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10 CFR 50.4, Written Communications
UFTR Operating License R-56, Docket 50-83

Subject: **UFTR Responses to Request for Additional Information (TAC NO. ME1586)**

By letter dated July 18, 2016, the NRC requested additional information regarding proposed UFTR license renewal Technical Specifications (ML15336A796). By letter dated July 25, 2016, the NRC requested additional information regarding the technical details provided for UFTR license renewal (ML15336A005). A total of 114 RAIs were requested with a scope that includes requests for new analyses and updated analyses. Both sets of RAIs requested that changes be made to the proposed license renewal Technical Specifications.

Since then the UFTR and NRC have been communicating on a frequent basis to clarify the RAIs and ensure the result of this process is a thorough and contemporary licensing basis that eliminates the need for further RAIs.

Attached are the UFTR responses to the formal RAIs as well as the revised Technical Specifications. The revised Technical Specifications incorporate the changes resulting from the formal RAIs as well as those from ongoing recent discussions with NRC staff. The revised SAR and ALARA plan will be provided by November 30, 2016.

This submittal has been reviewed and approved by UFTR management and by the Executive Committee of the Reactor Safety Review Subcommittee.

I declare under penalty of perjury that the foregoing and attached are true and correct to my knowledge.

Executed on October 31, 2016.

Brian Shea
Reactor Manager

cc: Dean – College of Engineering
Reactor Safety Review Subcommittee
Facility Director
Licensing Engineer
NRC Project Manager

1. *TSs are fundamental criteria necessary to demonstrate facility safety and are required by 10 CFR 50.36 for each license authorizing operation of a production or utilization facility of a type described in 10 CFR 50.21. TSs are derived from the analyses and evaluation included in the safety analysis report and submitted pursuant to 10 CFR 50.34. Due to the importance to overall facility safety of a uniform interpretation by both the licensee and regulator of terms and phrases used in TSs, definitions shall be included where necessary to ensure that the technical specification criteria necessary for compliance with regulatory requirements is uniformly understood by the licensee and regulator. The following questions pertain to Section 1, Introduction, Definitions and Surveillance Intervals of the proposed UFTR TSs. Provide a response that addresses each issue identified, propose a suitable alternative in your response, or explain why one is not needed.*

   (a) Section 1.0, Introduction, states "...requirements to which the licensee must adhere." Replace the term "licensee" with a specific reference to the UFTR licensee to which these TSs apply that identifies the person or organization holding the UFTR license or explain why this is not needed.

   After further discussion with staff this RAI was deemed unnecessary.

   (b) The term "channel" is used throughout the UFTR TSs, however, it is not defined. Provide a suitable definition for "channel" for the purposes of the UFTR TSs that is consistent with the ANSI/ANS-15.1-2007 definition of this term, justify an alternative definition, or explain why one is not needed.

   The definition has been added.

   (c) The UFTR TSs include a facility-specific definition for "CORE ALTERATION," that states "[s]uspension of core alterations shall not preclude completion of movement of a component to a safe position." Being that "CORE ALTERATION," is defined in the UFTR TS to be movement of any reactor fuel assemblies, etc. the statements in the definition appear contradictory. Further, the definition for Core Alteration references "component," which is not defined in the UFTR TS. Provide a revised definition for CORE ALTERATION that fully explains the intended meaning of the term as used in the UFTR TS and that does not use undefined terms.

   The definition has been revised.

   (d) The UFTR TS definition for "MOVABLE EXPERIMENT," states "where it is intended that all or part of the experiment may be moved into or adjoining the core or into and out of the core while the reactor is critical." As 'critical' has a very narrow definition, and would not appear to be an intended limitation for movement of a moveable experiment, consider revising the definition to be consistent with the guidance in ANSI/ANS 15.1-2007, defined in terms of Table 1.1-1, or explain the definition as written.

The definition has been revised.

(e) *The UFTR TSs include a definition for "SECURED EXPERIMENT." This term is not used in the UFTR TSs, remove this definition or justify why it is necessary for inclusion in the UFTR TSs.*

The definition has been removed.

(f) *The UFTS TS definition for "SHUTDOWN MARGIN," (SDM), states "...instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition...." The ANSI/ANS-15.1-2007, guidance defines SDM in terms of the minimum shutdown reactivity necessary to provide confidence the reactor can be made subcritical "from any permissible operating condition ....without further operator action." Explain how the UFTR TS definition of SDM meets the intent of the ANSI/ANS-15.1-2007, definition (i.e., "instantaneous" vs. "minimum," "present condition" vs. "any permissible condition, without regard for operator action," and other provisions that are important to the determination of SDM, such as temperatures, xenon reactivity, effect of experiment reactivity, etc.) or modify the definition to be consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1 and ANSI/ANS-15.1-2007.*

The definition has been revised.

