PROPOSAL TO IRRADIATE POWER BURST FACILITY FUEL ELEMENTS
UNDER STEADY STATE CONDITIONS IN THE ETR M-3 LOOP

by

R. K. Marshall
Fuel Design and Analysis

April 1971

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1.0 INTRODUCTION AND SUMMARY

Battelle-Northwest was requested to provide assistance to the Idaho Nuclear Corporation to conduct a steady-state irradiation of Power Burst Facility (PBF) fuel elements in the ETR M-3 loop.

The Idaho Nuclear Corporation will provide the PBF fuel rods, dummy rods, spacers, and hardware design. BNW will provide the hardware and has herein provided the design analyses.

The final design of the test assembly consists of a 3x3 rod array with the rods located on a 0.792 inch pitch. Seven of the positions in the array are occupied by PBF fuel rods and two are occupied by dummy rods. The dummy rods occupy the corner positions nearest to the reactor. The radial flux gradient across the facility causes the power generated in fuel rods located in these positions to exceed the maximum design heat rating.

The maximum design heat rate is 14.5 kW/ft. The test flow velocity is 11 ft/sec, the loop operating pressure is 600 psi, and the coolant inlet temperature is 200°F.

The PBF production rod fuel contains 30.6 wt% UO_2 (18.3% enriched), 62.3 wt% ZrO_2 , and 7.1 wt% CaO .

2.0 DESCRIPTION OF TEST COMPONENTS

2.1 FUEL PELLETS AND RODS

The fuel consists of a ternary mixture of UO_2 , ZrO_2 , and CaO compacted into cylindrical pellets. The active length of the fuel column in each rod is 36 inches. The pellets are contained in 304L stainless steel tubes which are sealed by welding caps into each end. A zirconia insulator sleeve is located between the pellet column and the inside surface of the cladding. The cold nominal radial gap between the pellets and sleeve is 0.019 inch and the radial gap between the insulator sleeve and cladding is limited to 0.0005 inch.

A plenum at the top of the rod provides space for fission gases released from the pellets and for axial expansion of the fuel column. A coil spring loading a bearing spool is located in the plenum to maintain a compact fuel column. The fill gas is helium at 0.5-1.0 psig. The design of the fuel rods is shown in Phillip's Petroleum Company (PPCO) Drawing SPT-E-8743, Revision 1. See Drawing Section.

The design parameters for the fuel pellets and rods are shown in Table 2.1-1.

2.2 TEST ASSEMBLY

The seven fueled and two dummy rods are contained in a tube, called a basket, and are supported by a mounting plate attached to the bottom of the basket. Inserts are positioned between the fuel rod array and the inside surface of the tube to provide a 2.376 in. x 2.376 in. square flow channel for the full length of the fuel rods. Two egg

TABLE 2.1-1 MECHANICAL DESIGN PARAMETERS FOR PBF FUEL RODS

Fuel Rod

Fuel Material	UO ₂ -ZrO ₂ -CaO
Urania Content, wt%	30.6
Zirconia Content, wt%	62.3
Calcium Content, wt%	7.1
Enrichment, wt% ²³⁵ U in Total U	18.3
Pellet Diameter (in.)	0.559 ± .001
Pellet Length (in.)	1.100 (max.) 0.560 (min.)
Clad Material	304L Stainless Steel
Clad OD (in.)	0.75
Clad Thickness (in.)	0.027 (min.)
Zirconia Sleeve Thickness (in.)	0.048
Nominal Cold Helium Gap (in.)	0.019 (Radial)
Fuel Column Length (in.)	36.000 ± .036

crate type spacers are located along the length of the fuel rods to maintain the fuel rod pitch. The design of the test assembly is shown in INC Drawing Nos. 402396, 402397 Sheets 1-3, 402398, and 402399. Drawings 402396, 402397 Sheet 1, and 402398 are included in the Drawing Section.

Each fuel rod is attached to the mounting plate by a bolt which extends through the mounting plate and through a captive plate and is screwed into the end fitting of the fuel rod. Projections on the end fittings of the fuel rods mate with slots in the mounting plate to prevent rotation of the fuel rods.

All hardware components are fabricated from 304 or 304L stainless steel except the spacers which are fabricated from aluminum, and the basket inserts which are fabricated from Inconel Alloy 600.

The dummy rods consist of iron pellets clad with 304L stainless steel and have the same outside dimensions as the fuel rods.

