April 6, 2005

Dr. William G. Vernetson
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SUBJECT: UNIVERSITY OF FLORIDA—REQUEST FOR ADDITIONAL INFORMATION RE: LICENSE RENEWAL FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR (TAC NO. MB 5804)

Dear Dr. Vernetson:

We are continuing our review of your request for renewal of Amended Facility License No. R-130 for the University of Florida Training Reactor which you submitted on July 29, 2002. During our review of your request, questions have arisen for which we require additional information and clarification. Because of the recent Department of Energy decision to move forward on conversion of the University of Florida Training Reactor from high enriched to low enriched uranium fuel, the staff will discuss with you the timing of your responses to these questions. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation. Following receipt of the additional information, we will continue our evaluation of your amendment request.

If you have any questions regarding this review, please contact me at (301) 415-1127.

Sincerely,

/RA/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-83

Enclosure: As stated

cc w/enclosure: See next page

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REQUEST FOR ADDITIONAL INFORMATION
UNIVERSITY OF FLORIDA TRAINING REACTOR
DOCKET NO. 50-83

1 THE FACILITY

1-1 The Introduction to the SAR does not discuss shared facilities and equipment. Does this mean that the University of Florida Training Reactor (UFTR) does not have shared facilities and equipment as discussed in Section 1.4 of NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors?" Please confirm that there are no shared facilities and equipment or describe the shared facilities and equipment.

2 SITE CHARACTERISTICS

2-1 Section 2.1.1.2, Boundary and Zone Area Maps, page 2-1, and Section 2.1.1.3, Boundaries for Establishing Effluent Release Limits, page 2-2. Reference is made to definitions from 10 CFR Part 100. This regulation is not applicable to research reactors. How does the facility and site meet the definitions in 10 CFR Part 20 (e.g., restricted area and site boundary) and 10 CFR Part 73 (e.g., protected area)? The area of the facility and site proposed under the reactor license should be clearly described.

2-2 Is there any railroad station or line located near the UFTR site so that a derailment accident could affect the reactor building?

2-3 Is there any military installation (e.g., aircraft flight path) near the UFTR site so that military activities in the area could affect the reactor building?

2-4 Local meteorological measurements for use in evaluating accidental effluent releases from the UFTR do not appear to be available. Explain where this information will be obtained if needed.

3 DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3-1 Section 3.1, Design Criteria, page 3-1. The UFTR reactor building is divided into two distinct areas. The reactor area is 30 ft. by 60 ft. by 29 ft. high and is located at the north end of the building. The remaining area of the building is used for research and teaching laboratories, faculty and graduate student offices, and work areas. The reactor area is on a one foot thick slab resting on undisturbed or compacted earth. The thickness of this slab is increased to 18 inches under the reactor. The walls of the reactor area are constructed of one foot thick monolithic reinforced concrete resting on mat footings. The 3-inches thick roof of the reactor area is built-up of a precast roof tile supported by steel-bar joists spaced 2 ft. on centers. Please discuss the following:

a. What building Code was used while constructing the reactor building? Did the building design include any seismic and wind loads?

b. What administrative controls exist regarding the use of the overhead crane during reactor operations?

c. Describe the design of the brick flue and/or the reactor stack that carries the exhaust air above the top of the reactor building. How tall is the stack?

4 REACTOR DESCRIPTION

4-1 Section 4.1.1, General Reactor System Design. Demineralized water is used as the primary coolant. Has the UFTR experienced any water chemistry excursions which have resulted in material degradation of the fuel or other core components?

4-2 Section 4.1.2, Design and Performance Characteristics. The SAR addresses nuclear and thermal design characteristics. Discuss if other issues may limit fuel integrity, including water chemistry issues, physical stresses from mechanical or hydraulic forces, fuel burnup, radiation damage to fuel, and fission product retention.

4-3 Section 4.1.2, Design and Performance Characteristics. The transmittal letter (Dr. W. G. Vernetson to NRC, dated July 25, 2002), states that the reason for the change on control blade drop time from 1.0 seconds to 1.5 seconds was to prevent unnecessary unstacking and entry into the core to make repairs to assure meeting the 1.0 second limit.

   a. Why were the control blades not able to meet the 1.0 second limit?

   b. What is the quantitative impact on the reactor safety margins due to this change?

   c. Section 4.2.2.1, Table 4-1: Table 4-1states that the worth of the three safety shim arms are as follows: #1, 122(sic)% Îk/k; #2, 1.35% Îk/k; and #3, 1.83% Îk/k. However in the technical specifications (TSs) (Section 5.5) the blade worths are stated to be between about 1.3 and 2.3% Îk/k. Please make the SAR and TSs consistent.

