AFRRI

TECHNICAL REPORT

Maximum Temperature Calculation and Operational Characteristics of Fuel Follower Control Rods for the AFRRI TRIGA Reactor Facility

AFRRI TR91-1

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Opeational requirements of the Armed Forces Radiobiology Research Institute (AFRRI) TRIGA reactor necessitate the implementation of fuel follower control rods (FFCR's). This technical report discusses the operational and safety aspects of FFCR installation. Thermalhydraulic analysis shows that FFCR's can be operated safely in the AFRRI TRIGA reactor. The maximum calculated fuel temperature at the maximum steady-state power level is well within the limit of the maximum safe operating temperature set by the Technical Specifications.

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INTRODUCTION

Operational requirements of the Armed Forces Radiobiology Research Institute (AFRRI) TRIGA reactor facility necessitate the implementation of fuel follower control rods (FFCR's). Fuel follower control rods are like the standard TRIGA control rods as described in section 4.10.1 of the AFRRI TRIGA Safety Analysis Report (SAR) except that they have a fuel-filled follower rather than an air or aluminum follower. The primary purpose of the FFCR's is to offset the long-term effects of fuel burnup.

The Code of Federal Regulations (CFR; Title 10, Part 50.59) requires that modifications of a portion(s) of a licensed facility, as described in the facility SAR, be documented with a written safety analysis. The SAR ensures that all safety issues associated with the implementation of FFCR's have been reviewed. This technical report will show that implementing FFCR's will allow the standard control rods to function in their intended purpose and will restore core reactivity economically. FFCR's have been implemented in approximately a dozen TRIGA reactors and have been used for over 20 years without reported failure.

This report has been submitted to the AFRRI Radiation Facility Safety Committee to ensure that all safety questions have been reviewed before submission to the U.S. Nuclear Regulatory Commission (NRC), as required under 10 CFR 50.59.

GENERAL DESCRIPTION OF FUEL FOLLOWER CONTROL RODS

The current AFRRI TRIGA standard control rods were installed in 1964. The standard control rod consists of a sealed aluminum tube (0.065 inch thick) approximately 1.25 inches in diameter and 31 inches long. The upper 15.25 inches of the tube contain a compacted borated graphite rod (B_4C with 25-percent free boron or other boron compounds), which functions as a neutron absorber or poison. The lower end of the tube contains a 15.25-inch long and 1.125-inch diameter solid aluminum rod called the aluminum follower. The follower functions as a mechanical guide for the control rod as it is withdrawn from or inserted into the reactor core.

The proposed FFCR's differ from the current standard control rods in the following respects:

- The aluminum cladding is replaced by smooth stainless steel (SS304) cladding with a wall thickness of 0.020 inch. The inner and outer diameters are 1.085 inches and 1.125 inches, respectively.
- The length of the control rod is increased to 37.75 inches; the absorber and fuel follower section are both nominally 15 inches long.
- The outer diameter of the absorber section and the fuel follower are both 1.085 inches.
- The fuel follower has a solid zirconium rod as its central core with an outer diameter of 0.225 inch.

The absorber or poison material of the proposed FFCR's is, however, identical to the standard control rods presently installed.

The fuel contained in the FFCR consists of a fuel-moderator element in which zirconium hydride is homogeneously mixed with partially enriched uranium. The FFCR fuel element contains 12 percent uranium by weight and has a nominal enrichment of 20 percent in the ^{235}U isotope. The FFCR fuel element contains about 30.0 grams of ^{235}U--this is 79% of the ^{235}U loading of a standard AFRRI TRIGA fuel element. The nominal hydrogen-to-zirconium ratio in the FFCR fuel element is 1.7 with a range between 1.6 and 1.7. The FFCR fuel element contains no burnable poison. The stainless steel cladding on the FFCR fuel element has a hardness greater than the aluminum control rod guide tubes, so wearing will occur on the guide tubes rather than on the FFCR fuel elements.