3.0 OPERATING CONDITIONS

The operating conditions for the test are shown in Table 3.0-1. The most important test condition which can be controlled is the inlet water temperature. The operating inlet temperature for the PBF Reactor coolant is approximately 130°F.

M-3 loop heat exchanger calculations were made which conservatively predicted that loop inlet temperatures of 200°F are achievable at the calculated total heat generation of 231 kW with tube side and shell side flow rates of 100 gpm. The characteristics of the M-3 loop heat exchanger are shown in Figure 3.0-1.

Operation of the inlet temperature of the loop coolant to as close to 130°F as practicable is requested.

The test is scheduled to operate for one cycle only.

4.0 SPECIAL REQUIREMENTS

Orientation of the basket relative to the reactor is critical. Refer to INC Drawing No. 402396, Fuel Rod Test Hardware - PBF. Note the indexing toggle located at the end of the extension rod attached to the quick disconnect. With the assembly in the reactor, the toggle, when in the horizontal position, must point due south.

TABLE 3.0-1 PBF TEST OPERATING CONDITIONS

Reactor Power Level (MW)	175
Total Heat Generated, Fission and Gama (kW)	231
Operating Pressure (psig)	600
Inlet Temperature (°F)	200
Loop Outlet Temperature (With Hot Spot Factor) (°F)	230
Flow Rates* (gpm)	
Total	82
Basket Bore	57
Basket - Pressure Tube Annulus	25
Average Flow Velocity (ft/sec)	11.0
Pressure Drop Across Test Assembly (psi)	10
Maximum Heat Rating of Fuel Rod (kW/ft)	14.5
Maximum Fuel Temperature	Molten
Maximum Cladding Surface Temperature (With Hot Spot Factor) (°F)	300
Saturation Temperature (°F)	488
pH of Coolant	5.5 - 6.0

* Calculated values. Flow tests will be run on test hardware. The results of the flow tests will be used for the in-reactor test.

FIGURE 3.0-1 M-3 LOOP HEAT EXCHANGER CHARACTERISTICS FOR PBF TEST

5.0 MECHANICAL ANALYSIS

5.1 EXTERNAL COLLAPSE PRESSURE, OPERATING PRESSURE, AND STRESS

5.1.1 Collapse and Operating Pressure

A tube is considered to be "free-standing" if yielding of the material does not occur at the maximum net external pressure and temperature conditions. An ideal tube will collapse when the net external pressure reaches:⁽¹⁾

$$P_{cr} = \frac{E}{4(1-\nu^2)} \left(\frac{h}{R} \right)^3 \quad (1)$$

where:

P_{cr} = Critical Collapse Pressure, psi

E = Modulus of Elasticity = 28×10^6 psi

ν = Poisson's Ratio = 0.30

h = Minimum Tube Wall Thickness, 0.027 inch

R = Tube Mean Radius, 0.36325 inch.

$$P_{cr} = \frac{28 \times 10^6}{4(1-0.09)} \left[\frac{0.027}{.36325} \right]^3$$

$$P_{cr} = 3158 \text{ psi}$$

For a tube with less than ideal dimensions, collapse will occur at less than P_{cr} due to effects of ovality. The equation from Reference 1 for this condition accounts for both tube ovality and yield strength. In this case, the critical collapse pressure for yielding becomes:

$$P_{cry} = 1/2 [A - \sqrt{A^2 - 4B}] \quad (2)$$

where:

$$A = \sigma_y \left(\frac{h}{R} \right) + \left[1 + 6 \frac{U_o}{R} \frac{R}{h} \right] P_{cr} \quad 2)$$

$$\sigma_y = \text{Yield Strength} = 40,000 \text{ psi}$$

h/R - as previously stated in Equation (1)

$$U_o = 1/4 (ID_{\max.} - ID_{\min.})$$

P_{cr} = That value determined by Equation (1), and

$$B = \sigma_y \frac{h}{R} P_{cr}$$

$$A = (40,000) \left(\frac{.027}{.36325} \right) + \left(1 + \frac{1.5 (.002)}{0.027} \right) 3158$$

$$A = 6482$$

$$B = (40,000) \left(\frac{.027}{.36325} \right) 3158$$

$$B = 9.39 \times 10^6$$

$$P_{cr_y} = 1/2 \left[6482 - \sqrt{42.02 \times 10^6 - 37.56 \times 10^6} \right]$$

$$P_{cr_y} = 2175 \text{ psi}$$

A maximum loop operating pressure of 600 psi was selected which provides a safety factor of 3.6.