4-4 Section 4.2.1, Fuel System Design. The TSs (Appendix 14.1, Section 5.3.1) permit several fuel matrix fabrication options. Which of these processes was used for the UFTR fuel? Do the different fuel fabrication options present any unique issues or limitations regarding the use of the fuel for the UFTR?

4-5 Sections 4.2.1, Fuel System Design and 4.2.2.1, Control Rods. What is the lifetime of the fuel assembly and control rods at UFTR? How does the control rod worth change with time?

4-6 Section 4.2.2.1, Control Rods. This section states the usual detailed information is not provided since the ‘control rod systems are previously operated systems’, however no information on this system is discussed in Section 16.1, Prior Use of Reactor Components. Please provide a reference which discusses the operating history of the control rods.

4-7 Section 4.2.3, Neutron Moderator and Reflector. The SAR alludes to aging effects and the hope to load new reactor grade graphite into the UFTR core. Discuss the aging effects observed, and any operating changes or restrictions which have been needed in response to the aging effects. What is the remaining life of the existing graphite?

4-8 Section 4.2.4, Neutron Startup Source. What special handling restrictions are in place applicable to the neutron startup sources? Where are the sources stored when not in use?

4-9 Section 4.2.5, Core Support Structure. The SAR states that the core support structure materials will continue to be adequate given the current operating conditions. What is the estimated remaining life for the core support structure?

4-10 Section 4.3, Reactor Tank. How would leakage from the aluminum reactor tanks be detected? What is the minium leakage rate that can be detected, and what is the maximum time duration that this leakage can occur before detection? What would the impact on public health and safety be from tank leakage?

4-11 Section 4.3, Reactor Tank. This section discusses a fuel safety limit of 200EF. It appears that the purpose of the limit is to protect the structural integrity of the reactor tank and not the integrity of the fuel cladding. Please clarify.

4-12 Section 4.4 Biological Shielding. Is the addition of Poly-B-Pb in the shielding going to happen? If it is, its addition should be discussed in more detail.

4-13 Section 4.4, Biological Shielding, Page 4-10 states the actual exposure at the north and south faces is approximately 3mR/hr, but Table 4-4 shows 2 and 0.8 mR/hr at 1 foot. Is the 3mR/hr on contact and the table value measured at a distance from the reactor?

4-14 Section 4.4, Biological Shielding. Is ground water and soil activation possible? If so, please discuss.

4-15 Section 4.4. Is 2.5 mR/hr the normal and expected dose rate at the three area monitors or is that an unusual level?

4-16 Section 4.4, Biological Shielding. Could radiation damage and heating of the shielding during the 20-year renewal period along with potential radiation-induced degradation and activation of the material impact the integrity of the shielding? Is there any potential for streaming of radiation along the shielding? Discuss shielding of experimental facilities, if any.

4-17 Section 4.4, Biological Shielding. Please address the shielding of spent fuel.

4-18 Section 4.5.1, Normal Operating Conditions. What administrative (operating procedures and TS limits) and physical constraints (interlocks and trips) exist to prevent inadvertent addition of positive reactivity?

4-19 Section 4.5.1.1, Flux Distribution. There seems to be ~10^{12} difference in the fluxes quoted in Table 4-6 and Figures 4-23 and 4-24. Please address.

4-20 Section 4.5.1.1, Flux Distribution. Please state the neutron energy divisions for the 4 group calculations.

4-21 Section 4.5.1.2, Control Blade Worth, Shutdown Marging and Excess Reactivity. The values for the control rod worths are not consistent between Tables 4-1 and 4-9. Please explain the difference.

4-22 Section 4.5.2, Reactor Core Physics Parameters. For what type of core were these coefficients determined, fresh, end-of-life or, some other condition?

4-23 Section 4.5.2, Reactor Core Physics Parameters. Do the parameters change significantly with burn up?

4-24 Section 4.5.2, Reactor Core Physics Parameters. Please describe the axial and radial flux densities.