FFCR MAXIMUM FUEL TEMPERATURE CALCULATION

A thermal-hydraulic analysis of the FFCR fuel element to determine the maximum fuel temperature uses the following model:

- The neutron mean free path for neutrons of all energies is smaller than the diameter of the TRIGA fuel rods, so the reactor must be treated as a heterogeneous reactor. Thus, the active volume of the core is taken to be the volume of fuel contained within the reactor core.
- The ratio of power in a fuel element with 12 wt-% uranium versus 8.5 wt-% uranium is 1.21. This is determined by General Atomics design calculations.[1]
- The reactor is operating at a steady-state power level of 1.0 MW, and the heat flux across the fuel element is described by Fourier's law of thermal conduction:[2]

$$q''(\mathbf{r}) = -k\nabla T(\mathbf{r}) \tag{1}$$

where
$q''(\mathbf{r})$ = heat flux at position $\mathbf{r}$
k = thermal conductivity
$T(\mathbf{r})$ = temperature at position $\mathbf{r}$

For steady-state heat transport, the heat production rate and the rate of energy loss due to heat transport are equal. This can be generally expressed as

$$q'''(\mathbf{r}) = \nabla\cdot q''(\mathbf{r}) \tag{2}$$

where
$q'''(\mathbf{r})$ = volumetric heat rate (heat production rate) at position $\mathbf{r}$.

Substituting equation (1) into equation (2) yields the time-independent equation of thermal conduction:

$$q'''(\mathbf{r}) = -\nabla\cdot k\nabla T(\mathbf{r}) \tag{3}$$

Equation (3) is, thus, the second-order ordinary differential equation that must be solved to determine the maximum temperature attained in the fuel portion of the FFCR.

Using this model to determine the maximum fuel temperature divides the analysis into two separate tasks: determining the power density in the FFCR in a D-ring grid position and solving equation (3) for the given power density.

Power Density in FFCR Fuel Element

The anticipated fuel loading for the AFRRI TRIGA reactor core with FFCR's installed will consist of 77 standard TRIGA fuel elements and the three FFCR fuel elements. Presuming that the control rods are fully withdrawn to achieve a power level of 1.1 MW, the total active fuel volume will be 30,597.9 cm^3. Thus, the average power density at 1.1 MW will be 36.0 W/cm^3.

The maximum fuel temperature is the important parameter, so only the radial variation of the core centerline power density is considered. To determine the maximum power density in the D-ring location of the FFCR fuel element, the following calculations are made:

For the AFRRI TRIGA, the radial and axial peak-to-average power ratios are 1.55 and 1.30, respectively.[3] Thus the maximum power density (heat rate) will be

$$\begin{aligned} q'''_{max} &= (1.55)(1.30)q'''_{ave} \\ &= 72.4 \text{ W/cm}^3 \end{aligned} \tag{4}$$

To determine $q'''_{D\text{-ring}}$ relative to q'''_{max}, it is useful to compute a scaling factor from the gross variation of thermal neutron flux in the radial direction (thermal flux and power density are directly proportional). The normalized radial flux distribution for the AFRRI TRIGA core is best represented by a Bessel function of the first kind of order zero:

$$\phi_{therm} = J_0\left(\frac{2.405r}{R_e}\right) \tag{5}$$

where
R_e = 21.78 cm, the extrapolated core radius
r = 11.99 cm, radial position of D-ring element

and the Bessel function scaling factor is J_o (1.3240) = 0.6074.

The power density for the D-ring is thus computed to be

$$q'''_{D\text{-ring}} = (0.6074)\ q'''_{max} = 44.0 \text{ W/cm}^3 \tag{6}$$

Because the FFCR fuel element differs from the standard fuel element in concentration of uranium, the power density in an FFCR fuel element is greater than the power density in a standard fuel element by a factor of 1.21.[1]

Taking the above scaling factor into account, the power density of an FFCR fuel element is found to be

$$q'''_{FFCR} = (1.21)q'''_{D\text{-}ring} \qquad (7)$$
$$= 53.2 \text{ W/cm}^3$$

Note that the calculation of q'''_{FFCR} takes into account the most limiting condition for power peaking in a 12 w/o fuel element versus an 8.5 w/o fuel element. As developed in equation (7), q'''_{FFCR} is still considerably less than the theoretical maximum q''' as determined in equation (4). A less conservative approach would have also accounted for the reduced volume in an FFCR fuel element.