5.1.2 Pressure Stress

The hoop stress is determined from the equation:

$$\sigma_\theta = \frac{P_o R}{h}$$

The bending stresses due to ovality are calculated from:

$$\sigma_b = \pm \sigma_\theta \left[6 \frac{U_o}{h} \frac{P_{cr}}{(P_{cr} - P_o)} \right]$$

where: σ_{θ} = Hoop stress, psi; σ_b = Bending stress, psi

P_o = External pressure, psi

$$\sigma_{\theta} = (600)(.36325)/(.027) = 8100 \text{ psi}$$

$$\sigma_b = \pm (8100)(.111)(1.234) = \pm 1100 \text{ psi}$$

$$\sigma_{\theta} + \sigma_b = 7100 \pm 1100 \text{ psi}$$

5.2 CLADDING THERMAL STRESSES

The thermal stresses are directly proportional to the temperature gradient across the cladding wall. The thermal stresses at the surface may be written as:

$$\sigma_{\theta} = \pm \frac{E \alpha \Delta t}{2(1-\nu)}$$

where:

E = Modulus of Elasticity = 28×10^6 psi

α = Coefficient of Thermal Expansion = 8.0×10^{-6} in/in/°F

Δt = Temperature Gradient Across Cladding Wall, 73°F @ 14.5 kW/ft

ν = Poisson's Ratio = 0.30

σ_{θ} = Hoop Stress, psi

$$\sigma_{\theta} = \pm \frac{(28 \times 10^6)(8.0 \times 10^{-6})(73)}{(2)(0.70)}$$

$$\sigma_{\theta} = \pm 11,700 \text{ psi}$$

5.3 COMBINED STRESSES

The maximum stress occurs at the inner surface of the cladding and results from a combination of the hoop and thermal stresses. The results

below are at a heat rating of 14.5 kW/ft, an external pressure of 600 psi, and an internal pressure of 0 psi.

$$\text{Maximum operating stress, } \sigma_m = 8100 + 1100 + 11,700$$

$$\sigma_m = 20,900 \text{ psi}$$

This operating stress is 52% of the yield stress of 40,000 psi.

5.4 INTERNAL FISSION GAS PRESSURE

For each megawatt day (MWd) of burnup, 1.3×10^{-3} gm-mols of Kr + Xe are formed. Using this data and assuming all of the gas generated is released from the fuel, the total mols of Kr + Xe can be calculated:

Tons of U per fuel rod:

$$\frac{238 \text{ gmU}}{6.04 \times 10^{23} \text{ atom U}} \times \frac{41.6 \times 10^{20} \text{ atoms U}}{\text{cm}^3 \text{ fuel}} \times \frac{\pi}{4} (.559)^2 (36) (16.34) \frac{\text{cm}^3 \text{ fuel}}{\text{rod}} \times \frac{\text{MT}}{10^6 \text{ gm U}}$$

$$= 2.37 \times 10^{-4} \text{ Metric tons U/rod}$$

Heat generated per fuel rod:

$$= \frac{14.5 \text{ kW} \times 3 \text{ ft MW}}{1.32 \text{ ft rod } 1000 \text{ kW}} = 0.0330 \frac{\text{MW}}{\text{Rod}}$$

Burnup after one cycle, assume 30 days

$$\text{Burnup} = 0.0330 \frac{\text{MW}}{\text{Rod}} \frac{\text{Rod}}{2.37 \times 10^{-4} \text{ MT}} \times 30 \text{ Days}$$

$$\text{Burnup} = 4200 \frac{\text{MWd}}{\text{MTM}}$$

$$\text{Total Kr + Xe} = 1.31 \times 10^{-3} \frac{\text{gm-mols}}{\text{MWd}} \times 2.37 \times 10^{-4} \frac{\text{MT}}{\text{Rod}} \times 4200 \frac{\text{MWd}}{\text{MT}}$$

$$\text{Total Kr + Xe} = 1.3 \times 10^{-3} \text{ gm-mols of Kr and Xe/rod}$$

Internal Pressure:

Assume that all fission gas is collected in the plenum and is at the average temperature of the coolant.

$$PV = n RT$$

$$n = 1.3 \times 10^{-3} \text{ gm-mols}$$

$$R = 40.6 \frac{\text{in-lb}}{\text{gm-mol } ^\circ\text{R}}$$

Volume: (V)

