolant Film

$$\Delta T = \frac{Q'}{hA}$$

$$= \frac{38.7 \frac{\text{kw}}{\text{ft-rod}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}}}{5310 \frac{\text{BTU}}{\text{hr-ft}^2\text{-}^\circ\text{F}} \times \frac{(\pi)(.593)\text{ft}^2}{12 \text{ ft-rod}}}$$

$$= 160 \text{ F (89 C).}$$

With F_h correction:

$$\Delta T = 160 \text{ F} \times 1.89 = 302 \text{ F (168 C).}$$

IV. Maximum ΔT Across the Jacket

$$\Delta T = \frac{Q' \ln \frac{D_o}{D_i}}{2\pi k}$$

$$k \text{ for Zr-2 at } 500 \text{ C} = 8.15 \frac{\text{BTU}}{\text{hr-ft-}^\circ\text{F}} \quad (6)$$

$$\Delta T = \frac{38.7 \frac{\text{kw}}{\text{ft}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}} \times \ln \frac{.593}{.533}}{2\pi \times 8.15 \frac{\text{BTU}}{\text{hr-ft-}^\circ\text{F}}}$$

= 275 F (153 C).

V. Maximum ΔT Across the Bond Layer

The bond layer conductance can be assumed to be 30,000⁽⁷⁾ BTU/hr-ft²-°F.
Then:

$$\Delta T = \frac{Q'}{hA} = \frac{38.7 \frac{\text{kw}}{\text{ft-rod}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}}}{30,000 \frac{\text{BTU}}{\text{hr-ft}^2\text{-}^\circ\text{F}} \times \frac{\pi (.533)^2 \text{ft}^2}{12 \text{ft-rod}}}$$

= 32 F (18 C).

VI. Maximum ΔT Across the Fuel Core

$$\Delta T = \frac{272 Q'}{k} \quad (8)$$

$$k \text{ for uranium at } 925 \text{ F} = 20.6 \frac{\text{BTU}}{\text{hr-ft-}^\circ\text{F}}$$

$$\Delta T = \frac{272 \times 38.7}{20.6} = 511 \text{ F (284 C)}.$$

VII. Total Heat Output

The ratio of unperturbed peak axial flux in the GEH-R33P7 loop to its average unperturbed axial flux is 1.28.⁽⁹⁾ The average, perturbed, axial flux will therefore be:

$$\phi = 1.63 \times 10^{14} \text{ nv} \times \frac{0.516}{1.28} = 0.657 \times 10^{14} \text{ nv}.$$

$$q = \text{heat/fission} \times \phi V.$$

For the natural uranium fuel elements:

$$q = 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.200 \frac{\text{fission}}{\text{cm-neutron}} \times 0.657 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2\text{-sec}} \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times \frac{7 (.533)^2 \pi}{4} \times 16.8 \text{ in}^3$$

= 166 kw at full reactor power.

At 66 percent of full reactor power

$$q = 166 \text{ kw} \times 0.66 = 110 \text{ kw} = 376,000 \text{ BTU/hr}.$$

(7) Ibid., page A100-5.

(8) Ibid., page A100-3.

(9) Personal Communication from R. Neidner.

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For the 1.6% enriched uranium

$$q = 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.444 \frac{\text{fission}}{\text{cm-neutron}} \\ \times 0.657 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2\text{-sec}} \times 0.8 \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times 7 \frac{(.533)^2}{4} \times 10 \text{ in}^3$$

= 176 kw at full power.

