10 CFR 50.90

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June 28, 2013

Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE APPLICATION FOR LICENSE RENEWAL (TAC NO. ME1587)

Sir:

By letter dated May 3, 2013, the Nuclear Regulatory Commission requested additional information necessary to allow processing of our research reactor license renewal application (License R-84, Docket 50-170). Answers to those questions are enclosed. In addition, following a discussion with our license renewal Project Manager at the AFRRI facility on June 27, 2013, a correction to Table 4-19 from Chapter 4 of the proposed Safety Analysis Report submitted on March 4, 2010 is enclosed.

If you need further information, please contact Mr. Steve Miller at 301-295-9245 or stephen.miller@usuhs.edu.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge. Executed on June 28, 2013.

L. Andrew Huff

L. Andrew Huff
COL, USAF
Director

A020
NRR

Financial Qualifications

1. *Pursuant to 10 CFR 50.33(f) (2), "[t]he applicant shall submit estimates for total annual operating costs for each of the first five years of operations of the facility." Since the information included in the previous correspondence was for the period of fiscal years (FYs) 2012 through 2016, please provide the following additional information:*

    (a) *Projected operating costs of the AFRRI facility for each of the FY2013 thru FY2018 (the first five year period after the projected license renewal). If the cost estimates have not changed since the previous submittal for the period of FY2013 through FY2016, please so state.*

The cost estimates for FY 2013 through FY 2016 remain the same.

Estimated operating costs ($ millions) to support AFRRI for FY 2017 and 2018 are:

FY	AFRRI	TRIGA reactor
2017	$15.7	$1.3
2018	$16.0	$1.4

(b) *Has the source(s) of funding to cover the operating costs for the above FYs changed since the August 13, 2010, submittal?*

No

2. *By letter dated August 13, 2010, you provided an updated decommissioning cost estimate for the facility that was developed using NUREG/CR-1756, "Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors." The decommissioning cost estimate was $14.831 million in 2011 dollars. The cost estimate summarized costs by labor, radioactive wastes disposal, energy, and a 25-percent contingency factor.*

    (a) *Please indicate if the basis for how the cost estimate was developed has changed. If NUREG/CR-1756 is still the basis, please so state.*

NUREG/CR-1756 remains the basis.

    (b) *Please indicate if there are any changes to the means of adjusting the cost estimate and associated funding level periodically over the life of the facility.*

No changes.

3. *AFRRI provided a Statement of Intent (SOI), dated August 11, 2010, stating that "[f]unding will be sought from the [U.S.] Department of Defense in accordance with established programming and budgeting procedures," per 10 CFR 50.75(e)(1)(iv).*

   (a) Please indicate if there have been any changes to the SOI and if decommissioning funding obligations of the AFRRI facility continue to be backed by the full faith and credit of the U.S. Government.

There are no changes to the SOI and decommissioning funding obligations of the AFRRI facility remain backed by the full faith and credit of the U.S. Government.

Technical Specifications

1. *Technical Specification (TS) 4.1 (a) states, "the reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months or following any significant core configuration changes". The term "significant core configuration changes" is not defined in your submissions. Please submit a definition for significant core configuration changes or justify why it is not necessary.*

The following definition has been added to TS 1.0:

> CORE CONFIGURATION: The core configuration includes the number, type, or arrangement of fuel elements and standard control rods/transient rod occupying the core grid.

TS 4.1 (a) has been changed and now reads:

> The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months or following any significant (>$0.25) core configuration changes.

2. *TS 4.1 (e) states, "The core excess reactivity shall be measured at the beginning of each day of operation involving the movement of control rods, or prior to each continuous operation extending more than a day. During extended shutdown periods, the core excess reactivity shall be measured at least annually, not to exceed 15 months." This TS conforms to the recommendation in American Nuclear Society/American National Standards Institute, Inc. (ANS/ANSI) 15.1 except that ANS/ANSI 15.1 states "and following significant core configuration and/or control rod changes". Please justify why it is not necessary to measure core excess reactivity following significant core configuration changes and/or control rod changes or submit a change to your TSs.*

TS 4.1 (e) has been changed and now reads:

> The core excess reactivity shall be measured at the beginning of each day of operation involving the movement of control rods, or prior to each continuous operation extending more than a day, and following any significant (>$0.25) core configuration changes. During extended reactor shutdown periods, the core excess reactivity shall be measured at least annually, not to exceed 15 months.

