UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 25, 2016

Kelly A. Jordan, Ph.D.
Director, University of Florida Training Reactor
106 UFTR Building
University of Florida
Gainesville FL 32611-6400

SUBJECT: UNIVERSITY OF FLORIDA – REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE RENEWAL FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR (TAC NO. ME1586)

Dear Dr. Jordan:

The U.S. Nuclear Regulatory Commission (NRC) is continuing our review of your request for renewal of Amended Facility Operating License No. R-56 for the University of Florida Training Reactor submitted by letter dated July 18, 2002 (available on the NRC's public Web site at www.nrc.gov under Agencywide Documents Access and Management System (ADAMS) Accession No. ML022130185), as supplemented.

During our review of your renewal request, questions have arisen which require additional information and clarification. We request that you provide responses to the enclosed request for additional information within 30 days from the date of this letter.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.30(b), "Oath or affirmation," you must execute your response in a signed original document under oath or affirmation. Your response must be submitted in accordance with 10 CFR 50.4, "Written communications." Information included in your response that is considered sensitive or proprietary, that you seek to have withheld from the public, must be marked in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Any information related to security should be submitted in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements." Following receipt of the additional information, we will continue our review of your renewal request.

If you have any questions, or need additional time to respond to this request, please contact me at (301) 415-3724, or by electronic mail at Duane.Hardesty@nrc.gov.

Sincerely,

/RA/

Duane A. Hardesty, Senior Project Manager
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-83
License No. R-56

Enclosure:
As stated

cc: See next page

cc:

Administrator
Department of Environmental Regulation
Power Plant of Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, FL 32301

State Planning and Development Clearinghouse
Office of Planning and Budgeting
Executive Office of the Governor
The Capitol Building
Tallahassee, FL 32301

Chief, Bureau of Radiation Control
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Brian Shea
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Test, Research and Training Reactor Newsletter
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202 Nuclear Science Building
Gainesville, FL 32611-8300

If you have any questions, or need additional time to respond to this request, please contact me at (301) 415-3724, or by electronic mail at Duane.Hardesty@nrc.gov.

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Duane A. Hardesty, Senior Project Manager
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Enclosure:
As stated

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ADAMS Accession No.: ML15336A005 * concurrence via email NRR-088

OFFICE	NRR/DPR/PRLB	NRR/DPR/PRLB*	NRR/DPR/PRLB	NRR/DPR/PRLB
NAME	DHardesty	NParker	AAdams (JAdams for)	DHardesty
DATE	7/21/2016	7/25/16	7/25/16	7/25/16

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OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ADDITIONAL INFORMATION

REGARDING LICENSE RENEWAL FOR THE

UNIVERSITY OF FLORIDA TRAINING REACTOR

LICENSE NO. R-56; DOCKET NO. 50-83

The U.S. Nuclear Regulatory Commission (NRC) is continuing our review of your application for renewal of Amended Facility Operating License No. R-56 for the University of Florida Training Reactor (UFTR) submitted by letter dated July 18, 2002, as supplemented. During our review of your renewal request, questions have arisen which require additional information and clarification. These questions are in reference to the most recent proposed UFTR safety analysis report (SAR), as provided on August 30, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13252A141), December 12, 2013 (ADAMS Accession No. ML13353A174), February 18, 2014 (ADAMS Accession No. ML14070A061), and April 9, 2014 (ADAMS Accession No. ML14112A317). We request that you provide responses to the enclosed request for additional information within 30 days from the date of this letter.

Title 10 of the *Code of Federal* Regulations (10 CFR), Part 20 "Standards for Protection Against Radiation," requires that dose to members of the public be limited. To support meeting the public dose limits, 10 CFR Part 20 also limits the release of radioactive materials (e.g., 10 CFR Part 20, Appendix B, Table 3). Further, the relicensing renewal application is required by 10 CFR 50.34, "Contents of applications; technical information," paragraph (b)(2) to include information that provides a description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. The regulations in 10 CFR 50.9, "Completeness and accuracy of information," requires that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

Technical Specifications (TSs) are fundamental criteria necessary to demonstrate facility safety and are required by 10 CFR 50.36, "Technical specifications," for each license authorizing operation of a production or utilization facility of a type described in 10 CFR 50.21, "Class 104 licenses; for medical therapy and research and development facilities." Additionally, 10 CFR 50.36(c) provides requirements to include Safety limits (SLs), limiting safety system settings (LSSS), Limiting conditions for operation (LCO), Surveillance requirements, design features (DFs), and administrative requirements. These TSs are derived from the analyses and evaluation included in the SAR and submitted pursuant to 10 CFR 50.34.

NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," provides guidance for the format and content for license renewal applications that is acceptable to the NRC staff for research and test reactors (RTR).

This guidance, and recent additional guidance under which this review is being conducted (SECY-08-0161) also includes American National Standards Institute/American Nuclear Society-15.1-2007, “The development of technical specifications for research reactors.” The NRC staff takes the position that the statements in these documents provide acceptable guidance to licensees and, unless acceptable alternatives are justified by the licensee, should be utilized whenever appropriate.

1. UFTR SAR, Chapter 4, “Reactor Description,” describes the general purpose Monte Carlo N-Particle code (MCNP6) model used to determine the SLs for the UFTR. The NRC staff reviewed the analysis, and is not clear regarding the use of any additional uncertainties, peaking factors, or power shapes, to represent the power gradients which result in a conservative peak fuel temperature and minimum departure from nucleate boiling ratio (DNBR).

   NUREG-1537, Part 2, Section 4.6, “Thermal-Hydraulic Design,” states, in part, that the SLs should include conservative consideration of the effects of uncertainties. Provide the peaking factors, or power shapes, used in the neutronics analysis, as well as any uncertainties, or justify why no additional information is needed.

2. UFTR SAR Chapter 4.5, “Nuclear Design,” describes the analysis of four fuel bundle cores, (22-BOL, 22-EOL, 24-BOL, 24-EOL). The UFTR analysis states that core 22-BOL had the highest fuel bundle power and the 22-BOL bundle configuration value is used as the limiting bundle configuration for the DNBR evaluation. However, the NRC staff review noted that the MCNP6 model used an average fuel depletion method, not a spatially distinct fuel depletion method, and is not clear if this depletion method will result in a conservative DNBR evaluation.

   NUREG-1537, Part 2, Section 4.6, “Thermal-Hydraulic Design,” states, in part, that the SLs should include conservative consideration of the effects of uncertainties. Provide a description of the effect of the fuel depletion on the DNBR analysis in order to demonstrate that the analysis is conservative, provide a justification that demonstrates that the assumptions used as inputs to the DNBR analysis are conservative, or justify why no additional information is needed.

3. UFTR SAR Section 4.5.5, “Comparison of Calculated and Measured Core Parameters,” describes calculated and measured excess reactivity and control blade reactivity worths. However, the NRC staff is not clear as to the source of the calculated excess reactivity and control blade reactivity worths.

   NUREG-1537, Part 2, Section 4.5.2, “Reactor Core Physics Parameters,” states, in part, that the calculational assumptions and methods shall be justifiable. Provide a description of the calculational methods used, and confirm whether these calculations use the same MCNP6 model used to describe the Limiting Core Configuration (LCC) in UFTR SAR Chapter 4 and the safety analyses in UFTR SAR Chapter 13, or justify why no additional information is needed.

4. The NRC staff review of the UFTR SAR did not find an analysis of an uncontrolled rod withdrawal (URW) event.

NUREG-1537, Part 2, Section 4.5.3, “Operating Limits,” provides guidance that the licensee should provide a transient analysis that involves an instrumentation failure that drives the most reactive control rod out in a continuous ramp mode in the most reactive core region. The URW analysis should include suitable assumptions (e.g., the sequence of events, control blade worths and initial positions, initial power, coolant temperature, blade trip time, reactor trip functionality, etc.). Provide an URW analysis, or justify why no additional information is needed.

5. UFTR SAR Section 4.6.3, “Thermal-Hydraulic Analysis Results,” states that the thermal-hydraulic (T-H) analysis was performed “under nominal full-power conditions for the core.” However, the NRC staff is not clear whether the T-H analysis was performed using the guidance in NUREG-1537, which provides that the T-H analysis should be done using the LSSS setpoints provided in the TSs.

