ATTACHMENT 2

FUEL FOLLOWER CONTROL ROD SAFETY ANALYSIS FOR THE AFRRI TRIGA REACTOR FACILITY

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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. Such documentation ensures that all safety issues associated with the change are reviewed. Based on the analysis in this report, the authors have determined 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.

The proposed modifications require minor administrative changes to the technical specifications of the current reactor license (R-84) and the SAR. These administrative changes are specified in Appendix B of this report.

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 USNRC, 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 in thick) approximately 1.25 in in diameter and 31 in long. The upper 15.25 in of the tube contains 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-in long and 1.125-in 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 The follower functions as a mechanical guide for the control rod as it is withdrawn from or inserted to the reactor core.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 in. The inner and outer diameters are 1.085 in and 1.125 in, respectively.
- The length of the control rod is increased to 37.85 in; the absorber and fuel follower section are both 15 in long.
- The outer diameter of the absorber section and the fuel follower are both 1.085 in.

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 31.4 grams of ^{235}U--this is 83% of the ^{235}U loading of a standard AFRRI TRIGA fuel element.

SAFETY ANALYSIS OF FFCR IMPLEMENTATION

The two principal safety issues that must be addressed are the maximum excess reactivity limit of $5.00 set by Technical Specification 3.1.3.(a) and the maximum fuel temperature of 600°C set by Technical Specification 2.2. With regard to the maximum excess reactivity limit, Reactor Operating Procedure VII, Reactor Core Loading and Unloading (Appendix D), ensures that the $5.00 limit on excess reactivity is not breached.

A thermalhydraulic 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 heterogenous 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]

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

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

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

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:

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

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 1.0 MW, the total active fuel volume will be 30,597.9 cm^3. Thus, the average power density at 1.0 MW will be 32.7 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

$$\begin{aligned} q'''_{max} &= (1.55)(1.30)q'''_{ave} \\ &= 65.9 \ W/cm^3 \end{aligned} \qquad (4)$$

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:

$$\phi_{therm} = J_o\left(\frac{2.405\ r}{Re}\right) \qquad (5)$$

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

and scaling factor $= J_o\ (1.3240) = 0.6074$

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

$$q'''_{D\text{-ring}} = (0.6074)\ q'''_{max} = 40.0\ W/cm^3 \qquad (6)$$

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

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

$$\begin{aligned} q'''_{FFCR} &= (1.21)q'''_{D\text{-ring}} \\ &= 48.4\ W/cm^3 \end{aligned} \qquad (7)$$

Maximum Temperature in FFCR Fuel Element

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

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

which has a general solution of the form

$$T(r) = -q''' \frac{r^2}{4k_f} + C_1 \ln(r) + C_2 \tag{9}$$

The boundary conditions required to solve for the constants of integration in equation (9) are as follows:

$$\frac{dT}{dr} = 0 \text{ at } r = 0 \quad \text{and} \quad T = T_m \text{ at } r = 0 \tag{10}$$

Parameters of interest are represented in figure 1. The solution to equation (8) takes the form

$$T_m - T_f = \frac{q'''R^2}{4k_f} + \frac{q'''R^2}{2}\left[\frac{1}{k_c}\ln\left(\frac{R+c}{R}\right) + \frac{1}{h(R+c)}\right] \tag{11}$$

where

T_m = maximum centerline fuel temperature
T_f = coolant temperature
R = 1.38 cm (radius of FFCR fuel element)
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)

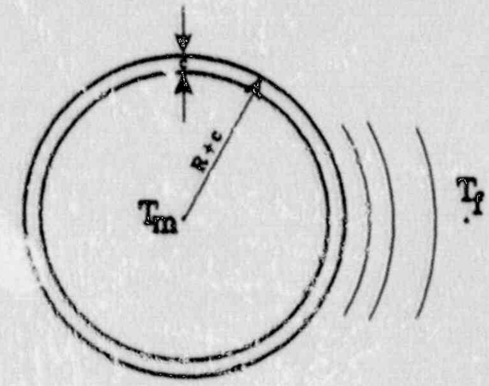

Figure 1. Cross-section of FFCR fuel element.

It should be noted 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 (11) using a volumetric heat rate of 48.4 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 212.4°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 fully withdrawn FFCR tends to run cooler than the standard TRIGA fuel elements in the D-ring. Solving equation (8) by applying the boundary conditions of the annular design (with respect to heat production) of the standard TRIGA fuel element with heat flow out of the outer surface, the maximum calculated temperature is 255.8°C. Thus, during pulse operations, we can expect the FFCR to attain a lower peak temperature than the standard TRIGA fuel elements in the same fuel ring. The maximum temperatures measured in the instrumented TRIGA fuel element in the B-ring for $3.50 and $2.00 pulses are 532.5°C and 267.0°C respectively, so it is expected that the FFCR temperature behavior will be well within technical specifications limits.