At 66 percent of full power

$$q = 176 \text{ kw} \times 0.66 = 116 \text{ kw} = 396,000 \text{ BTU/hr.}$$

$$\text{Total heat output for both elements} = 226 \text{ kw} = 772,000 \frac{\text{BTU}}{\text{hr}}$$

VIII. Coolant ΔT Across the Fuel Charge

$$\Delta T = \frac{q}{wcp}$$

cp for the average bulk temperature across the fuel element (190 F) is

$$0.997 \frac{\text{BTU}}{\text{lb-}^\circ\text{F.}}$$

The ΔT across the enriched element is then:

$$\Delta T = \frac{396,000 \frac{\text{BTU}}{\text{hr}}}{24,600 \frac{\text{lb}}{\text{hr}} \times 0.997 \frac{\text{BTU}}{\text{lb-}^\circ\text{F}}} = 16 \text{ F (9 C).}$$

With the F_b correction the $\Delta T = 16 \text{ F} \times 1.38 = 22 \text{ F.}$

The ΔT across the natural elements is:

$$\Delta T = \frac{376,000 \frac{\text{BTU}}{\text{hr}}}{24,600 \frac{\text{lb}}{\text{hr}} \times 0.997 \frac{\text{BTU}}{\text{lb-}^\circ\text{F}}} = 15 \text{ F (8 C).}$$

The total ΔT across the fuel charge is

$$\Delta T = 16 + 15 = 31 \text{ F (17 C).}$$

With the F_b correction the $\Delta T = 31 \times 1.38 = 43 \text{ F (24 C).}$

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IX. Temperature Distribution Through the Fuel

<u>Item</u>	<u>Uncorrected Temperature</u>	<u>Temperature Using the F_h and F_b Corrections</u>
Inlet temperature	175 F (79 C)	175 F (79 C)
Bulk ΔT (across the enriched element)	<u>16</u>	<u>22</u>
Local bulk water temperature	191 F (88 C)	197 F (92 C)
ΔT across coolant film	160	302
ΔT add	—	<u>6</u>
Cladding surface temperature	351 F (177 C)	505 F (263 C)
ΔT across the jacket	<u>275</u>	<u>275</u>
Bond layer surface temperature	626 F (330 C)	780 F (416 C)
ΔT across the bond layer	<u>32</u>	<u>32</u>
Uranium surface temperature	658 F (348 C)	812 F (434 C)
ΔT across the fuel core	<u>511</u>	<u>511</u>
Uranium core temperature	1169 F (632 C)	1323 F (718 C)

X. Specific Power of the Fuel Elements

To calculate the peak specific power of the cluster fuel element as a whole, we will use the peak unperturbed flux and apply the flux depression factor.

$$\phi = 1.63 \times 10^{14} \text{ nv} \times 0.516 = 0.841 \times 10^{14} \text{ nv}.$$

The peak specific power in the natural uranium elements will be:

$$\begin{aligned}
 Q &= 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.200 \frac{\text{fission}}{\text{cm-neutron}} \\
 &\times 0.841 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2\text{-sec}} \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times 7 \text{ rods} \\
 &\times \frac{(.533)^2 \pi \times 12 \text{ in}^3}{4 \text{ ft-rod}} = 152 \frac{\text{kw}}{\text{ft}} \text{ at full reactor}
 \end{aligned}$$

power. At 66 percent of full reactor power $Q = 152 \frac{\text{kw}}{\text{ft}} \times 0.66 = 100 \frac{\text{kw}}{\text{ft}}$.

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For the 1.6 percent enriched element

$$Q = 184 \frac{\text{Mev}}{\text{fission}} \times 1.60 \times 10^{-16} \frac{\text{kw-sec}}{\text{Mev}} \times 0.444 \frac{\text{fission}}{\text{cm-neutron}} \\ \times 0.841 \times 10^{14} \frac{\text{neutron}}{\text{cm}^2\text{-sec}} \times 0.8 \times \frac{\text{cm}^3}{0.061 \text{ in}^3} \times 7 \text{ rods} \\ \times \frac{(.533)^2 \pi \times 12 \text{ in}^3}{4 \text{ ft-rod}} = 270 \text{ kw/ft at full reactor power. At 66 percent} \\ \text{of full reactor power } Q = 270 \text{ kw/ft} \times 0.66 = 178 \text{ kw/ft.}$$