Maximum Temperature in FFCR Fuel Element

Equation (3) takes the following form for cylindrical geometry with axial and azimuthal symmetry (see Figure 1):

$$\frac{1}{r}\left[\frac{d}{dr}kr\frac{dT}{dr}\right] + q''' = 0 \qquad (8)$$

The boundary conditions required that constrain equation (8) are as follows:

$$\frac{dT}{dr} = 0 \text{ at } r = R_i \text{ and } T = T_i \text{ at } r = R_i \qquad (9)$$

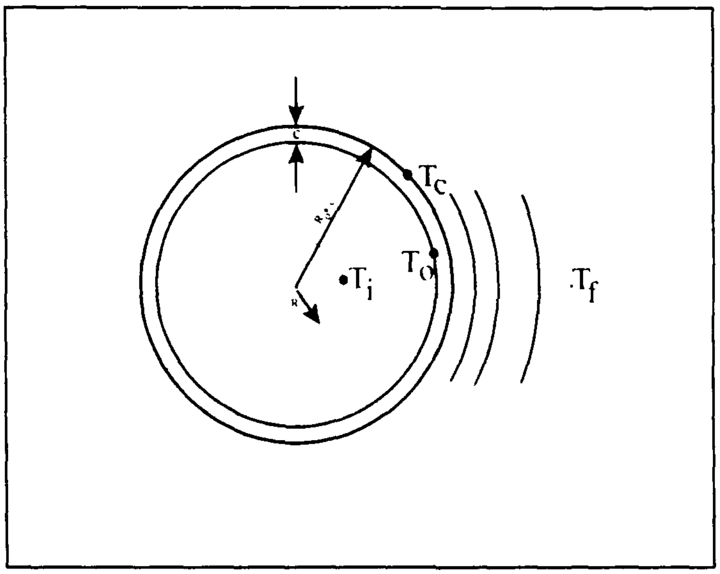

Figure 1. Cross-section of FFCR fuel element.

Integrating equation (8) twice yields a general solution of the form

$$T(r) = -q''' \frac{r^2}{4k_f} + C_1 \ln(r) + C_2 \qquad (10)$$

Applying the boundary conditions to solve for the temperature difference between the outer edge of the zirconium rod and the inner surface of the cladding,

$$C_1 = \frac{q'''R_i^2}{2k_f} \qquad (11)$$

$$C_2 = T_i - \frac{q'''R_i^2}{2k_f} (\ln(R_i) - \frac{1}{2})$$

$$T_i - T_o = \frac{q'''R_i^2}{4k_f} [(\frac{R_o}{R_i})^2 - 2\ln (\frac{R_o}{R_i}) - 1] \qquad (12)$$

To account for the transfer of heat from the fuel through the cladding to the coolant, we must consider the heat conduction between the inner and outer surfaces of the cladding, q_{clad}, and the heat conduction from the outer surface of the cladding to the coolant, q_{fluid}. We make the assumption that no heat is produced in the cladding or the coolant, so the heat conduction from the outer surface of the fuel, q_{fuel}, must be equal to q_{clad} and q_{fluid}. The heat conduction leaving the fuel is given by

$$q_{fuel} = \pi(R_o^2 - R_i^2)Lq''' = \pi((R_o/R_i)^2 - 1)LR_i^2q''' \qquad (13)$$

where
L = length of fuel element

$$q_{clad} = -k_cA \left.\frac{dT}{dr}\right|_{clad} = \frac{2\pi k_cL(T_o - T_c)}{\ln(\frac{R_o + c}{R_o})} \qquad (14)$$

$$q_{fluid} = hA(T_c - T_f) = 2\pi(R_o + c)Lh(T_c - T_f)$$

Note that the area, A, used in computing q_{clad} is the logarithmic mean area of the cladding. Recall that