CONCLUSION

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 utilization.

The neutronic characteristics for the TRIGA core will remain unchanged[1] as the prompt negative temperature coefficient of the 12 wt-% fuel followers is virtually the same as that of the standard 8.5 wt-% TRIGA fuel element. Because the maximum temperature achieved in a FFCR fuel element at 1.0 MW is nearly 790°C less than the AFRRI TRIGA technical specification limit of 1000°C, the FFCR's can be safely implemented with no danger of damage to the FFCR fuel element cladding.

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-ft lengths of chromal-alumel thermocouple wires fused together at one end, encased in a 16-ft-long, 3/8-in-diameter aluminum (Al) tube, and the thermocouple display readout on the AFRRI computerized reactor control console (Figure A-1).

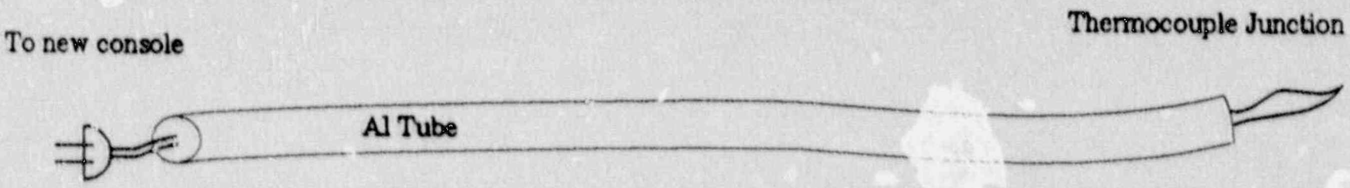

Figure A-1. Experimental apparatus.

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-sec 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 in below midpoint (14 in 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 in above top of fuel region.

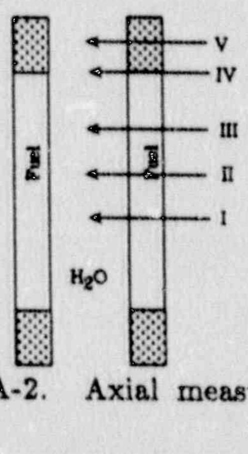

Figure A-2. Axial measuring points.

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 will displace only 0.043 in^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 make the conservative estimate 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/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.

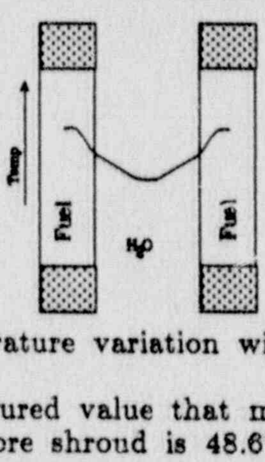

Figure A-3. Temperature variation within a cooling channel.

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/cm^2-°C for the free convective heat transfer coefficient.

APPENDIX B. REQUESTED ADMINISTRATIVE CHANGES

Changes to the AFRRI TRIGA technical specifications as effected by the implementation of FFCR's are as follows:

Section 1.0 - Definitions

Page 2, Section 1.9: Replace the definition for a fuel element in its entirety to accommodate fuel follower control rods as follows:

1.9 FUEL ELEMENT

A fuel element is a single TRIGA fuel rod, or the fuel portion of a fuel follower control rod.

Section 2.0 - Safety Limits and Limiting Safety System Settings

No changes.

Section 3.0 - Limiting Conditions for Operations

No changes.

Section 4.0 - Surveillance Requirements

Page 22, Section 4.2.5: Replace the Specifications in its entirety as follows to clarify the requirement for fuel element surveillance and to accommodate the fuel follower control rods:

All the fuel elements present in the reactor core, to include fuel follower control rods, shall be inspected for damage or deterioration, and measured for length and bow at intervals separated by not more than 500 pulses of insertion greater than $2.00 or annually (not to exceed 15 months), whichever occurs first. Fuel elements in long-term storage need not be measured until just prior to being returned to core; however fuel elements routinely moved to temporary storage shall be measured every 500 pulses of insertion greater than $2.00 or annually (not to exceed 15 months), whichever occurs first.

Section 5.0 - Design Features

Pages 25, Section 5.2.1: Replace the Applicability to accommodate fuel follower control rods with the following:

Applicability

These specifications apply to the fuel elements, to include fuel follower control rods, used in the reactor core.

Page 26, Section 5.2.1: Replace part (a) of the Specifications in its entirety to accommodate fuel follower control rods with the following:

a. Uranium content: Maximum of 9.0 weight percent enriched to less than 20% uranium-235. In the fuel follower, the maximum uranium content will be a maximum of 12.0 weight percent enriched to less than 20% uranium-235.