4-25 Section 4.5.3, Operating Limits. This section of the SAR should contain the discussion and calculations to support the safety limits, limiting safety systems settings, limiting conditions of operation and surveillance requirements related to operation. (See Pages 4-11 and 4-12 of NUREG-1537, Part 1, for a list of detailed information that is expected to be included in this section covering Operating Limits.) Please provide this information or provide references where this information can be found.

5 REACTOR COOLANT SYSTEMS

5-1 Section 5.2, Primary Coolant System, page 5-1. The UFTR is designed for forced flow cooling while in operation. There is a heat exchanger (HX) in the forced flow loop of the primary coolant system to maintain the primary coolant temperature. The primary coolant cleanup system loop is also part of the primary coolant system. The cleanup pump in this loop is interlocked with the primary pump to prevent its operation during normal operation of the system. The function of this cleanup system is to maintain the chemistry quality and conductivity of the primary coolant. The heat exchanger is cooled by an open loop secondary cooling system which uses deep well water to cool the primary coolant and discharges into the city storm sewer system. Please discuss the following:

a. Provide sketches or layout drawings to depict the location of the primary coolant system and associated systems (i.e., secondary coolant, primary coolant cleanup and primary coolant makeup water systems) with respect to the building structures of the reactor building. Specifically, identify the portions of these systems including associated major components that are located inside and outside of the reactor building confinement.

b. If there were a reactor coolant piping/component failure outside of the reactor core, describe where the primary water would be collected in the building? How would this be detected, measured, and alarmed?

c. It is stated that the graphite rupture disk is set to burst at 7 psia, which is 2 psi above normal operating pressure. What is the normal operating pressure of the primary coolant? Note that page 3-5 states that the system operates at ambient pressure and a low temperature below 155EF.

5-2 Section 5.3, Secondary Coolant System, page 5-3. The pressure of the secondary water is maintained higher than the primary system to prevent contamination of secondary coolant. What are the normal operating pressures of the primary and secondary coolant systems? How are these pressures monitored? If a leak were to develop in the primary/secondary boundary, how would this be detected? Since the secondary water is tested weekly for radiological contamination (Appendix 14.1, Section

4.3, Item (4)), is there any way to identify such contamination that may be occurring in-between the weekly testing period (e.g., on continuous basis)?

5-3 Section 5.3, Secondary Coolant System. Normal secondary flow is 200 gpm. At 140 gpm a low flow warning signal is sent to the control room and at 60 gpm a reactor trip is initiated if the reactor is at or above 1 kW after a 10-sec warning. When city water is used, a less than 8 gpm flow in the input line will initiate a reactor trip for power levels above 1 kW. What is the normal flow rate of city water when it is used as the backup to well water?

5-4 Section 5.6, Nitrogen-16 Control System, page 5-4. Is there any shielding around the piping from the reactor to the coolant storage tank and the coolant storage tank area? If not, explain any administrative procedures to restrict entry into this area during reactor operation and to allow time for N-16 decay?

6 ENGINEERED SAFETY FEATURES

6-1 Section 6 states that because the reactor is self-limiting, there is no additional requirement for engineered safety features. Confinement is achieved through keeping a negative pressure on the control and reactor rooms. Dilution is used to keep both postulated accident and operational radioactivity releases within specifications. TS 3.4, Reactor Vent System, states that this system shall be operational during operation of the reactor. What is the purpose of this system? Since the backup to control blade insertion is allowing the water (the moderator) to run out of the fuel boxes making the reactor sub-critical, is the vent system heat removal capability required to prevent cladding damage to the fuel? If this system was credited as a heat removal mechanism for accident mitigation it should be considered an ESF. Please explain if this is the case, it is not clear in the SAR.

7 INSTRUMENTATION & CONTROL

7-1 Section 7.1, fourth paragraph, states that "...system instruments are hardwired analog instrument type with the exception of the temperature monitor and record system which is a digital system instrument type." Section 7.2.1 indicates that the control blade position indicators and master console clock are now also digital instruments/displays. Please clarify.

7-2 Sections 7.2.3.4.2 , 7.3.2, and 5.3 list a low secondary coolant system flow trip of 60 gpm when the deep well pump is the coolant water source and the reactor is operating above 1 kW. The low flow trip first illuminates a red scram warning light on the reactor control console and then trips the reactor after approximately a 10 second delay. What is the basis of the 10 second delay?