3. *A description of measurements of the fuel elements and fuel follower control rods is included in TS 4.2.5 and 5.2.2(e).*

   (a) *TS 4.2.5 states, in part, "Fuel elements and fuel follower control rods indicating an elongation greater than 0.100 inch, a lateral bending greater than 0.0625 inch, or significant visible damage shall be considered damaged, and shall not be used in the reactor core." This statement appears to be a limiting condition for operation (LCO). Please justify why this TS is a surveillance and not an LCO or submit a change to the TSs.*

TS 3.9 has been changed and now reads:

3.9. FUEL PARAMETERS

Applicability

This specification applies to all TRIGA fuel elements and fuel follower control rods.

Objective

The objective is to maintain integrity of the fuel element cladding.

Specification

The reactor shall not operate with damaged fuel elements or fuel follower control rods, except for the purpose of locating damaged fuel elements or fuel follower control rods. A fuel element or fuel follower control rod shall be considered damaged and must be removed from the core if:

a. The transverse bend exceeds 0.0625 inches over the length of the cladding;

b. The length exceeds its original length by 0.100 inches;

c. A cladding defect exists as indicated by the release of fission products; or

d. Visual inspection identifies bulges, gross pitting, or corrosion.

Basis

Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation and bend limits are the values found acceptable to the USNRC (NUREG-1537).

TS 4.2.5 has been changed and now reads:

4.2.5. <u>REACTOR FUEL ELEMENTS AND FUEL FOLLOWER CONTROL RODS</u>

<u>Applicability</u>

This specification applies to the surveillance requirements for the fuel elements and fuel follower control rods.

<u>Objective</u>

The objective is to verify that the specifications for fuel elements and fuel follower control rod conditions are met.

<u>Specification</u>

Fuel elements and fuel follower control rods shall be inspected visually for damage or deterioration and measured for length and bend in accordance with the following:

a. Before being placed in the core for the first time or following long-term storage;

b. Every two years (not to exceed 30 months), or at intervals not to exceed 500 pulses of insertion greater than $2.00, whichever comes first, for elements in the B, C, and D rings and for fuel follower control rods;

c. Every four years (not to exceed 54 months) for elements in the E and F rings;

d. If damage, deterioration, or unacceptable length or bend measurements are found in one or more fuel elements or FFCRs, all fuel elements and FFCRs in the core shall be inspected for damage or deterioration and measured for length and bend.

<u>Basis</u>

The frequency of inspection and measurement is based on the parameters most likely to affect the fuel cladding of a pulse reactor. Inspecting fuel elements in rings with higher power factors will provide early indication of fuel damage while significantly reducing the amount of fuel movement required.

(b) TS 5.2.2(e), Reactor Core, states that "fuel elements indicating an elongation greater than 0.100 inch, a lateral bending greater than 0.0625 inch, or significant visible damage shall be considered damaged, and shall not be used in the reactor core." Please justify why this is a design feature and not an LCO.

See response to RAI 3 (a).

4. *The safety analysis report states that the calculation of the maximum temperature during the loss of coolant accident (LOCA) vs. fuel rod power density during operation was based on decay heating after operation at 72 hours per week for 40 years. What is the basis for this assumption-is there a limit which caps the number of operating hours per week? Please explain why a Technical Specifications constraint of 72 hours is not necessary or propose a Technical Specification.*

A previous RAI was submitted to address this issue. In summary, there are two analyses that address the LOCA at AFRRI.

The bounding analysis is based on General Atomics Report No. E-1 17-196. This report identifies the ability of natural convection to maintain fuel cladding temperature below 900°C following infinite operation and instantaneous coolant loss, as long as a power density of 21 kW per element is not exceeded. Operation of the AFRRI TRIGA at a full power of 1.1 MW yields a peak power density of 19.4 kW in the B04 fuel element position. Given that the B04 location represents the highest power density within the AFRRI core, this scenario is bounded by the GA analysis.

However, infinite operation of the AFRRI TRIGA reactor is a purely hypothetical scenario. Fission product build up within the AFRRI core limits full power operation to less than 24 continuous hours, at which point the negative reactivity from fission products prevent operation at full power. Historically, the AFRRI TRIGA has operated an average of approximately 500 kW-hrs per week. In order to provide a more accurate and useful analysis of a LOCA at the AFRRI TRIGA reactor, a comparison was made to a LOCA following 1 MW operation for 72 hours per week for 40 years. Although this analysis is still substantially higher than the run history at AFRRI, it further indicates the margin of safety when considering a LOCA. In addition to this, a comparison to the Oregon State TRIGA Reactor (OSTR) SAR calculations for a power history of 70 MW-hrs per week for 40 years was also discussed.