XI. Water Velocity Past Fuel

As noted above, we assumed a constant mass flow rate through the loop for all temperature conditions. Therefore, for the first part of the test the water velocity will be

$$\text{velocity} = \frac{\text{volume/sec}}{\text{flow area}} = \frac{24,600 \frac{\text{lb}}{\text{hr}} \times \frac{\text{ft}^3}{60.7 \text{ lb}} \times \frac{\text{hr}}{3600 \text{ sec}}}{0.00274 \pi \text{ ft}^2} \\ = 13.1 \text{ ft/sec}$$

XII. Maximum Surface Heat Flux

$$\text{Surface heat flux} = \frac{Q'}{A} = \\ = \frac{38.7 \text{ kw}}{\frac{.593 \pi}{12} \text{ ft-rod}} \times \frac{3413 \text{ BTU}}{\text{kw-hr}} = 851,000 \frac{\text{BTU}}{\text{hr-ft}^2}$$

XIII. Average Burnup Rate

$$\text{The average burnup rate} = \frac{Q}{\text{weight of fuel}}$$

For the natural uranium:

$$\text{The average burnup rate} = \frac{0.110 \text{ MW}}{8086.7 \text{ gm} \times \frac{2.2 \text{ lb}}{1000 \text{ gm}} \times \frac{\text{ton}}{2000 \text{ lb}}} \\ = 12.4 \frac{\text{MW}}{\text{ton}}$$

For the 1.6 percent enriched uranium the average burnup rate:

$$= \frac{0.116 \text{ MW}}{4813.5 \text{ gm} \times \frac{2.2 \text{ lb}}{1000 \text{ gm}} \times \frac{\text{ton}}{2000 \text{ lb}}} = 21.8 \frac{\text{MW}}{\text{ton}}$$

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XIV. Re-calculation for Rupture Conditions

At the time the fuel element is ruptured and until the reactor is shut down following the rupture, we want to maintain the maximum surface temperature of the enriched fuel element quite close to the saturation temperature of the loop water. We do not want boiling to occur in the loop at any place and yet we want the maximum surface temperature to be as hot as we can safely operate it (without boiling) so that the surface temperature at the defect will be as close to 570 F as we can get it. In order to meet these conditions, and yet keep the core temperature below beta phase temperature, it will be necessary to reduce total reactor power to 46 percent of full reactor power. Under these conditions, the peak specific power for the enriched fuel element will be:

$$Q' = 58.7 \frac{\text{kw}}{\text{ft-rod}} \times 0.46 = 27.0 \frac{\text{kw}}{\text{ft}}$$

(a) Average Water Film Coefficient

The properties of water at 575 F and 2000 psia are:

$$\mu = 0.20 \text{ lb/hr-ft}$$

$$\rho = 44.3 \text{ lb/ft}^3$$

$$k = 0.310 \text{ BTU/hr-ft-}^\circ\text{F}$$

$$c_p = 1.35 \text{ BTU/lb-}^\circ\text{F}$$

then

$$h = \frac{0.023 \times 0.310}{0.0214} \left(\frac{0.0214 \times 2.86 \times 10^6}{0.20} \right)^{0.8} \left(\frac{1.35 \times 0.20}{0.310} \right)^{0.3}$$

$$= 8160 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$$

(b) Maximum ΔT Across Coolant Film

$$\Delta T = \frac{27.0 \frac{\text{kw}}{\text{ft-rod}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}}}{8160 \frac{\text{BTU}}{\text{hr-ft}^2\text{-}^\circ\text{F}} \times \frac{\pi (.593)^2 \text{ft}^2}{12 \text{ ft-rod}}} = 73 \text{ F (41 C)}$$

With F_h correction:

$$\Delta T = 73 \text{ F} \times 1.89 = 138 \text{ F (77 C)}.$$

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(c) Maximum ΔT Across the Jacket

k for Zr-2 at 700 F = 8.19 BTU/hr-ft-°F.