   NUREG-1537, Part 2, Section 4.6, “Thermal-Hydraulic Design,” provides guidance that the licensee should describe the T-H conditions of the reactor core, including the minimum DNBR that supports the LSSS setpoints used in the TSs.

   Provide a T-H analysis using the UFTR LSSS setpoints (e.g., Power – 119 percent (119 kilowatts (kW)-thermal), Temperature - 99 degrees Fahrenheit (F), and 41 gallons per minute, or justify why no additional information is needed.

6. NUREG-1537, Part 1, Chapter 7, “Instrumentation and Control Systems,” provides guidance regarding reactor instrumentation for RTR. The UFTR SAR Chapter 7 provides information describing the reactor instrumentation used at UFTR. Provide a response that either implements a change to your application that addresses each issue identified, or proposes a suitable alternative.

   a. UFTR SAR Section 7.3.1, “Trip Circuits,” describes two types of reactor protection system (RPS) trips; a blade trip and a full trip. THE UFTR SAR states:

      Full-trip, which involves the insertion of the control blades into the core and the dumping of the primary water into the storage tank (this type of trip will dump primary water only if 2 or more control blades are not at bottom position);

      Blade-trip, which involves only the insertion of the control blades into the reactor core (without dumping of the primary water).

      UFTR SAR Section 7.3.1 also states, “[o]ne of five conditions must exist for the initiation of the Full-trip...” It is unclear whether the stated conditions cause a full trip or are required conditions (i.e., permissives) for a full trip. Similarly, UFTR SAR Section 7.3.1 states, “The conditions that must exist for the initiation of a Blade-trip include...” but is unclear if these conditions cause a blade trip or are required conditions for a blade trip. It is also unclear what other conditions may, or may not, need to exist for a blade trip to take place (e.g., interlocks, permissives, etc.).

      Revise the UFTR SAR to more fully explain the conditions that must exist for the initiation of full-trips and blade-trips, identify which trips are used in the performance of your accident/safety analysis, including an analysis of consequences of full trip, or justify why no additional information is needed.

b. NUREG-1537, Part 1, Section 7.4, “Reactor Protection System,” states, “[t]he RPS is designed to detect the need to place the reactor in a subcritical, safe shutdown condition (scram) when any of the monitored parameters exceeds the limit as determined in the SAR. Upon detecting the need, the RPS should promptly and automatically place the reactor in a subcritical, safe-shutdown condition (scram) and maintain it there.” The trip list in UFTR SAR 7.3.1 lists 18 automatic trips and the manual trip.

   The proposed UFTR TS, Table 3.2.2-1, “Specifications for Reactor System Trips,” lists only 4 automatic trips and the manual trip. The NRC staff review of the UFTR trips finds that the proposed list is inconsistent with the actual configuration of the UFTR reactor, the previous UFTR SAR, and the current UFTR TSs. Justify the reduction in the number of trips in your TS or any alteration of the associated setpoints, by: (i) identifying the trips excluded from the TSs, (ii) explain whether that trip will still be utilized in the operation of UFTR if the requested change is approved, and (iii) provide justification and/or analysis for the change to the UFTR TS, or removal of, this trip from the UFTR TS. Otherwise, revise TS Table 3.2.2-1 to fully incorporate all RPS trips and their previously approved setpoints. Alternately, justify why no additional information is needed.

c. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, “Format and Content of Technical Specifications for Non-Power Reactors,” Section 3.2(5) “Interlocks,” states that “interlocks that inhibit or prevent control rod withdrawal or reactor startup should be specified by a table (see Table 14.2 as an example). Interlocks should be specific to the facility and should be based on the SAR." UFTR SAR Section 3.1, “Design Criteria,” states that “[t]he instrumentation and control systems provide a series of alarms, interlocks and reactor trips preventing the occurrence of operating situations that are outside the bounds of the normal operating procedures." UFTR SAR Section 7.1, “Design of Instrumentation and Control Systems,” states that the reactor coolant system includes “one interlock system…” and the RPS includes “the Control-Blade Withdrawal Inhibit System..." UFTR SAR Sections 7.1.3.1.1, 7.1.3.1.2, 7.2.2, and 9.1.2 describe the importance of interlocks.