Annulus between pellet column and insulator sleeve ID

$$\frac{\pi}{4} (.5975^2 - .570^2)(36 + 2.5) = 0.971 \text{ in}^3$$

Annulus between bearing spool and cladding ID

$$\frac{\pi}{4} (.691^2 - .562^2)(2) = 0.254 \text{ in}^3$$

Cavity in bearing spool

$$\frac{\pi}{4} (.469^2)(4.312) = 0.745 \text{ in}^3$$

Free volume in spring chamber

$$\frac{\pi}{4} (.691^2)(4.250) - \frac{.087 \text{ lbs in}^3}{\text{spring } .280 \text{ lbs}} = 0.972 \text{ in}^3$$

$$\text{Total volume} = 0.971 + 0.254 + 0.745 + 0.972 = 2.942 \text{ in}^3$$

Assume temperature of coolant is $\sim 250^\circ\text{F}$ or 710°R

$$P = \frac{(1.3 \times 10^{-3})(40.6)(710)}{2.942}$$

$$P = 13 \text{ psi from fission gas}$$

6.0 HEAT GENERATION

6.1 FISSION HEAT GENERATION

The fission heat generation calculations for the test assembly with seven fueled rods and two dummy rods were performed with the THERMOS and the 2DB computer codes.^(2,3) The calculations and the heat generation results obtained⁽⁴⁾ from the computer runs were based on experimental flux data obtained from the irradiation of GEH-10-101-148 in the M-3 loop during cycle 85.

The thermal flux in the M-3 loop during cycle 112 could be approximately 10% less than during cycle 85⁽⁵⁾. The heat generation results which were based on cycle 85 were reduced 8% to compensate for the expected lower flux in the M-3 loop during cycle 112. Based on heating rates normalized to cycle 112, the predicted maximum linear fission heat rating in kW/ft for each fuel rod is shown in Figure 6.1-1.

Reactor

Figure 6.1-1. PBF Fuel Rod Maximum Fission Heat Rate - kW/ft

The total fission power in the assembly is based on an axial peak to average factor of 1.32 and an active fuel length of three feet.

$$\text{Total Fission Power} = \frac{[9.5 + 9.8 + 11.7 + 10.3 + 11.7 + 13.2 + 13.9]}{1.32} \quad (3)$$

Total Fission Power = 182 kW

6.2 GAMMA HEAT GENERATION

The gamma heat generation is based on an average gamma heating rate of 1.0 watt/gm in the active region. The dummy rods, fuel, insulating sleeve, cladding, basket, and pressure tube were considered in the calculation. The results are summarized in Table 6.2-1 and the detailed calculations are presented in Appendix A.

<u>Component</u>	<u>Component γ Heat Rating kW</u>	<u>Number of Components per Test Assembly</u>	<u>Total γ Heat Rating kW</u>
Dummy Rods	1.1	2	2.2
Fuel	0.8	7	5.6
Sleeve	0.3	7	2.1
Cladding	0.3	7	2.1
Basket	13.9	1	13.9
Pressure Tube	22.6	1	22.6
Total			48.5

TABLE 6.2-1. Gamma Heat Generation Rates for PBF Test Assembly

6.3 TEST ASSEMBLY AND FUEL ROD HEAT GENERATION

6.3.1 Test Assembly

The total heat generated in the test assembly is the sum of the gamma heat generated in the fuel and hardware components and the fission heat generated in the fuel.

$$\text{Total Heat Generated} = \text{Gamma} + \text{Fission}$$

$$\text{Total Heat Generated} = 48.5 + 182 = \underline{230.5 \text{ kW}}$$

6.3.2 Maximum Linear Heat Rating in Highest Powered Fuel Rod

The maximum linear heat rating occurs in the fuel rod nearest the reactor and is the sum of the gamma and fission heating in the rod. The gamma heating rates are from Table 6.2-1; the fission heating rate is from Figure 6.1-1.

$$\text{Gamma} = \frac{(0.8 + 0.3 + 0.3)}{3} (1.32) = 0.6 \text{ kW/ft}$$

$$\text{Fission} = 13.9 \text{ kW/ft}$$

$$\text{Maximum Linear Heat Rating} = 13.9 + 0.6 = \underline{14.5 \text{ kW/ft}}$$

7.0 THERMAL-HYDRAULIC ANALYSIS

7.1 HYDRAULIC CHARACTERISTICS

The pressure drop versus flow rate characteristics for the basket bore and for the basket-pressure tube annulus are presented in Figure 7.1-1 and are summarized below for the loop operating conditions.