7-3 Sections 7.2.3.4.2, 7.3.2, and 5.3 list a low secondary coolant system flow trip of 8 gpm when the city water supply is the coolant water source and the reactor is operating above 1 kW. Please explain the difference in the low flow setpoints when using the two different sources of secondary cooling water. Also, the last sentence in paragraph four of Section 5.3 is not clear: Is there a time delay associated with the city water 8 gpm low flow trip? If so, what is the basis of this time delay?

7-4 Figure 7-10 shows the temperature monitor and recorder system. Are any of the reactor scram or alarm functions dependent on software? If so, what validation and verification process was used on the software. It appears that a CPU is used in the monitor temperature virtual instrument. This system appears to be digital based. If so, what validation and verification process was used on the software. Does the reactor operator make operational decisions based on the output of the monitor temperature virtual instrument? Does the instrument store temperature data that is used to show compliance with license requirements?

7-5 Section 7.2.3.4.2 states that there is a key operated switch inside the reactor control console rear door to switch secondary coolant system low flow scram modes from the well water source mode to the city water source mode. This switchover is apparently a manual action; is it covered in the facility operating procedures? Is there any indication on the front of the reactor control console that informs the operator which secondary coolant source (and low flow reactor trip setpoint) is in effect? If not, please describe the administrative controls that make this information available.

7-6 Sections 7.2.3.4.1 and 5.2 describe a coolant flow switch in the return line of the primary coolant system to the primary coolant storage tank which will scram the reactor in case of loss of return flow. The switch serves as a backup to the primary coolant low flow reactor trip instrument in the fill line. What is the setpoint for the return line flow instrument? Is the surveillance frequency the same as for the flow instrument in the fill line?

7-7 Many older analog components have become obsolete and are no longer available; ‘equivalent' replacement components are not always true replacements. How are replacement electronics components to repair the analog instruments and electronic circuit boards qualified? Following repairs, how are acceptance tests selected to certify that circuit boards and logic modules, and the equipment in which they are installed, are functionally operable?

8 ELECTRICAL POWER

8-1 Provide single-line drawing(s) depicting supply feed(s) and distribution of normal and emergency sources of AC and DC electrical power systems (for example: is the voltage supplied at the desired service levels (230V and 115V) or is a step-down transformer used? How is power distributed inside the facility? Is there a main distribution center (motor control center) or are there multiple/individual distribution panels with individual/separate feeds from outside sources?).

8-2 Section 8.2. The fail safe behavior of the reactor protection system and control blades was described. Upon loss of power, does the primary coolant system dump value also drain the primary system?

8-3 Describe the design features (e.g., design and location of electrical wiring) provided to ensure that electrical power circuits are sufficiently isolated to avoid electromagnetic interference with safety-related instrumentation and control systems.

8-4 Describe any needs for electrical power that may be required for placing/maintaining experimental equipment in a safe condition.

8-5 Section 8.3, Emergency Electrical System. This section states that no credit is taken for the back-up electrical diesel generator for safety analysis considerations. In the event of an extended loss of the normal AC power source, will operation of the emergency power source (Diesel Generator) be relied on to ensure the availability/operation of systems which provide for personnel safety, habitability of the reactor facility, reactor status instruments, and radiation monitoring systems?

9 AUXILIARY SYSTEMS

9-1 Section 9.2.1, New Fuel Storage. This section states that loading and unloading of fuel into (and out of) the reactor core will only be performed by 'qualified reactor operators and staff.' Define what 'staff' members are permitted to perform these functions (as defined in Appendix 14.1 TSs). If other positions are included as 'staff', what are the qualification requirements for these individuals.

9-2 Section 9.2.3, Bridge Crane. The bridge crane is described as a 3-ton crane. Are there any restrictions or safety factors for the crane which limits the actual load which can be safely handled? Briefly describe what preventive maintenance or inspections are performed on the crane to ensure continued safe operation. Are there any restrictions with regard to handling heavy loads over the core? What is the weight of the fuel transfer cask?

9-3 The criticality accident requirements of 10 CFR 70.24 are applicable to the UFTR. Please discuss how this regulation is met.

9-4 Section 9.2.2, Spent Fuel Storage. What is the temperature as a function of storage time for the dry-stored spent fuel in the storage pits?

9-5 Section 9.6.3, Equipment and Floor Drainage System. This section states that the reactor building floor drainage system is designed so that liquid effluents go directly to the hold-up tank. But the section also states that there are no drains leading directly to the hold-up tank. Please clarify.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10-1 Confirm that loss of AC power is considered during the experiment approval process.