$$\Delta T = \frac{27.0 \frac{\text{ft-rod}}{2\pi} \times \frac{3413 \text{ BTU}}{\text{kw-hr}} \times \ln \frac{.593}{.533}}{8.19 \frac{\text{BTU}}{\text{hr-ft-°F}}}$$

$$= 191 \text{ F (106 C)}.$$

(d) Maximum ΔT Across the Bond Layer

$$\Delta T = \frac{27.0 \frac{\text{kw}}{\text{ft-rod}} \times \frac{3413 \text{ BTU}}{\text{kw-hr}}}{30,000 \frac{\text{BTU}}{\text{hr-ft}^2\text{-°F}} \times \frac{\pi (.533) \text{ ft}^2}{12 \text{ ft-rod}}} = 22 \text{ F (12 C)}.$$

(e) Maximum ΔT Across the Fuel Core

k for uranium at 1000 F = 21.4 $\frac{\text{BTU}}{\text{hr-ft-°F}}$.

$$\Delta T = \frac{272 \times 27.0}{21.4} = 343 \text{ F (194 C)}.$$

(f) Total Heat Output

For the natural uranium fuel elements:

$$q = 166 \text{ kw} \times 0.46 = 76 \text{ kw} = 261,000 \frac{\text{BTU}}{\text{hr}}$$

For the enriched elements:

$$q = 176 \text{ kw} \times 0.46 = 81 \text{ kw} = 276,500 \frac{\text{BTU}}{\text{hr}}$$

$$\text{Total for both elements} = 157 \text{ kw} = 537,500 \frac{\text{BTU}}{\text{hr}}$$

(g) Coolant ΔT Across the Fuel Charge

cp for the average bulk temperature (540 F) across the fuel elements =

$$1.27 \frac{\text{BTU}}{\text{lb-°F}}$$

The ΔT across the natural uranium elements is:

$$\Delta T = \frac{261,000 \frac{\text{BTU}}{\text{hr}}}{24,6000 \frac{\text{lb}}{\text{hr}} \times 1.27 \frac{\text{BTU}}{\text{lb-°F}}} = 8 \text{ F (5 C)}.$$

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The ΔT across the enriched element is:

$$\Delta T = \frac{276,500 \frac{\text{BTU}}{\text{hr}}}{24,600 \frac{\text{lb}}{\text{hr}} \times 1.27 \frac{\text{BTU}}{\text{lb} \cdot ^\circ\text{F}}} = 9 \text{ F } (5 \text{ C})$$

With the F_b correction:

$$\Delta T = 9 \text{ F} \times 1.38 = 12 \text{ F } (7 \text{ C})$$

The total ΔT across the fuel charge is:

$$\Delta T = 8 \text{ F} + 9 \text{ F} = 17 \text{ F } (9 \text{ C}).$$

With the F_b correction $\Delta T = 17 \text{ F} \times 1.38 = 23 \text{ F } (13 \text{ C}).$

(h) Temperature Distribution Through the Fuel

<u>Item</u>	<u>Uncorrected Temperature</u>	<u>Temperature Using the F_b & F_h Correction</u>
Inlet Temperature	533 F (278 C)	533 F (278 C)
Bulk ΔT (across the enriched element)	<u>9</u>	<u>12</u>
Local bulk water temperature	542 F (283 C)	545 F (285 C)
ΔT across coolant film	73	138
ΔT add	<u> </u>	<u>6</u>
Cladding surface temperature	615 F (323 C)	689 F (365 C)
ΔT across the jacket	<u>191</u>	<u>191</u>
Bond layer temperature	806 F (430 C)	880 F (472 C)
ΔT across the bond layer	<u>22</u>	<u>22</u>
Uranium surface temperature	828 F (441 C)	902 F (483 C)
ΔT across the fuel core	<u>343</u>	<u>343</u>
	1171 F (633 C)	1245 F (675 C)

(i) Specific Power of the Fuel Elements

For the natural elements the peak specific power will be:

$$Q = 152 \frac{\text{kw}}{\text{ft}} \times 0.46 = 70 \frac{\text{kw}}{\text{ft}}.$$