	<u>Flow Rate</u> <u>gal/min.</u>
Basket Bore	57
Basket-Pressure Tube Annulus	25
Total	82

The pressure drop across the assembly of 10 psi is calculated for these conditions. The calculations were made using the equation shown in Appendix B.

7.2 FUEL ROD SURFACE AND COOLANT TEMPERATURES

The fuel rod surface and coolant temperatures are summarized below and the detailed calculations are presented in Appendix B.

	<u>Calculated</u>	<u>Adjusted</u>
Loop Inlet, °F		200
Loop Outlet, °F	221	230
Basket Outlet, °F	225	234
Surface Temperature (Max.) °F	272	300
Saturation Temperature @ 600 psi		488
Surface Heat Flux, (Maximum), Btu/hr-ft ²		251,000

FIGURE 7.1-1 HYDRAULIC CHARACTERISTICS FOR PBF TEST IN THE M-3 LOOP

7.3 FUEL THERMAL ANALYSIS

The maximum design linear heat rating for the fuel is 14.5 kW/ft.

Thermal analyses of the fuel were performed to determine the heat rating at which incipient fuel melting would occur and the extent of fuel melting at 14.5 kW/ft.

Two cases were considered to determine the linear heat rating at which incipient fuel melting (4260°F) would occur:

- (1) The fuel pellets crack and fill the gap between the pellets and insulator sleeve.
- (2) The fuel pellets do not crack but thermally expand 2-1/2% leaving a radial gap filled with helium between the pellets and insulator sleeve.

The calculated linear heat rating at which incipient melting occurs is 12.0 kW/ft for both cases. A plot of fuel centerline temperatures versus linear heat rating is shown in Figure 7.3-1 for case 1.

In case 2, the transfer of heat by both conduction and radiation was considered. The combined conduction and radiation heat transfer coefficient is 275 Btu/hr-ft²-°F. The thermal conductivity for the uncracked fuel (case 2) is significantly higher than the thermal conductivity for the cracked fuel (case 1).

For the extent of fuel melting at 14.5 kW/ft, the calculations predict that 54% of the fuel pellet radius will be above the melting point. This condition will produce a combination of the solid and liquid phase and the liquid phase.

FIGURE 7.3-1 FUEL CENTERLINE TEMPERATURE - PBF FUEL IN ETR

The thermal conductivities of the stainless steel cladding, the ZrO_2 insulator sleeve, and the cracked and uncracked fuel pellets were taken from Reference 6. Contact coefficients of $1000 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$ for the cracked fuel-insulator interface and of $2000 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$ for the insulator-cladding interface were obtained from R. P. Watkins⁽⁷⁾ of INC. No credit was taken for fuel restructuring.

The THERMOS⁽²⁾ code was used to calculate the thermal flux depression across the fuel. The CHEAT1⁽⁸⁾ code was used to calculate the temperature distribution across the fuel rod.

The elongation in the fuel column length at temperatures immediately below the melting point is 3%.⁽⁹⁾ This elongation would cause a nominal 1.08 inch increase in fuel column length. The remaining plenum spring deflection to solid height is a nominal 1.57 inch. Thus, sufficient clearance is available for fuel column elongation.

8.0 HAZARDS ANALYSIS

8.1 FUEL ROD-TO-FUEL ROD CONTACT

The pitch of the rods in the 3x3 array produces a compact bundle of fuel rods with a nominal clearance between adjacent rods of 0.040 inch. When consideration is given to tolerances, initial non-straightness of rods, and bowing as a result of the radial flux gradient across the fuel rods, the possibility of adjacent rods touching cannot be discounted.

An analysis was made to determine the effect on the cladding temperature of contacting fuel rods. A quarter section of the cladding cross section was divided into 9 nodes of 10° each. The surface heat transfer coefficient for two adjacent nodes was set equal to essentially zero to simulate fuel rod contact. A 20° section which corresponds to a circumferential distance of 0.132 inch is considered a conservative distance over which contacting rods would restrict the flow of heat. The heat transfer coefficient for the remaining surface area was assumed to be 5000 Btu/hr-ft²-°F. The linear heat rating was 14.5 kW/ft.

The results of the analysis predicted local maximum cladding temperatures of 560°F and maximum surface temperatures of 525°F. A surface temperature of 525°F is 37°F above the saturation temperature of 488°F. Localized boiling can be expected.