Page 26, Section 5.2.1: Add the following paragraph to the Basis to accommodate fuel follower control rods:

The power density of a 12.0 weight percent fuel follower element of the same diameter as a control rod, which is smaller than the standard TRIGA element, will produce the same power density in the local area as the standard 8.5 weight percent TRIGA elements due to its increased hydraulic diameter.

Page 27, Section 5.2.3: Replace part (a) of the Specifications in its entirety to accommodate fuel follower control rods with the following:

a. The standard control rods shall have scram capability, and shall contain borated graphite, B4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless-steel cladding. These rods may have an aluminum, air, or fuel follower. If fuel followed, the fuel region will conform to the Specifications of 5.2.1.

Section 6.0 - Administrative Controls

No changes.

The following modification paragraph of 4.10.1 of the Safety Analysis Report is required to accommodate the implementation of the fuel follower control rods:

4.10.1 Standard Control Rods and Guide Tubes

The shim rod (SHIM), safe rod (SAFE), and regulating rod (REG) constitute the three standard control rods and are located in core positions D-1, D-7, D-13, respectively (Figure 4-3). A standard control rod (Figure 4-7) consists of a sealed stainless steel tube (0.020-inch thick) approximately 37.85 inches long and 1.125 inches in diameter. The upper half of the tube contains a 15 inch long compacted borated graphite rod (25 percent free boron or boron compounds) as the neutron absorber, or poison. The lower end of the tube (the follower) contains a solid 15 inch long, 1.085 inch diameter rod of UZrH fuel which is is 12 w-% in uranium and has a nominal enrichment of 20 %. The control rod guide tubes are attached to the lower grid plate and provide space for inserting and withdrawing the control rods and pass through the upper and lower grid plates.

Figure 4-7 will be modified to reflect the implementation of FFCR's.

APPENDIX C. RESULTS OF RRFSC REVIEW

The FFCR safety analysis and proposed technical specification changes were reviewed by the full Reactor and Radiation Facility Safety Committee (RRFSC) on March 27, 1990. The result of this review was unanimous approval for the required technical specification changes and installation of FFCR's.

APPENDIX D. PROCEDURE FOR REACTOR CORE LOADING AND UNLOADING

Revised: March 1990

PROCEDURE VII 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. These procedures are superseded during CET Operations (see Procedure I, Tab B) and during annual shutdown maintenance (see the current Annual Shutdown Checklist).

Specific:

1. Setup
   a. Ensure that at least one nuclear instrumentation channel is operational.
   b. Ensure that the source is in core.
   c. Ensure that an operator monitors the reactor console during all fuel movements.
   d. Check new fuel elements before insertion into the core; this includes cleaning, visual inspection, and length and bow measurements.
   e. 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 U-235).
      (6) Predict critical loading.
      (7) Insert ALL rods; continue to next step.
   b. Load elements in the following order:
      (1) Load the B-ring thermocouple element.
      (2) Load the C-ring thermocouple element.
      (3) Install temperature measurement system (to measure fuel temperature).
      (4) Install any other thermocouple elements.
      (5) Complete loading of B- and C-ring elements (total of 18 standard elements plus 3 FFCR's).
      (6) Load D-ring (total of 33 standard elements plus 3 FFCR's)
      (7) Load the following E-ring elements in order:
          16, 17, 18, 20, 6, 8, 9, 10 (total of 41 elements plus 3 FFCR's).
      (8) Complete the E-ring by loading the following elements in order:
          15, 21, 11, 5, 14, 22, 4, 12, 13, 1 (total of 57 standard elements plus 3 FFCR's)
      (9) 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.

(10) Load core to \$2.00 excess reactivity by loading two elements per step using the loading order in instruction 9.
(12) Estimate control rod worth using rod drop techniques.
(13) Measure control rod worth; calculate value of remaining elements as they are added.
(14) 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).
(15) Calibrate the TRANS rod.
(16) Estimate the shutdown margin.
(17) Estimate K-excess with a fully loaded core (must not exceed \$5.00).
(18) Load core to fully operational load using loading order in instruction 9, and recalibrate all control rods.

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.

REFERENCES

1. General Atomics, letter to Mr. Mark Moore on fuel follower control rods, 28 October, 1988.

2. El-Wakil, M.M., Nuclear Heat Transport, The American Nuclear Society, Lagrange Park, IL, 1978.

3. Defense Atomic Support Agency, AFRRI/USAEC Facility License R-84, Complete with Applications and Amendments, Bethesda, MD, 1962.

4. Wallace, W.P. and Simnad, M.T., Metallurgy of TRIGA Fuel Elements, GA-1949, General Atomic, San Diego, CA, 1961.