10-2 Provide a current copy of UFTR SOP-A.5 (Experiments).

10-3 Section 10.2.6, Automatic Transfer System (Rabbit). Provide a more detailed description of the design and operation of the automatic pneumatic sample transfer (Rabbit) system and the administrative controls governing its use. Specific topics to be addressed include the size (diameter) of tube and rabbit, potential consequences of a stuck/immovable rabbit assembly and design features and/or administrative controls provided to preclude or mitigate this occurrence.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11-1 Please provide calculations to show that doses to the reactor staff and members of the public from the production of normal gaseous effluents from reactor operations is acceptable. The calculations should be based on continuous reactor operation (unless you want to limit reactor operation by license condition) and should consider both argon-41 and nitrogen-16. Doses should be determined for staff members and the maximum exposed member of the public, at the closest residence to the reactor and at any other points of special interest (e.g., dormitories), if applicable.

11-2 Section 11.1.2.3.2, Ventilation. This section refers to a 200 to 1 stack dilution factor. Please discuss the basis of this factor.

11-3 Section 11.1.2.4, Health Physics Program. This section states that the Radiation Control Officer supervises the actions of the UFTR RSR Subcommittee. Please explain why the Radiation Control Officer, an ex-officio member of the subcommittee, has supervisory responsibility over the RSR Subcommittee and how that affects the independence of the subcommittee.

11-4 In this chapter there is no mention of the special nuclear material and byproduct material limits in your current license. Please confirm that you want to maintain similar limits in your renewed license.

12 CONDUCT OF OPERATIONS

12-1 Section 12.1, Organization. The organizational chart (Figure 12.2) contains many lines with arrowheads and a diamond-shaped "or" box that are not completely clear. Please show reporting lines by solid lines and communication lines by dotted lines. Also show reporting responsibilities by arrows.

12-2 Section 12.1.3, Staffing, and TS section 6.1.3, Staffing. 10 CFR 50.54(m)(1) requires that an SRO shall be present at the facility for three specified activities. For example, an SRO shall be present at the facility during recovery from an unplanned or unscheduled shutdown. The SAR and TSs both use the words "direction" rather than presence. Further, the SAR uses the wording, "documented verbal concurrence from a Senior Reactor Operator is sufficient." The use of "sufficient" rather than "required" when discussing the verbal concurrence of the SRO seems to imply that the SRO may give concurrence for recovery without being present at the facility. The intent is to have the SRO present and to document their concurrence with the restart. Please update the wording or explain why it meets the requirements of 10 CFR 50.54(m)(1).

12-3 Section 12.1.4, Selection and Training of Personnel. The selection of personnel should meet the guidance in ANSI/ANS 15.4-1988. This is quoted in the TS but the SAR cites ANSI/ANS 15.4-1977. Please correct.

12-4 Section 12.1.5, Radiation Safety. Does the radiation safety staff have the ability to raise safety issues with the review and audit committee or university upper management and do they have the clear responsibility and ability to interdict or terminate licensed activities that they believe are unsafe? If not, how does the radiation safety staff deal with activities they believe are unsafe?

12-5 Section 12.1.5.1, Reactor Safety Review Subcommittee, and TS 6.2 Review and Audit. A quorum is defined as at least three members. But the membership is defined as at least five members. If there are more than six members a quorum of three would be less than half. The quorum should be at least three and at least half, also with the operating staff not constituting a majority (to meet ANSI/ANS 15.1). Also the Radiation Control Officer is referred to as both a member and an ex-officio member. Please address.

12-6 Section 12.1.5.1, Reactor Safety Review Subcommittee, and TS 6.2 Review and Audit. The SAR and TS should specify that all reports and minutes of findings and recommendations of the subcommittee should be submitted to Level I management; and should also specify which Level I manager(s). Please address.

12-7 Please address how you meet the requirements of 10 CFR 50.54(i) or (l).

13 ACCIDENT ANALYSIS

13-1 Ad hoc criteria were used in the SAR to extrapolate the BORAX I and II results to the UFTR core. Is there any transient calculation that shows the excursion energy for the UFTR in a nuclear excursion? What is the predicted maximum fuel temperature in the most limiting nuclear excursion? Is there any requirement on the coolant void and temperature reactivity feedback such that the maximum excursion energy is limited to 32 MW-sec?