The temperature gradients across the cladding and circumferentially along the cladding at the location of contact increase, which causes

an increase in the cladding thermal stress. An order of magnitude stress calculation was made which predicts that the stress in the cladding would be greater than yield and less than the ultimate stress. Yielding of the cladding would be restricted by the insulating sleeve.

Stainless steel clad fuel rods have failed under high stress conditions because of stress-assisted intergranular corrosion attack.⁽¹⁰⁾ Because of the short duration of this test (one cycle), stress-assisted intergranular corrosion is not expected to cause cladding failure.

8.2 FUEL ROD FAILURE

Intentionally defected UO_2 pellet and vipac fuels have been irradiated under conditions which produce fuel melting or near melting in the ETR P-7 loop,⁽¹¹⁻¹⁴⁾ the Fuel Element Rupture Test Facility (FERTF) in PRTR,⁽¹⁵⁾ and in the X-6 loop of the NRX Reactors in Canada.⁽¹⁶⁾ An intentionally defected Zircaloy-clad element containing Al-Pu alloy fuel was successfully irradiated in the ETR P-7 loop.⁽¹¹⁾ Intentionally defected mixed-oxide fuel rods have been successfully irradiated in the ETR M-3 loop.⁽¹⁷⁾

In Big Rock Point Reactor over a period of three years, 145 known failures were known to have occurred.⁽¹⁰⁾ Failures were gross and exposed large surfaces of UO_2 to the reactor water. In a number of failures there was significant loss of UO_2 to the reactor environment.

During this period of time fuel operated with a large volume of the UO_2 molten. Reference (10) did not state whether or not rods operating with molten cores had failed. The authors concluded that fuel failures were more of an economic problem than a safety problem.

In view of the operating experience with failed fuel rods in both test and commercial reactors, fuel failure is not considered a safety problem for this test.

9.0 QUALITY ASSURANCE

Quality assurance documentation will be submitted in another report.

10.0 DRAWINGS OF PBF FUEL ROD AND TEST ASSEMBLY

- NOTES
1. IDAHO NUCLEAR QUALITY ASSURANCE ESSENTIALITY LEVEL: III
 2. SURFACE ROUGHNESS TO BE $\sqrt{V}$ UNLESS OTHERWISE SPECIFIED.
 3. REMOVE ALL BURRS AND SHARP EDGES.
 4. DIMENSIONING AND TOLERANCING PRACTICES PER ANSI Y 14.5.
 5. WELD SYMBOLS PER AWS A2.0.
 6. GTAW WELDING PER INC-STD-90003, CLASS 2.
ELECTRON BEAM WELDING PER MIL-W-46132.
- Δ EACH LEVEL TO BE ROTATED 60°

QTY REQD	SYM	CODE IDENT	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL / SPECIFICATION	ITEM NO.
1			-1	BASKET	347 SST. ASTM A167	1
1			-3	END RING	304 SST. ASTM A276 OR A167	2
2			-4	LUG	304 SST. ASTM A276	3
1					ER 308 AWS A5.9	4

UNLESS OTHERWISE SPECIFIED		CONTRACT NO.	
DIMENSIONS AND TOLERANCES ARE IN INCHES		DATE	
DECIMALS: .X = .1 XXX = .003 XXXX = .0001		DRAWN: A.E. BRADFORD 3-15-71	
FRACTIONS: 1/16		CHECKED: [Signature] 3-22-71	
ANGULAR: ± 2°		DESIGN: [Signature] 3-14-71	
DO NOT SCALE DRAWING		REQUESTER: [Signature] 3-14-71	
SIMILAR TO		INDEX CODE NUMBER	
ACT. WT. / CALC. WT.		DRAWING NUMBER	
FORM 3		E 532064296400 402398	
ADDITIONAL APPROVALS		RELEASE DATE	
		3-22-71	

41101-330-120 1 INC 19

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APPENDIX A

GAMMA HEAT GENERATION CALCULATIONS

GAMMA HEAT GENERATION CALCULATIONSGAMMA HEAT GENERATION

The gamma heat generation is based on an average gamma heating rate of 1.0 watt/gm.