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For the enriched element, it will be:

$$Q = 270 \frac{\text{kw}}{\text{ft}} \times 0.46 = 124 \frac{\text{kw}}{\text{ft}}$$

(j) Maximum Surface Heat Flux

$$\begin{aligned} \text{Surface heat flux} &= 27.0 \frac{\text{kw}}{\text{ft-rod}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}} \\ &= \frac{.59377}{12} \frac{\text{ft}^2}{\text{ft-rod}} \\ &= 593,8000 \frac{\text{BTU}}{\text{hr-ft}^2} \end{aligned}$$

(k) Average Burnup Rate

For the natural uranium elements:

$$\begin{aligned} \text{Average burnup rate} &= \frac{0.076 \text{ MW}}{8.0867 \text{ kg} \times 2.2 \frac{\text{lb}}{\text{kg}} \times \frac{\text{ton}}{2000 \text{ lb}}} \\ &= 8.55 \frac{\text{MW}}{\text{ton}} \end{aligned}$$

For the enriched element:

$$\begin{aligned} \text{Average burnup rate} &= \frac{0.081 \text{ MW}}{4.8135 \text{ kg} \times 2.2 \frac{\text{lb}}{\text{kg}} \times \frac{\text{ton}}{2000 \text{ lb}}} \\ &= 15.3 \frac{\text{MW}}{\text{ton}} \end{aligned}$$

(l) Water Velocity Past the Fuel

$$\text{Velocity} = \frac{65 \text{ gal/min} \times 0.1337 \frac{\text{ft}^3}{\text{gal}} \times \frac{\text{min}}{60 \text{ sec}}}{0.00274 \pi \text{ft}^2} = 16.8 \text{ ft/sec}$$

XV. Surface Temperature at Defect

The surface temperature at the defect during the period of rupture operation should approach NPR conditions as closely as possible. Outlet water temperature will approach 300 C, (10) and surface temperatures will be somewhat in excess of 300 C because of the film drop which will exist. The unperturbed, axial flux which will exist at the defect will be $0.88 \times 10^{14} \text{ nv}$. (11) The ratio of this flux to the peak, unperturbed, axial flux is:

$$\frac{0.88}{1.63} = 0.58$$

(10) Carlson, P.A., NPR Fuel Element Designs, HW-57669, Rev., Feb. 4, 1959 (Secret).
(11) Personal Communication from R. Neidner.

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The defect will be assumed to be oriented so that the rod containing the defect is closest to the center of the reactor core and, therefore, will be the highest heat generating rod. If the same flux correction factors which were used before are assumed to apply at the defect location, the specific power at the defect location will be equal to the peak specific power times the above ratio. Thus, at the defect

$$Q' = 27.0 \frac{\text{kw}}{\text{ft-rod}} \times 0.58 = 15.65 \frac{\text{kw}}{\text{ft-rod}}$$

The average film temperature will be approximately 550 F. The properties of water at 550 F are:

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

$$\rho = 45.9 \text{ lb/ft}^3$$

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

$$c_p = 1.26 \text{ BTU/lb-}^\circ\text{F}$$

(a) Average Water Film Coefficient

$$n = \frac{0.023 \times 0.325}{0.0214} \left(\frac{0.0214 \times 2.86 \times 10^6}{0.22} \right)^{0.8} \left(\frac{1.26 \times 0.22}{0.325} \right)^{0.3}$$

$$= 7480 \text{ BTU/hr-ft}^2\text{-}^\circ\text{F}$$

(b) ΔT Across Coolant Film

$$\Delta T = \frac{15.65 \frac{\text{kw}}{\text{ft-rod}} \times 3413 \frac{\text{BTU}}{\text{kw-hr}}}{7480 \frac{\text{BTU}}{\text{hr-ft}^2\text{-}^\circ\text{F}} \times \frac{\pi (.593)^2 \text{ft}^2}{12 \text{ ft-rod}}} = 46 \text{ F (26 C)}$$

The surface temperature will, therefore, be approximately 533 F + 46 F = 579 F (304 C).