$$q_{fuel} = q_{clad} = q_{fluid} \qquad (15)$$

So we can solve for the temperature differences in the above equations in terms of q''':

$$T_o - T_c = \ln\left(\frac{R_o + c}{R_o}\right)\left[\left(\frac{R_o}{R_i}\right)^2 - 1\right]\frac{R_i^2}{2}q'''$$

$$T_c - T_f = \frac{1}{(R_o+c)h}\left[\left(\frac{R_o}{R_i}\right)^2 - 1\right]\frac{R_i^2}{2}q''' \tag{16}$$

Adding equations (16) with equation (12) and solving for T_i gives us the expression for the maximum temperature in the FFCR.

$$T_i = T_f + \frac{q'''R_i^2}{4k_f}\left[\left(\frac{R_o}{R_i}\right)^2 - 2\ln\left(\frac{R_o}{R_i}\right) - 1\right] + \frac{q'''R_i^2}{2}\left[\left[\left(\frac{R_o}{R_i}\right)^2 - 1\right]\left[\frac{1}{k_c}\ln\left(\frac{R_o + c}{R_o}\right) + \frac{1}{h(R_o + c)}\right]\right] \tag{17}$$

where
T_i = maximum fuel temperature
T_f = bulk coolant temperature
R_o = 1.38 cm (radius of FFCR fuel element)
R_i = 0.286 cm (radius of Zr rod)
c = 0.051 cm (cladding thickness)
k_f = 0.18 W/cm-°C (thermal conductivity of UZrH)[4]
k_c = 0.138 W/cm-°C (thermal conductivity of SS304)[2]
h = 1.339 W/cm^2-°C (free convective heat transfer coefficient of water)
q''' = 53.2 W/cm^3 (from equation (7))

Note that the free convective heat transfer coefficient, h, was an experimentally derived quantity. The method by which h was determined is presented in Appendix A. Solving equation (17) using a volumetric heat rate of 53.2 W/cm^3 and a bulk water temperature of 48.6°C (the conditions at which h was determined) yields a maximum fuel temperature of 210.2°C. The maximum temperature achieved in the FFCR is nearly 180°C less than the normal temperature of 390°C in a standard fuel element in the B-ring during a 1.0 MW steady-state power operation.

Fuel Temperature in Pulse Mode Operation

The Nordheim-Fuchs model predicts the maximum fuel temperature achieved in a pulse mode operation.[5] The fundamental assumptions of this model are as follows:

- The neutron flux in the reactor is separated into a spatial component (shape factor) and a time-dependent component (amplitude factor), such that

$$\Phi(\mathbf{r},t) = vn(t)\Psi(\mathbf{r}) \tag{18}$$

where
v = neutron velocity
n(t) = neutron density (amplitude factor), proportional to power
$\Psi(\mathbf{r})$ = shape factor

The shape factor is assumed to remain constant during a pulse. This is called the point-reactor model.

- The production of delayed neutrons and the effects of source neutrons are neglected.

- The pulse from a thermodynamic standpoint is adiabatic, so

$$\frac{dT}{dt} = Kn(t) \tag{19}$$

where
T = fuel temperature
K = reciprocal of heat capacity

From the first and second assumptions we can write the time-dependent neutron density as

$$\frac{dn}{dt} = \frac{\rho - \beta}{\ell} n \tag{20}$$

where
ℓ = mean lifetime of neutrons in the reactor
β = delayed neutron fraction
ρ = reactivity

To account for a step insertion of reactivity, we can write

$$\rho = \rho_o - \alpha\Delta T \tag{21}$$

where
α = negative of the temperature coefficient of reactivity
ρ_o = step insertion of reactivity

Taking the derivative with respect to time of the above equation and substituting the result from equation (18),

$$\frac{d\rho}{dt} = -\alpha Kn \tag{22}$$

Applying the chain rule to equation (20) yields

$$\frac{dn}{d\rho} = -\frac{(\rho - \beta)}{\alpha K l} \tag{23}$$

Integrating equation (23) and solving for the constant of integration gives us the result

$$n = \frac{1}{2\alpha K1} \left[(\rho_o - \beta)^2 - (\rho - \beta)^2 \right] \tag{24}$$