   The NRC staff review of the UFTR interlocks finds that the proposed list is consistent with the actual configuration of the UFTR reactor, the previous UFTR SAR, and list of interlocks in your proposed TS is comprehensive with respect to the actual interlocks used in the facility. However, the proposed TS lists only the Reactor Cell Evacuation Alarm Interlock (3.4) and that interlock is not discussed in the UFTR SAR. If the intent of your application is to reduce the number of interlocks in your TS or alter their conditions, then provide a response that: (i) identifies the interlocks affected, (ii) explains whether that interlock will still be utilized in the operation of UFTR if the requested change is approved, and (iii) provide justification and/or analysis for the change to the UFTR TS, or removal of, any interlocks from the UFTR TS. Otherwise, revise the proposed UFTR TS to fully incorporate all interlocks implemented in your approved TS and their associated conditions into your proposed TS. Alternately, justify why no additional information is needed.

7. NUREG-1537, Part 1, Chapter 13, “Accident Analyses,” provides guidance regarding accident analysis for RTR. The UFTR SAR Chapter 13, “Accident Analyses,” provides information describing the accident analyses for UFTR. Provide a response that implements a change to your application that addresses each issue identified or proposes a suitable alternative.

   a. The UFTR SAR, Section 13.2.1 “Maximum Hypothetical Accident (MHA) and Fuel Handling Accident (FHA),” provides estimates of occupational and public doses from fuel failure related accidents, and appears to follow the methodology in NUREG/CR-2079, “Analysis of Credible Accidents for Argonaut Reactors.” The NRC staff review identified the following issues that do not appear consistent with the guidance in NUREG/CR-2079. Provide an explanation or justify why no additional information is needed for the following:

      i. UFTR SAR Section 13.2.1.1, “Initiating Events and Scenarios,” describes the MHA analysis which assumes that the hottest fuel bundle has a power of 5.3 kW. However, the LCC neutronics analysis states that the highest power bundle is 5.45 kW (element/bundle 2-3 in the 22 bundle core in Table 4-8). Provide an explanation for the use of different power levels or justify why no additional information is needed.

      ii. UFTR SAR, Section 13.2.1.1 states that, for the FHA, it is postulated that the fuel bundle will split into two pieces exposing the fuel surface area equivalent to a guillotine break of all 14 fuel plates. The calculated fractional release of fission gases is stated to be 4.57E-3 percent of the total gaseous activity in the source term. However, the NRC staff’s confirmatory calculations using the UFTR assumptions indicate that this value is 4.60E-2 percent. (Note: The NRC staff analysis indicates that the estimate of the cross-sectional area of the broken fuel plates in the UFTR analysis is significantly smaller than the NRC staff’s estimate of that same parameter). Provide an explanation for the order of magnitude difference in gaseous activity levels or justify why no additional information is needed.

      iii. SAR Section 13.2.1.2.4, “Public Exposure,” states that the [EPA approved] COMPLY code was used to estimate the public dose from the MHA. The NRC staff has not generally accepted the use of COMPLY for the FHA and MHA analysis. The method in Regulatory Guide 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” is approved for the assessment of dose from accidents as is use of the code HOTSPOT. Provide a dose estimate using an NRC accepted method.

      iv. UFTR SAR Table 13-1 “Calculated Radionuclide Inventories (Ci) Three Days after Shutdown,” provides a list of radionuclide inventories (halogens and noble gases) with a 3-day decay. Section 13.2.1.2.1, “Radionuclide Inventories,” states that the ORIGEN-S code was used “under the assumptions stated in Section 13.2.5.1.” However, the NRC staff cannot find Section 13.2.5.1. Provide the reference location for these assumptions.

      v. UFTR SAR Table 13-1 radionuclide list includes I-132, and the quantity is listed as greater than that for I-131. The NRC staff does not understand how the quantity of

I-132 can be greater than the quantity of I-131 given their relative half-lives. Provide the un-decayed inventory of radionuclides for this accident and explain how the decay of this inventory was accomplished.