This is probably somewhat lower than the surface temperatures of NPR fuel elements will be but will be acceptable.

HOT SPOT AND HOT CHANNEL FACTORS

	<u>F_b</u>	<u>F_h</u>	<u>AT 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	1.04	
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	1.02	1.02	
4. Effect of D variation	1	1	
C. Correlation Equation on h	1	1.2	
D. Determination of Critical Parameters			
1. Flow measurement	1.03	1.03	1.4
2. Temperature measurements (metal)			
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	

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	<u>F_b</u>	<u>F_b</u>	<u>ΔT Add</u>
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	1.2	1.2	
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	1.05	
2. Mechanical effects	1	1	
I. Effects Peculiar to Experiment	1	1	
Products (total ΔT add)	1.38	1.89	5.7

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APPENDIX II

ASSEMBLY PROCEDURE

The fuel basket, fuel elements, thermocouples, and hydraulic mechanism will be received at the ETR unassembled. Two irradiated defect elements will be shipped in a cask. One irradiated element will be used in the test, the other is a backup in case the first is damaged in handling or shipping. Each irradiated fuel element is inside an aluminum cylinder. These cylinders were placed inside an aluminum shipping container and the shipping container placed inside a shielded cask. The shipping container is to be removed from the cask. The cylinders are to be removed from the shipping container, and one end cut off one cylinder. The fuel element then will slide out of its cylinder. Care must be used in cutting, so that the fuel element is not damaged. The fuel element will then be inspected for damage, and the assembly procedure outlined below will be carried out unless there is evidence of gross damage to the element.

The downstream thermocouple will be attached to the basket and then threaded through the quick disconnect and the 3/8 inch tube. The two heater elements will then be inserted in 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 defect element will be installed in the assembly jig, see Figure 8, and the rupture mechanism and hydraulic line will be attached to it. The defect element and rupture mechanism will be lifted together and inserted in the basket. The quick disconnect and 3/8 inch tube will be threaded over the hydraulic line, pushed down into position in the basket, and locked into place. A plug which contains slots for the thermocouples and hydraulic line is then pushed into the end of the 3/8 inch tube and silver-soldered in place.

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APPENDIX IIIINSERTION AND REMOVAL OF THE TEST ASSEMBLY

After the reactor water is lowered and the GEH R3X3P7 process tube cap is removed, the process tube is ready for charging. The test assembly is to be transferred in the canal to the discharge chute and the thermocouples and hydraulic line threaded through the discharge chute and into the reactor. The test assembly will then be lifted into the reactor and lowered into the process tube. The test assembly is seated properly when 6-1/2 inches of the 3/8 inch tubing containing the thermocouples and hydraulic lines remain above the process tube opening. The thermocouples and hydraulic line will then be threaded through the process tube cap and the cap seated on the tube. The Conax seal fittings will then be threaded on the thermocouples and hydraulic line and the thermocouples and hydraulic line are to be threaded through the external thermocouple hole. The Conax seals will then be tightened up 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 7. The hydraulic line will then be bled and filled with water.

To discharge the assembly, the reactor water is first lowered. The thermocouples and hydraulic line are then cut just above the 3/8 inch tubing. The Conax seal on the process tube cap is then loosened and the process tube cap is removed. The basket removal tool shown in Figure 9 is then lowered into the reactor and attached to the tube by means of the studs located on the flange that has been previously attached to the process tube. The reach rod will then be lowered and the basket engaged. The basket will then be pulled into the rupture discharge assembly. The ball valve on the rupture discharge assembly will then be closed, and the reach rod unscrewed at the joint directly above the discharge tube. The discharge tube assembly will now be unfastened from the process tube and discharged through the reactor discharge chute. This method of discharge will prevent contamination of the ETR reactor water.

Print No. H-3-9266, giving the details of the rupture removal tool, is in J. H. Johnson's file.