Dummy Rods

Dia = 0.559, Active length = 36"

$$= \frac{\pi}{4} (.559)^2 (36) \text{ in}^3 (7.92) \frac{\text{gm}}{\text{cm}^3} (16.39) \frac{\text{cm}^3}{\text{in}^3} (.001) \frac{\text{kW}}{\text{gm}}$$

$$= 1.12 \text{ kW/rod}$$

Fuel (UO₂-ZrO₂-CaO)

Dia = .559", Active length = 36"

$$\frac{\pi}{4} (.559)^2 (36) \text{ in}^3 (5.82) \frac{\text{gm}}{\text{cm}^3} (16.34) \frac{\text{cm}^3}{\text{in}^3} (.001) \frac{\text{kW}}{\text{gm}}$$

$$= 0.84 \text{ kW/rod}$$

Insulating Sleeve (ZrO₂)

OD = 0.694"

ID = 0.598"

Active length = 36"

$$\frac{\pi}{4} (.694^2 - .598^2) (36) \text{ in}^3 (5.6) \frac{\text{gm}}{\text{cm}^3} (16.39) \frac{\text{cm}^3}{\text{in}^3} (.001) \frac{\text{kW}}{\text{gm}}$$

$$= 0.32 \text{ kW/rod}$$

Cladding (304SS)

OD = .755"

ID = .695"

Active length = 36"

$$\frac{\pi}{4} (.755^2 - .695^2) (36) \text{ in}^3 (7.92) \frac{\text{gm}}{\text{cm}^3} (16.39) \frac{\text{cm}^3}{\text{in}^3} (.001) \frac{\text{kW}}{\text{gm}}$$

$$= 0.32 \text{ kW/rod}$$

Basket (304 SS)

$$OD = 3.313''$$

$$\text{Inside (Square)} = 2.375'' \times 2.375''$$

$$\text{Active Length} = 36''$$

$$\frac{\pi}{4} (3.313^2 - (2.375^2)) (36) \text{ in}^3 (7.92) \frac{\text{gm}}{\text{cm}^3} (16.39) \frac{\text{cm}^3}{\text{in}^3} (.001) \frac{\text{kW}}{\text{gm}}$$
$$= 13.93 \text{ kW}$$

Pressure Tube (Zr-2)

$$OD = 4.391''$$

$$ID = 3.441''$$

$$\text{Active length} = 36''$$

$$\frac{\pi}{4} (4.391^2 - 3.441^2) (36) \text{ in}^3 (6.55) \frac{\text{gm}}{\text{cm}^3} (16.39) \frac{\text{cm}^3}{\text{in}^3} (.001) \frac{\text{kW}}{\text{gm}}$$
$$= 22.58 \text{ kW}$$

APPENDIX B

THERMAL-HYDRAULIC CALCULATIONS

APPENDIX B

THERMAL-HYDRAULIC CALCULATIONSHYDRAULIC CHARACTERISTICS

The pressure drop versus flow rate characteristics for the basket-bore and basket-pressure tube annulus were calculated according to the following equation:

$$\Delta P = \left[\frac{\rho V^2}{2g(144)} \right] \left[\frac{fl}{D_e} + K_{exp} + K_{cont} \right]$$

where:

ΔP = Pressure drop, psi

ρ = Density of water, lbs/ft³

g = 32.2 ft/sec²

f = Friction factor

l = Length of flow channel, ft

D_e = Equivalent diameter, ft

K_{exp} , K_{cont} - Expansion and contraction loss coefficients

V = Velocity, ft/sec

The results of the calculations are presented in Figure 7.1-1.

FUEL ROD SURFACE AND COOLANT TEMPERATURESBulk Water Temperature Rise

$$\Delta t_b = \frac{q_c}{w C_p \rho}$$

Δt_b = Bulk water temperature rise

q_c = Heat transferred to coolant

w = Flow rate, lb/hr (loop flow rate = 75 gpm)

C_p = Specific heat of water at 600 psi and 225°F

ρ = Density of water at 600 psi and 225°F

$$\Delta t_b = \frac{231 \text{ kW} \times 3413 \text{ (Btu/hr)(gal/min) (ft}^3\text{)}}{(82)(\text{gal/min})(8.02)(\text{ft/hr})(\text{kW})(1.0)(\text{Btu/lb}^\circ\text{F})(59)(1\text{b})}$$

$$\Delta t_b = 21^\circ\text{F}$$

Loop Outlet Water Temperature (Inlet = 200°F)

$$t_b = t_{in} + \Delta t_b$$

$$t_b = 200^\circ\text{F} + 21 = 221^\circ\text{F}$$

With hot channel factor:

$$A_{dj} t_b = (\text{calc, } \Delta t_b) F_B + \Delta t_{add} + t_{in}$$

$$A_{dj} t_b = (21)(1.14) + 6 + 200$$

$$A_{dj} t_b = 230^\circ\text{F}$$

Basket Outlet Water Temperature

$$\begin{aligned} \text{Heat transferred to basket coolant} &= \text{Total} - (\text{Pressure tube} + 1/2 \text{ Basket}) \text{ heat} \\ &= 231 - (23 + 7) = 201 \text{ kW} \end{aligned}$$