13-2 The staff believes that the design basis accident (or maximum hypothetical accident [MHA]) chosen for the reactor in the SAR is extremely unrealistic and conservative. The purpose of the MHA is to conservatively, but realistically, bound the worse case radionuclide release that could occur. The staff has accepted a core crushing accident as the MHA for an Argonaut reactor and NUREG/CR-2079 has analyzed this accident for a generic Argonaut reactor. However, the NUREG/CR-2079 analysis is highly conservative and could be made more realistic by considering items such as, decay following reactor shutdown and isotope plateout. Also, as explained in NUREG-1537, the accident dose limits found acceptable to the NRC staff for reactors initially licensed before January 1, 1994, has been 5 rem whole body and 30 rem thyroid for occupational exposure and 500 mrem whole body and 3 rem thyroid for members of the public. Please reevaluate your MHA or provide justification as to why the MHA presented in the SAR is realistic.

13-4 In Section 13.3.5 the urban boundary was set to a distance of 0.5 miles. Instead, doses should be determined for staff members and the maximum exposed member of the public, at the closest residence to the reactor and at any other points of special interest (e.g., dormitories), if applicable.

13-5 Appendix 13-1. The same ratio was shown in Equation 13A-1 and Equation 13B-4 and it was used to adjust the BORAX non-melting excursion energy for the UFTR. What is the significance of the ratio? Was it an indication of the heat capacity of the fuel plate (per discussion on p. 13-A.1) or an indication of the heat transfer capability of the fuel plate (per discussion on p. 13-B.5)?

13-6 Appendix 13-B. In the last paragraph on p. 13-B.2 it was stated, "… the reactor could operate in the absence of protective actions at an equilibrium power level about 10 times higher than its normal maximum with little or no net steam production." Does this statement apply to the current normal power level of 100 kW or the original licensed power of 10 kW?

13-7 Appendix 13-B. On p.13-B.2 the heat removal capacity of 107 kWth was based on an assumed outside air temperature of 0°C. A more realistic outside temperature would significantly reduce the heat removal capability of the reactor coolant system. Please state whether this is an appropriate assumption.

13-8 Appendix 13-B. In Section 13B.2, what is the reference for correspondence between the excess reactivity of 0.6% Δk/k and the asymptotic period of 0.8 seconds?

13-9 Appendix 13-B. What is the source of Figure 13B-1? Has its applicability to UFTR been demonstrated?

13-10 Appendix 13-C. Are the constants a_1 and a_2 in Equations 13C-1 and 13D-1 defined in Table 13D-1?

13-11 Appendix 13-D. Decay Heat Effects. In Section 13D.2, what is the reference for the calculation of the unit thermal conductance between the fuel plate and the fuel box? What are the bases for the assumed 50% Al-Al contact. What are the bases for the assumed contact pressure and the thickness of the air wall?

13-12 Appendix 13-D. What are the bases for the assumed 50% air and 50% graphite in the wall separating the fuel box and the graphite?

13-13 Appendix 13-D. What is the temperature of the heat sink (graphite) and how is it justified?

14 TECHNICAL SPECIFICATIONS

14-1 Technical Specifications (TSs). Bases are given for many of the TSs as required by 10 CFR 50.36. Please ensure that the bases for the TSs can be traced back to an analysis in the SAR. It is not clear when some of your TSs are applicable. For example, is TS 3.2.1(4) required to be met at all times or is it a requirement to take the reactor critical? Please review all TSs and ensure that it is clear under what conditions the TS applied.

14-2 Definitions. Please review your definitions to verify that they are used in the TSs or documentation that supports the operation of the reactor. Consider if definitions that are not used in operation of the facility are needed.

14-3 Section 2.0, Safety Limits and LSSS. As noted in the guidelines contained in NUREG-1537, Part 1, Appendix 14.1, Section 2.1.3, "… For plate type fuel… the applicant should determine a fuel cladding temperature below which cladding damage (softening or blistering) can be precluded. The applicant should then establish a corresponding power level, reactor conditions, and uncertainties that limit cladding temperature below the damage limit."