vi. UFTR SAR Table 13-3, "Summary of Occupational Radiological Exposure for the MHA," provides a dose of 0.136 rem for a 5-minute exposure. The NRC staff's confirmatory calculation of the occupational dose from a 5-minute exposure to I-131 in the reactor cell with a volume of 50,000 cubic feet indicates a dose of 0.485 rem given the dose conversion factor in SAR Table 13-2 (8.89 E-9 Sieverts per Becquerel) (Sv/Bq) and a released quantity of 0.208 Ci as provided in Table 13-1 of the UFTR SAR. Provide the details of the UFTR MHA calculations in order for the NRC staff to better understand the differences between the calculations.

vii. UFTR SAR Section 13.2.1.2.3, "Occupational Exposure," provides the MHA dose assumption of a free air volume for the reactor cell of 50,000 cubic feet. The proposed UFTR TS 5.1, "Reactor Cell," item (d.) indicates that the reactor cell dimensions of 30 feet by 60 feet by 29 feet (or a gross volume of 52,200 cubic feet). Based on these dimensions, the free volume for the MHA analysis is about 96% of the reactor cell total volume. This fraction is large and does not seem to account for reduction of volume from physical structures in Rx cell. Provide a description or calculation for how the reactor cell free volume was determined.

b. UFTR SAR Section 5.2, "Primary Coolant System," describes a DF of the Argonaut reactor to allow the water in the fuel boxes to drain into the coolant storage tank or drain into the equipment storage pit, similar to a loss of coolant accident. Furthermore, SAR Section 13.2.3, "Loss-of-Coolant Accident (LOCA) and Loss of Flow," provides a reference to a study [Wagner (Ref. 13.4)] that indicated that the hottest fuel bundle in a 625 kW highly-enriched uranium Argonaut core had an average power per plate of 4 kW (56 kW per assembly).

i. The NRC staff would like a copy or reference of the cited analysis for further review by the staff to support this Application. Provide a copy or reference for NRC staff review.

ii. The NRC staff is not clear how quickly the UFTR coolant water can be drained form the fuel boxes and the resulting maximum fuel temperature. Provide the drain time and maximum fuel temperature.

iii. The NRC staff is not clear which UFTR accident analyses include the possibility and consequences of the dump valve opening or rupture disk breaking in the possible accident sequence of events. Provide a list of the accidents which include a coolant dump valve opening or rupture disk breaking.

c. The UFTR SAR 13.2.4.2, "Analysis and Determination of Consequences," for an experimental malfunction states that the maximum diameter-to-thickness (d/t) ratio of an aluminum 6061 container of 2.30 is sufficient to contain the pressure spike from an explosion within the container. Furthermore, UFTR SAR Section 13.2.4.2 states that the pressure spike from the detonation of 25 milligrams equivalent Trinitrotoluene is 11.14 kilo bars (or 161,530 pounds per square inch (psi)). NRC staff confirmatory calculations reveals yield stress for aluminum 6061 is only 40,000 psi.

i. Provide confirmation that the yield strength of aluminum 6061 used in the UFTR SAR calculation was 40,000 psi.

ii. Provide a revised analysis to demonstrate that the requirements placed upon experiments having explosive potential are acceptability controlled

d. The UFTR SAR 13.2.4.2 states that "the Technical Specifications limit the quantity and type of fissile material." However, the proposed UFTR TS 3.8.3, "Experiment Failure and Malfunction," only restates the UFTR SAR and does not provide a gram limit or radioisotope limit (e.g., iodine isotopes) to establish the maximum inventory for consideration of this potential accident.

i. Explain how UFTR will apply a mass limit in applicable experiment approvals.

ii. An accompanying license condition is needed to permit the possession of fissionable material for irradiation. Provide a proposed license condition for the possession of fissionable material for irradiation including a mass (gram) limit and a description of the material form, or, if the UFTR does not intent to irradiate fissionable material, revise or delete the proposed UFTR TS, as appropriate

e. UFTR SAR Section 13.2.2, "Insertion of Excess Reactivity," provides the UFTR analysis of the insertion of excess reactivity performed using PARET-ANL [Program for the Analysis of REactor Transients-Argonne National Laboratory]. The analysis cites a 50% uncertainty in calculated fuel temperature due to the selection of input parameters and 50% uncertainty in calculated fuel temperature based upon experience using the Tong DNBR correlation. Provide documentation of the Tong correlation and explain the basis for the applicability of the Tong correlation to UFTR DNBR analysis including the basis for the uncertainty analysis.