Previous RAI response:

As described in 13.2.1 of the AFRRI SAR, the reactor fuel elements rely on ambient air natural convection through the core to cool the reactor fuel in the event of a loss-of-coolant accident (LOCA). A buoyancy force to drive this natural convection is developed by a hot air column within the core and a cooler column of air outside of the core. This accident has been analyzed for an instantaneous loss of water and a loss of water occurring over a 15 minute period. In addition, this analysis includes scenarios for infinite full power operation and 72 hour per week operation over a 40 year period.

If all reactor coolant was suddenly lost, the primary concern would be the integrity of the fuel cladding. Maintaining a fuel cladding temperature below the NUREG-1282 specification of 950°C ensures that the cladding maintains sufficient strength to prevent failure under the pressure of hydrogen gas buildup within the element. Therefore, operating conditions must be such that in the event of a LOCA the fuel cladding temperature will not exceed 950°C.
General Atomics Report No. E-1 17-196 provides a detailed analysis showing that air natural convection cooling is adequate to maintain fuel cladding below 900°C assuming power levels no higher than 21 kW per element are achieved. This calculation assumes an instantaneous loss of all coolant and infinite operation at 1 MW. If a 15 minute delay

between reactor scram and total coolant loss is assumed, fuel cladding will remain below 900°C up to a power level of 22 kW per element. Analysis provided in the Oregon State TRIGA Reactor SAR shows that if reactor operating history is assumed to be 70 MW-hrs per week over a 40 year period, these limits for maximum power level per element increase to 25.2 and 26.4 kW, respectively.

As described in 13.2.1.3 of the AFRRI SAR, assuming an operational history of 72 MW-hrs per week for 40 years, maximum fuel cladding temperatures following LOCA are 548°C assuming instantaneous coolant loss and 477°C assuming a 15 minute delay time. These calculations assume operation at 1 MW.

The current maximum licensed steady state power of the AFRRI reactor is 1.1 MW. Section 4.5.8 of the AFRRI SAR discusses the power peaking within the 85-3 core and determines the highest power factor of 1.552 to occur in position B04. Assuming 1.1 MW with 88 fuel elements, the B04 element has a power of 19.4 kW. This is within the 21 kW per element limit set for the worst-case LOCA, and substantially less than the more representative 26.4 kW per element limit. In reality, AFRRI's operating history is far below 70 MW-hours per week, with a historical average closer to 500 kW-hours per week.

5. *TS 1.10, the definition of Initial Reactor Startup is, "The first reactor startup following fuel element relocation within the core." This definition does not conform to the recommendations of NUREG-1537 which suggests that initial reactor startup is defined as "the first startup after the reactor is secured." Please justify why your definition is acceptable or submit a change to the TSs.*

The definition of Initial Reactor Startup (TS 1.10) and Reactor Startup (TS 1.26) have been removed.

6. *TS 1.24, the definition of Reactor Secured is, "Either sufficient fuel is removed to ensure a $1.00 (or greater) shutdown margin ... " does not conform to the recommendations of ANSI/ANS 15.1 which state "Either there is insufficient moderator available in the reactor to attain criticality ... " Please justify why this is acceptable or submit a change to the TSs.*

TS 1.24 (a) has been changed and now reads:

a. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

7. *TS 3.1.3(b), Reactivity Limitations states, "The shutdown margin provided by the remaining control rods with the most reactive control rod fully withdrawn shall be $3.50 ... " However, the definition for shutdown margin uses the term "the most reactive position." Is the term "the control rod fully withdrawn" equivalent to "the most reactive position" and do the two terms refer to the same position? Please justify why different terms were used, and why this is acceptable or submit a change to the TSs.*

TS 3.1.3(b) has been changed and now reads:

> The shutdown margin provided by the remaining control rods with the most reactive rod in the most reactive position shall be greater than ...

NOTE: The initial NRC question refers to TS 3.1.3(b). It is our understanding that there is a typo in the question and that $3.50 should actually be $0.50. The response assumes that this question refers to a $0.50 shutdown margin.

8. *TS 3.5.2, Effluents, Argon-41 Discharge Limit, and TS 6.6(b), Operating Reports, Gaseous Waste, indicate that Argon-41 is the only effluent measured. Please justify why other effluents are not measured and indicate how samples are obtained as well as the method used to determine that the samples are statistically representative.*

During normal reactor operations, Ar-41 and N-16 are the only airborne radioisotopes produced. Given the short half-life of N-16 (~7 seconds), the dose to members of the public from N-16 production in the AFRRI reactor pool is insignificant.