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APPENDIX IV - L. D. PerrigoDECONTAMINATION PROCEDURE

The following procedure is to be used in decontaminating the GEH-R3X3P7 loop following the rupture test.

A. Decontamination Sequence

1. Drain loop as soon as possible after reactor shutdown. At this time, valve-out cleanup ion exchange columns. (Take one gallon H₂O sample prior to draining loop.)
2. Flush loop for 15 minutes with H₂O.
3. Fill loop with demineralized H₂O. Pressurize and circulate. Valve in ion exchange column. Continue circulation at low temperature (< 180 F) until ready for decontamination operation. Draw one gallon water sample, prior to beginning Step 4.
4. Begin decontamination operation - Drain loop. Valve out ion exchange columns.
 - 5a. Fill loop with Solution I (Peroxide carbonate mixture). Pressurize to 1000 psi, circulate for 15 minutes with loop temperature 125 - 140 F. Draw a one gallon water sample just prior to beginning Step 5b.
 - 5b. Drain loop.
 - 5c. Flush loop.
 - 5d. Again fill loop with Solution I (Peroxide carbonate mixture). Pressurize to 1000 psi, circulate for 30 minutes with loop temperature 125° - 140°F. Draw a one gallon water sample just prior to beginning Step 6.
6. Drain loop.
7. Flush loop for 15 minutes with H₂O.
8. Fill loop with demineralized H₂O. Pressurize to 300 psi, circulate with main pumps for 10 minutes.
9. Drain loop.
10. Flush loop for 15 minutes with H₂O.

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11. Fill loop with Solution II (Caustic permanganate mixture). Pressurize to 300 psi, circulate for 1 hour with loop temperature at 220 F. One gallon water samples will be drawn at 30 minutes after circulation begins and just prior to beginning Step 12.
12. Drain loop.
13. Flush loop for 15 minutes with H₂O.
14. Fill loop with demineralized H₂O. Pressurize to 300 psi, circulate with main pumps for 10 minutes.
15. Drain loop.
16. Flush loop with H₂O until coolant pH is less than 10.0.
17. Fill loop with Solution III (Ammonium Citrate). Pressurize to 500 psi, circulate for 1 hour with loop temperature at 180 F. One gallon water samples will be drawn at 30 minutes after circulation begins and just prior to beginning Step 18.
18. Drain loop.
19. Flush loop for 15 minutes with H₂O.
20. Fill loop with demineralized H₂O. Pressurize to 300 psi, circulate with main pumps for 10 minutes.
21. Drain loop.
22. Flush loop for 15 minutes with H₂O.
23. Fill loop with demineralized H₂O. Pressurize to 300 psi, circulate with main pumps with coolant maintained below 200 F. Valve in H-OH cycle mixed bed ion exchange resin (XE-150.) Continue circulation with cleanup flow until H₂O quality exceeds 100,000 ohm-cm specific resistivity. Use portable monitoring techniques on cell to give complete mapping of cell activity levels.

B. Bearing Protection

To give added protection to the bearings, water flow through the bearings on the canned rotor primary pumps should be started at the beginning of Steps 5, 11, and 17. It is important that there be flow during each of these steps, but it is also essential that this flow be kept at a low value (approximately 0.1 gallons per minute) to prevent excessive dilution of the decontaminating solutions. This flow should be maintained until the pumps are stopped following each cleaning process (Steps 5, 11, and 17.)

C. Activity Monitoring

The cell and loop activity levels should be monitored continuously until

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the process is completed (end of Step 23).

D. Sampling

Each 1 gallon water sample should be identified when it is drawn with loop number, date, and time (to the minute).

One gallon water samples will be drawn as follows:

- a. Step 1 - Before loop is drained.
- b. Step 3 - Prior to beginning Step 4.
- c. Step 5 - Prior to beginning Step 5b and prior to beginning Step 6.
- d. Step 11 - 30 minutes after circulation begins; another just before starting Step 12.
- e. Step 17 - 30 minutes after circulation begins; another just before starting Step 18.