The pulse is terminated when n becomes negligibly small. This occurs when

$$\rho = 2\beta - \rho_o \tag{25}$$

Equation (25) gives us the condition for the total energy release from the pulse, which manifests itself as a temperature rise in the fuel element when it is substituted into equation (21).

$$\Delta T_{core,ave} = \frac{2(\rho_o - \beta)}{\alpha} \tag{26}$$

Calculations by General Atomics show that a complete core of 12 w/o fuel would have a temperature coefficient of reactivity, α, that is 75% of the value for an 8.5 w/o fueled core.[1] The value for $\alpha_{8.5\ w/o}$ is taken to be -\$0.0118/°C. (This value was experimentally verified with a series of 23 pulses ranging from \$1.30 to \$2.00 that resulted in an average $\alpha_{8.5\ w/o}$ of -\$0.0128/°C--within 8.5% of the published value.) The effect of adding three 12 w/o FFCR's would, however, have a negligible effect on the overall *temperature coefficient of reactivity* for the entire core.[1]

Applying the Nordheim-Fuchs model for self-limiting power excursions, we can determine the maximum average increase in temperature for the entire core using equation (26). For a maximum allowed reactor pulse with a \$4.00 step insertion of reactivity, the maximum attained average temperature rise is calculated to be 333°C. Applying the power-peaking factors from the previous section, the maximum calculated temperature rise in an FFCR would be $1.48 \cdot \Delta T_{core,ave}$; the temperature rise for an FFCR for a \$4.00 pulse is calculated to be 493°C. Assuming an initial temperature of 25°C, the maximum temperature value would be 518°C. Note that even in the limiting case, neither the technical specification safety limit of 1000°C nor the limiting safety systems setting of 600°C is violated.

FFCR OPERATIONAL CHARACTERISTICS

FFCR's are a standard design offered as a stock item by General Atomics and have been used in several TRIGA reactors for over 20 years. FFCR's are currently implemented in approximately a dozen TRIGA reactors. There has been no reported evidence of fuel failure as a result of FFCR use in the United States. The operational issues to be resolved are the effects of burnup on the FFCR and the influence of FFCR's on the temperature coefficients of reactivity, shut-down margin, and rod worth.

FFCR control rod worth curves will be generated the same way that standard control rod worth curves are. Since the poison section of the FFCR will be the same as that of the currently installed standard control rods, the worth of the poison section of the FFCR will be that of the currently installed control rods. Measurements made by AFRRI reactor staff of control rod worth of the currently installed standard control rods yielded a nominal rod worth of \$1.90. The fuel follower is expected to add at least \$0.70 of reactivity[6] when the control rod is fully withdrawn, so the total rod worth for an FFCR is estimated to be \$2.60. The transient control rod and its follower have been measured and have a total nominal worth of \$4.01. The shutdown margin, as established by ANS/ANSI 15.1, is computed as follows:

Total rod worth		\$11.81
k_{excess} (maximum)	-	\$ 5.00
		\$ 6.81
Worth of TRANS rod	-	\$ 4.01
Shutdown margin		\$ 2.80

The shutdown margin with the most reactive control rod removed from the reactor is \$2.80--well in excess of \$0.50 minimum allowed value.

Once operational rod worth curves are established and power monitoring channels have been calibrated by the thermal power calibration method, power coefficient of reactivity curves will be generated. The issues regarding the measurement of shut-down margin and excess reactivity are addressed in Appendix B, Reactor Core Loading and Unloading.

Structural changes in the FFCR's will be monitored on an annual basis as part of the annual shutdown and maintenance. Specific effects to be monitored are the elongation and lateral bending of the fuel. FFCR fuel elements that have an elongation greater than 0.100 inch or a lateral bend greater than 0.0625 inch will be removed from service.