In the introduction of Section 2.1 you have correctly described the purpose of SLs and identified the fuel cladding as the principal fission product barrier to be protected. The process variables chosen should be those that if exceeded will quickly threaten the integrity of the fuel clad. One of the reasons why if a safety limit is exceeded, the reactor must be shut down until approval for restart is given by the NRC is to ensure that the fuel clad was not damaged when the safety limit was exceeded. NRC has accepted an upper fuel temperature for aluminum-clad, aluminum matrix plate-type fuels of 530EC. Plate blistering, a possible forerunner of cladding failure, has been observed above this temperature. While fuel temperature would be the best process variable for the safety limit, the inability to measure this process variable leads to the need to use variables that can be measured and controlled. For reactors with forced convection flow, the staff has accepted controlling the process variables of reactor power, coolant temperature, coolant flow, and if credit was taken in the analysis, height of water above the core. Exceeding the limit on primary coolant resistivity does not lead to immediate fuel clad damage. The NRC staff accepts primary coolant resistivity being controlled as a limiting condition of operation. Please develop safety limits for reactor power, coolant temperature and coolant flow based on keeping fuel temperature limited to 530EC. Discuss the need for a safety limit on height of water above the fuel elements, and if justified, propose a safety limit. Justification of safety limits usually appears in Chapter 4 of the SAR and the accident analysis in Chapter 13 of the SAR usually forms the technical bases for the limiting safety system settings (LSSS) and the safety limits.

14-4 Safety Limits and LSSSs. For those LSSSs that protect safety limits, provide an analysis that shows that automatic protection at the LSSS limit will protect the safety limit considering process uncertainty, overall measurement uncertainty and the transient phenomena of the process instrumentation.

LSSS 2.2(6). This does not appear to be a process variable limit such as the flow rate. It appears to be an on-off condition that is better addressed as a LCO. Please justify this as an LSSS or move to the LCO section of the TSs.

LSSS 2.2(7). This does not appear to be a LSSS because it is not a limit on a process variable. This appears to be a LCO or design feature. Please justify this as a LSSS or move the requirement to a more appropriate section of the TSs.

LSSS 2.2(10). This does not appear to be a process variable limit such as the flow rate. It appears to be an on-off condition that is better addressed as a LCO. Please justify this as an LSSS or move to the LCO section of the TSs.

LSSS 2.2(11). This does not appear to be a LSSS because it is not a limit on a process variable. This appears to be a LCO equipment operability requirement. Please justify this as a LSSS or move the requirement to a more appropriate section of the TSs.

The full bases for the LSSSs should be presented in the appropriate sections of the SAR.

14-5 Section 3.1(2), Excess Reactivity. A statement is made in Section 13.1.1.1 of the SAR that the UFTR is not planned to contain more than about 1.2% Δk/k excess reactivity even when freshly loaded. Given that statement, please justify the need for a TS excess reactivity limit of 2.3% Δk/k.

14-6 Table 3-1 and 3-2. It is not clear if some of the safety system operability tests are testing the operability of the safety system feature of concern. How does loss of primary coolant flow show operability of the low inlet water flow? How does loss of primary coolant level show operability of the low water level in core safety system trip? How does the loss of shield tank water level show shield tank low water level?

14-7 Table 3-2. An operability test of the period and power channels is required following a shutdown in excess of 6 hours. What is the basis of the 6-hour time period. Does this apply if the reactor is secured? The table tests component or scram function. Do these tests confirm the scram function of the control rods or, as appropriate, the safety system trip function of the control rods and the dump valve?

14-8 TS 3.3(2), Reactor Coolant System. Please explain the purpose of the 6-hour reactor operation statement as related to primary coolant resistivity. Please explain why primary coolant pH is not controlled?

14-9 Table 3-4, Radiation Monitoring System Settings. The stack radiation monitor has a fixed alarm at 4000 cps. What hazard does a warning at this count rate represent? How are changes in the efficiency of the monitor with time or as components are replaced accounted for?

14-10 Section 3.6(4), Explosive Materials. The TS refers to "limited quantities" of explosive materials that may be irradiated. Please either propose and justify a quantity of explosives or discuss the basic restrictions that explosives must meet to be irradiated (e.g., irradiation container has ability to contain by a certain factor the energy released if the explosive is detonated).

14-11 Section 3.6(7), Fueled Experiments. The TS refers to "a limit should be established" on the inventory of fission products in fueled experiments. Please propose and justify an upper limit on the allowable fission product inventory.

14-12 Section 3.8 Fuel and Fuel Handling. LCO 3.8(3) and (4) prohibit reactor operation with failed fuel. Is the primary coolant surveillance described in 4.3(3) the only means of detecting fuel failure, or are there other indications used to provide a more rapid indication of failed fuel? If failed fuel were detected, how would the specific failed fuel assembly be identified?