These samples should be subject to a radioanalysis. The following information is required:

- a. Uranium concentration (wet chemical analysis)
- b. Ru¹⁰³ activity
- c. Zr-Nb⁹⁵ activity
- d. Cs¹³⁷ activity
- e. Total iodine activity
- f. Te¹³² activity
- g. Mo⁹⁹ activity
- h. Ba-La¹⁴⁰ activity
- i. Np²³⁹ activity
- j. Ce¹⁴³ activity

The results of the above analyses are to be transmitted to J. H. Johnson.

E. Decontamination Solutions

The amounts of the chemicals to be added to the mix tank have been calculated on the basis that eighty-five gallons of solution will be needed for each cleaning operation.

1. Solution I - This solution is used to oxidize the uranium in UO₂ from the + 4 valence to +6 valence, so as to facilitate its removal from the loop system.

The composition is as follows:

- a. 0.5 Molar H₂O₂
- b. 0.25 Molar NaHCO₃
- c. 0.25 Molar Na₂CO₃
- d. 3.5 gm/liter ethylenediamine tetra acetic acid (EDTA)
- e. 1.0 gm/liter 8-hydroxy quinoline
- f. .01 gm/liter Duponal.

The following quantities of chemicals will be added to the mix tank to

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compound Solution I:

- a. 15 pounds of NaHCO_3 (solid)
- b. 19 pounds of Na_2CO_3 (solid)
- c. .7 pounds of 8-hydroxy quinoline (solid)
- d. 2.5 pounds of EDTA (solid)
- e. 3.0 gm of Duponal (solid)
- f. 40 one-ound bottles of 30% H_2O_2 (liquid).

The NaHCO_3 , Na_2CO_3 , EDTA and 8-hydroxy quinoline will be added to the tank first. Add demineralized H_2O to the 80-gallon mark on the Mix tank. Mix for ten (10) minutes, heating is not necessary. Add Duponal and hydrogen peroxide, mix for one or two minutes. Introduce into the loop. Caution: This mixture should not be allowed to stand for any appreciable period of time before use in the loop. Standing will permit decomposition of the H_2O_2 to the point where it will not be useful.

2. Solution II - This caustic permanganate solution is used to condition the oxide films in the loop. Its composition is as follows:

- a. 5.0 Molar NaOH
- b. 0.23 Molar KMnO_4

The following quantities of chemicals will be added to the mix tank to compound Solution II:

- a. 140 pounds of NaOH (solid).
- b. 23 pounds of KMnO_4 (solid).

Add these two chemicals to the mix tank. Add demineralized H_2O to the 85-gallon mark on the mix tank. Stir and raise temperature to 230 F. Introduce into the loop.

3. Solution III - The ammonium citrate solution is used to remove the oxide films from the loop. The composition of the solution is as follows:

- a. .5 Molar ammonium citrate (solid).
- b. .01 Molar EDTA (solid).

The following quantities of chemicals will be added to the mix tank to compound Solution III:

- a. 80 pounds of ammonium citrate (solid).
- b. 2.5 pounds of EDTA (solid).

Add these chemicals in their dry state to the mix tank. Add demineralized H_2O to the 85-gallon mark on the mix tank. Stir and heat to 180 F. Introduce into loop.

R. D. Weed and L. D. Perrigo of the Coolant Systems Development Operation Irradiation Processing Department, will be at the ETR for this decontamination procedure to act as advisers to J. H. Johnson.

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FIGURE - 1

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FIGURE - 2

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IN-REACTOR RUPTURE MECHANISM

FIGURE 3

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FIGURE 4

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FIGURE 5

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RUPTURE EXPERIMENT
PORTABLE HYDRAULIC SYSTEM

FIG. 7

ASSEMBLY TOOL

RUPTURE EXPERIMENT

FIGURE 8

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RUPTURE REMOVAL TOOL
3 X 3 BALL VALVE

FIG. 9

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FIGURE 10.

DATE

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END