CONCLUSION

The analysis in this report shows that installing FFCR's in the AFRRI TRIGA reactor core will not result in an unsafe condition or violation of technical specifications. The primary parameter of interest, the maximum fuel temperature, was computed to be 210°C in the limiting case for steady-state operation and 518°C in the limiting case for pulse operation. Operational issues regarding maximum excess reactivity, shutdown margin, and burnup have also been addressed, and it has been determined that sufficient surveillance capabilities exist to prevent any unsafe or illegal condition.

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APPENDIX A: DETERMINATION OF FREE CONVECTIVE HEAT TRANSFER COEFFICIENT

Introduction

We can measure the bulk water temperature within the AFRRI TRIGA core to determine the average free convective heat transfer coefficient of the cooling water. This experiment involves inserting a temperature-measuring probe between the B- and C-ring fuel elements while the reactor is operating at a steady-state power level of 1.0 MW and measuring the water temperature at various axial positions. Once the bulk water temperature has been determined, Newton's law of cooling can be used to calculate the average free convective heat transfer coefficient.

Experimental Apparatus and Procedure

The equipment used in this experiment consists of two approximately 18-foot lengths of chromal-alumel thermocouple wires fused together at one end, encased in a 16-foot-long, 0.375-inch-diameter aluminum (Al) tube, and the thermocouple display readout on the AFRRI computerized reactor control console (Figure A-1).

The potential difference generated at the thermocouple junction as the water is heated by the reactor is amplified and displayed by the thermocouple circuitry in the AFRRI computerized reactor control console. The thermocouple is initially inserted into the core to correspond to position I. The thermocouple resides in each region for several minutes to allow it to attain thermal equilibrium. Once thermal equilibrium is attained, ten temperature readings are taken at 10-second intervals. After each temperature measurement, the thermocouple is withdrawn to the next position, and the temperature measuring procedure is repeated.

Figure A-2 shows that the temperature is measured in five axial positions: (I) 3 inches below midpoint (14 inches of thermocouple wire inserted into the core); (II) midpoint in axial dimension; (III) halfway between midpoint and bottom of graphite slug; (IV) at top of fuel region; (V) 1.5 inches above top of fuel region.

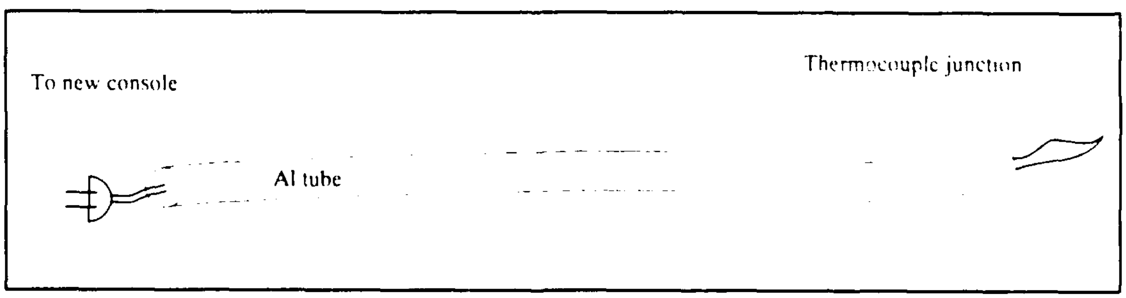

Figure A-1. Experimental apparatus.

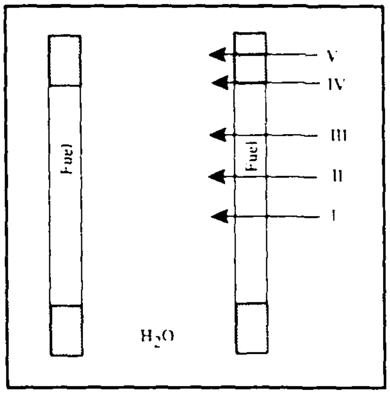

Figure A-2. Axial measuring points.