A description of the AFRRI Stack Gas Monitoring System (SGM) is found in Section 7.7.3.2 of the AFRRI SAR. Briefly, a sample of air exhausting from the reactor room and exposure rooms is directed through a NaI scintillation detection system located downstream of the reactor absolute air filters. Air samples are collected through a series of tubes within the stack. These tubes traverse the entire cross section of the stack and pull air from numerous locations within the same planar space. The alarms on the detector have local and remote readouts to alert reactor personnel if higher than normal radioactive effluent levels are present in the exhausting air.

9. *TS 3.6(b), Limitations on Experiments states, "Each fueled experiment shall be limited so that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.0 curie and the maximum strontium-90 inventory is not greater than 5 millicuries." The basis states that these limits assure that the dose to members of the public will not exceed the limits of 1 0 CFR Part 20. Please explain why this does not apply to workers or submit a change to the TS bases.*

The basis for TS 3.6(b) has been changed and will now read:

> The 1.0 curie limitation on iodine-131 through 135 assures that, in the event of a malfunction of a fueled experiment leading to total release of radioactive material including fission products, the dose to any individual will not exceed the limits specified in 10 CFR 20.

10. *TS 3.4, Ventilation states, "The reactor shall not been operated unless the facility ventilation system fan is operating, except for periods of time during which the dampers shall be closed. In the event of a release of airborne radioactivity in the reactor room above both routine reactor operation and normal background values, the ventilation system to the reactor room shall be secured via closure dampers automatically by a signal from the reactor deck air particulate monitor." Please explain what is meant by "except for periods of time during which the dampers shall be closed." Please describe analyses that have been completed with regard to releases/exposures under both open and closed positions of the dampers.*

TS 3.4 has changed and now reads:

3.4. <u>VENTILATION SYSTEM</u>

<u>Applicability</u>

This specification applies to the operation of the facility ventilation system.

<u>Objective</u>

The objective is to assure that the ventilation system is operable to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation.

<u>Specification</u>

The reactor shall not be operated unless the facility ventilation system is operating, except for periods of time not to exceed two (2) hours to permit repair, maintenance or testing of the ventilation system. In the event of a release of airborne radioactivity in the reactor room above both routine reactor operation and normal background values, the ventilation system to the reactor room shall be automatically secured via closure dampers by a signal from the reactor deck air particulate monitor.

<u>Basis</u>

During normal operation of the ventilation system, the concentration of argon-41 in unrestricted areas is below the limits allowed by 10 CFR 20. In the event of a fuel cladding rupture resulting in a substantial release of airborne particulate radioactivity, the ventilation system shall be shut down, thereby isolating the reactor room automatically by spring-loaded, positive sealing dampers. Therefore, operation of the reactor with the ventilation system shut down for short periods of time ensures the same degree of control of release of radioactive materials. Moreover, radiation monitors within the building independent of those in the ventilation system will give warning of high levels of radiation that might occur during operation with the ventilation system secured.

Analyses comparing doses to staff and members of the public in the open and closed damper positions are found in Chapter 13 of the proposed Safety Analysis Report. Specifically, the revised response to RAI #12 submitted to the NRC on January 17, 2012 discusses the bounding MHA for each damper position.

The following table will replace the existing Table 4-19 of the proposed SAR.

Table 4-19 AFRRI TRIGA RELAP5 Thermal Results Summary for Core Operating at 1.1 MW.

Parameter	Initial Core
Axial peaking factor – average element	1.316
Axial peaking factor – hot element	1.343
Hot element power factor	1.560
Inlet coolant temperature	48°C, 118°F
Coolant saturation temperature at core inlet	110.3°C, 230.5°F
Exit coolant temperature – average element	67.11°C, 152.8°F
Exit coolant temperature – hot element	82.51°C, 180.5°F
Average temperature in pool above core	60.2°C, 140.4°F
Coolant mass flow	13.60 kg/sec, 107,900 lb/hr
Average flow velocity	29.48 cm/sec, 0.967 ft/sec
Core average fuel temperature	247.1°C, 476.7°F
Peak fuel temperature in average fuel element	360.0°C, 679.9°F
Maximum wall temperature in hot element	149.2°C, 300.6°F
Peak fuel temperature in hot fuel element	440.7°C, 825.3°F
Average heat flux	27.87 W/cm^2, 88,362 BTU/hr-ft^2
Maximum heat flux in hot element	58.40 W/cm^2, 185,125 BTU/hr-ft^2
Minimum DNB ratio of 1.0	1.99 MW