Flow rate through basket = 57 gpm

$$\Delta t_b = \frac{(201)(3413)}{(57)(8.02)(1.0)(59)}$$

$$\Delta t_b = 25^\circ\text{F}$$

With hot channel factor:

$$A_{dj} t_b = (25)(1.14) + 6 + 200$$

$$A_{dj} t_b = 234^\circ\text{F}$$

Maximum Cladding Surface Temperature

$$t_s = t_b + \Delta t_f$$

$$\Delta t_f = \frac{(P'_{\max})(3413)(12)}{(\pi D)(h)} = \frac{\text{Surface Heat Flux}}{h}$$

$$h = b (V^{\cdot 8}/D_e^{\cdot 2}) \quad b = 475 \quad \text{From Reference 1, p. 52}$$

t_s = Cladding surface temperature, °F

t_b = Bulk water temperature, °F

Δt_f = Water film temperature drop, °F

$P'_{\max}$ = Maximum heat rate, kW/ft

D = Cladding diameter, in

h = Water film heat transfer coefficient, Btu/hr-ft²-°F

V = Coolant velocity, ft/sec

D_e = Flow passage equivalent diameter, in

b = Factor from Reference 1

Coolant Velocity (57 gpm)

$$V = \frac{(57)(.002228) (144) (\text{ft}^3)}{(2.376)^2 - [\frac{\pi}{4}(9)(.75)^2] (\text{ft}^2)(\text{sec})} = \frac{(57)(.002228)(144)}{1.669}$$

$$V = 11.0 \text{ ft/sec} \quad V^{\cdot 8} = 6.8$$

Basket Bore Equivalent Diameter

$$D_e = \frac{(4)(1.669)}{[(4)(2.376) + \pi (9)(.75)]}$$

$$D_e = 0.217 \text{ in} \quad D_e^{\cdot 2} = 0.737$$

Heat Transfer Coefficient

$$h = b(V^2/D_e^2)$$

$$h = (475)(6.8)/(.737)$$

$$h = 4400 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$$

Water Film Δt and Surface Heat Flux

$$\Delta t_f = \frac{(14.5)(3413)(12)}{(\pi)(.75)(4400)} = \frac{251,000}{4400}$$

$$\Delta t_f = 57^\circ\text{F}$$

$$\text{Surface heat flux} = 251,000 \text{ Btu/hr-ft}^2$$

Maximum Cladding Surface Temperature

With hot spot factor:

$$t_s = t_{in} + \Delta t_b \text{ (at hot spot)} + \Delta t_f F_h + \Delta t_{add}$$

$$t_s = 200 + 15 + (57)(1.39) + 6$$

$$t_s = 300^\circ\text{F}$$

The saturation temperature at 600 psi is 488°F .

HOT SPOT AND HOT CHANNEL FACTORS

	<u>Hot Channel Factor, F_b</u>	<u>Hot Spot Factor, F_h</u>	<u>Additive Factors ΔT Add</u>
A. Mechanical Tolerances (Hydraulic Effects)			
1. Effect on Mass Flow External Effects	1	1	
2. Effect on Mass Flow - Entrance and Exit Effects	1	1	
3. Effect Mass Flow Channel Effects	1	1	

	Hot Channel Factor, F_D	Hot Spot Factor, F_h	Additive Factors ΔT Add
4. Effect D Variation	1	1.01	
B. Mechanical Distortion			
1. Effect on Mass Flow External Effects	1	1	
2. Effect on Mass Flow Entrance Effects	1	1	
3. Effect Mass Flow Channel Effects	1	1	
4. Effect D Variation	1.01	1.01	
C. Correlation Equation on H	1	1.20	
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	
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	

	Hot Channel Factor, F_b	Hot Spot Factor, F_h	Additive Factors ΔT Add
2. Average Distribution	1	1	
3. Experimental Power Generation Variation Timewise	1.1	1.1	
H. Heat Flux Over Element Surface			
1. Fuel Homogeneity	1	1.01	
2. Mechanical Effects (Mechani- cal and Clad Thickness)	1	1	
	<u>1.14</u>	<u>1.39</u>	<u>6</u>

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