Safety Considerations

There are two safety considerations associated with this experiment: radiation streaming and an unintentional positive change in reactivity if the thermocouple wires are rapidly withdrawn from the reactor core while it is at power. Radiation streaming is avoided by flooding the aluminum tube with water and bending the tube so that it is at an angle not normal to the top of the core. The thermocouple wire displaces only 0.043 inch^3 of water when it is fully inserted in the core, so using the void coefficient of reactivity, the thermocouple wire represents a negative reactivity insertion of only 0.001 cents. If we were to estimate conservatively that the thermocouple wire had the same neutron-absorbing properties of a control rod, the maximum negative reactivity would be only 0.01 cents. Thus, there is no possibility of a reactivity accident associated with the apparatus used in this experiment.

Data

Table A-1 summarizes the data gathered during a 1.0 MW steady-state run of the AFRRI TRIGA reactor. The variation in the temperature measurements is most likely due to variance in the radial position of the temperature probe in the channel.

Table A-1. Bulk Water Temperature at Each Axial Position in the AFRRI TRIGA Reactor Core

Axial position	Inlet temp (°C)	Measured core bulk water temp (°C)
I	22	72.9
II	24	65.0
III	25	48.6
IV	26	51.6
V	27	59.7

Analysis and Conclusion

The purpose of this experiment is to determine the bulk water temperature within the core shroud; thus, it is the lowest measured value of the water temperature that is sought. Figure A-3 illustrates the temperature variation within a cooling channel.

Table A-1 shows that the measured value that most closely represents the bulk water temperature within the core shroud is 48.6°C.

The free convective heat transfer coefficient, h, is found by solving equation (8) for boundary conditions given by a standard TRIGA fuel element. Equation (A-1) gives the solution in terms of h.

$$h = \left(\frac{1}{r_o + c_o}\right)\left[\frac{(T_i - T_f) - \dfrac{q'''r_i^2}{4k_f}\left[\left(\dfrac{r_o}{r_i}\right)^2 - 2\ln\left(\dfrac{r_o}{r_i}\right) - 1\right]}{\dfrac{q'''r_i^2}{2}\left[\left(\dfrac{r_o}{r_i}\right)^2 - 1\right]} - \frac{1}{k_c}\ln\left(\frac{r_o + c_o}{r_o}\right)\right]^{-1} \qquad \text{(A-1)}$$

where

T_i = measured fuel temperature at 1.0 MW
T_f = measured bulk coolant temperature in the core
r_o = fuel outer radius, 1.816 cm
r_i = fuel inner radius, 0.229 cm
c_o = cladding thickness, 0.051 cm
k_f = thermal conductivity of fuel, 0.18 W/cm-°C
k_c = thermal conductivity of clad, 0.138 W/cm-°C
q''' = volumetric heat rate.

The measured fuel temperature in the B-ring at 1.0 MW steady-state power level is 390°C, and the calculated volumetric heat rate is 65.9 W/cm^3. Using the measured value of the bulk coolant temperature of 48.6°C yields a value of 1.339 W/C°-cm^2 for the free convective heat transfer coefficient.

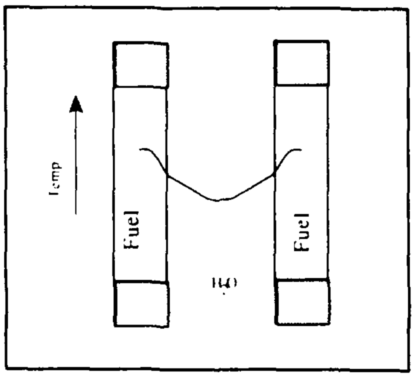

Figure A-3. Temperature variation within a cooling channel.

Newton's law of cooling expresses the linear relationship between the heat transfer rate, Q, and the temperature difference between the clad surface temperature, T_c, and the bulk water temperature, T_f, as

$$Q = hA(T_f - T_c)$$

where h is the overall convective heat transfer coefficient and A is the area of the fuel element.[7] The value of h determined for the AFRRI TRIGA is unique in that it takes into account the flow configuration, fluid properties, and the dimensions of the fuel elements. Assuming that the dependence of Q on the temperature difference $T_f - T_c$ is roughly linear, then the value of h, computed using data from B-ring elements with their higher heat transfer rate and local temperature difference, will be close to the value for h for D-ring positions.

APPENDIX B: REACTOR CORE LOADING AND UNLOADING

General

Loading and unloading of the reactor core shall be performed under the supervision of the Reactor Facility Director or the Reactor Operations Supervisor.

Specific

1. Setup

a. Ensure that at least one nuclear instrumentation channel is operational.

b. Ensure that the source is in the core.

c. Ensure that an operator monitors the reactor console during all fuel movements.

d. Check new FFCR's before insertion into the core; this includes cleaning, visual inspection, and length and bow measurements.

e. Install all control rods.

f. If irradiated fuel elements are to be removed unshielded from the pool, obtain a Special Work Permit (SWP) from the Safety and Health Department (SHD); do not remove fuel elements with a power history (greater than 1 KW) in the previous 2 weeks from the reactor pool.

2. Core Loading

a. After each step of fuel movement perform the following:

(1) Record detector readings.

(2) Withdraw control rods 50%; record readings.

(3) Withdraw control rods 100%; record readings.

(4) Calculate 1/M.

(5) Plot 1/M versus number of elements (and total mass of ^{235}U).

(6) Predict critical loading.

(7) Insert ALL rods; continue to next step.

b. Load elements in the following order:

(1) Load the B-ring and C-ring thermocouple elements.

(2) Connect thermocouple outputs to reactor control console display.

(3) Install any other thermocouple elements.

(4) Complete loading of B- and C-ring elements (total of 18 standard elements plus 3 FFCR's).

(5) Load D-ring (total of 33 standard elements plus 3 FFCR's).

(6) Load the following E-ring elements in order:

16, 17, 18, 20, 6, 8, 9, 10 (total of 41 standard elements plus 3 FFCR's).

(7) Complete the E-ring by loading the following elements in order:

15, 21, 11, 5, 14, 22, 4, 12, 13, 1 (total of 51 standard elements plus 3 FFCR's).

(8) Load the following F-ring elements in two elements per step until criticality is achieved, using the following loading order:

22, 23, 24, 21, 20, 25, 26, 27, 28, 29, 30, 1, 2, 3, 4, 5, 19, 18, 17, 16, 15, 14, 13, 6, 12, 7, 11, 8, 10, 9.

Once criticality has been achieved, perform control rod worth measurements at core position 500 by rod drop technique. Calculate shutdown margin (SDM):

$$\text{SDM} = \text{total control rod worth} - K_{\text{excess}} - \text{TRANS rod worth}$$

(9) Load core to \$2.00 excess reactivity by loading two elements per step using the loading order in instruction 8.

(10) Verify control rod worth using rod drop techniques; calculate SDM.

(11) Load the core to achieve a K_{excess} that will allow calibration of the TRANS rod based on the last available worth curve of the TRANS rod (approximately \$4.00). Calculate the reactivity value of each element as it is added.

(12) Calibrate all control rods.

(13) Calculate SDM.

(14) Estimate K_{excess} with a fully loaded core (must not exceed \$5.00).

(15) Load core to fully operational load using loading order in instruction 8, and recalibrate all control rods. Calculate SDM.

(16) Adjust the core loading pattern to meet operations requirements if necessary. Recalibrate all control rods. Calculate SDM.

3. Core Unloading

a. Unload the reactor core starting with the F-ring and ending with the B-ring.

b. Remove the fuel elements individually from the reactor core, identify them by serial number, and place them in the fuel storage racks or a shipping cask.

c. If elements are to be loaded into a shipping cask, clean the cask completely, and check for radiological contamination before placing the cask in or near the pool. Load cask in accordance with procedures specific to the cask.

d. Once the cask is loaded, perform an air sample and survey; check temperature and pressure inside cask, if necessary.

e. If elements are placed in temporary storage away from core monitoring, ensure that criticality monitoring in accordance with 10 CFR 70 is in place.

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