(g) *The UFTR TS for "Shutdown," (MODE 3) for the "MODES OF OPERATION," stated in Table 1.1-1, does not consider the conditions cited in ANSI/ANS 15 1 2007, such as including consideration of the reactivity worth of all installed experiments that have a positive effect on core reactivity. Revise the Table or provide a definition that meets the intent of the guidance for reactor shutdown in ANSI/ANS-15-1 2007, or justify why a revised definition is not necessary.*

A definition for "reactor shutdown" has been added.

(h) *Secured" is a defined mode (MODE 4) in the UFTR TS for the "MODES OF OPERATION" stated in Table 1.1-1. The UFTR TS definition does not address conditions cited in the guidance of ANSI/ANS-15-1-2007, such as including either the sufficiency of moderator or fissile material or that all scrammable blades are fully inserted, console switch position is off, maintenance is not being performed as detailed, and no experiments are being moved. Revise the Table or provide a definition that meets the intent of the guidance for reactor secured in ANSI/ANS-15.1-2007, or justify why a revised definition is not necessary.*

A definition for "reactor secured" has been added.

(i) *The UFTR TSs do not define the term "reference core condition," which is recommended for use to clarify the conditions under which the SHUTDOWN MARGIN and excess reactivity are evaluated and used to support UFTR TS 3.1.*

Include a definition consistent with NUREG-1537, Part 1, Appendix 14.1, and incorporate in terms of the modes defined in UFTR TS Table 1.1-1, or justify why a definition is not necessary.

The term "reference core condition" is not used in the UFTR TSs. Language ensuring no credit is taken for negative experiment worth, temperature effects, or xenon poisoning is included in the TSs where needed.

(j) The UFTR TSs do not define the term "Reactivity Worth of an Experiment," which is used to clarify the conditions under which the SHUTDOWN MARGIN is evaluated. Provide a definition that meets the intent of NUREG-1537, Part 1, Appendix 14.1, or justify why a definition is not necessary.

A definition for "reactivity worth of an experiment" has been added.

(k) The UFTR TSs do not include a definition for "Secured Shutdown," per the guidance of NUREG-1537, Part 1, Appendix 14.1. Secured shutdown is achieved when the reactor meets the requirements of the definition of "Reactor Secured," and the facility administrative requirements for leaving the facility with no licensed reactor operators present. Revise Table 1.1-1 or provide a definition in the UFTR TSs that meets the intent of the guidance for secured shutdown or justify why a definition is not necessary.

The Table 1.1-1 has been replaced with word definitions describing the modes of operation. The term "secured shutdown" is not used anywhere in the UFTR TSs. Staffing requirements are specified in TS 6.1.3.

(l) The UFTR TSs do not establish any criteria or parameters that characterize when fuel is acceptable for use nor define fuel damage as referenced in the UFTR TSs (e.g., UFTR TS 3.9.2). The guidance in NUREG-1537, Part 1, Appendix 14.1 includes consideration of damage (out of tolerance dimensional changes or indications of cladding failure as indicated by the presence of detectable amounts of fission products), deterioration (such as erosion, corrosion, blistering, observed defects, etc.), oxide buildup, and fission density. Provide a definition for acceptable fuel integrity consistent with the guidance in NUREG-1537, Part 1, Appendix 14.1, and supports UFTR TS 3.9.2.1, on damaged fuel or justify why it is not needed.

Definitions for "damaged fuel" and "fuel defect" have been added. A SR to periodically sample and evaluate reactor water for indications of "damaged fuel" has also been added.

(m) Table 1.1-1 and the UFTR TS definition for "Mode," references a "key condition," for each of the modes of operation. Modify the table header for Table 1.1-1 and the definition for Mode to indicate that the key condition referenced is for the console key or justify why a specific reference to the console key is not required.

Table 1.1-1 has been removed and replaced with word definitions describing the modes

of operation. The word definition of "reactor secured" makes reference to the console key.

(n) A combination of core reactivity condition, power level, key condition, and concrete block shielding are listed in Table 1.1-1 of the UFTR TSs for each of the UFTR Modes of Operation. The LSSS in UFTR TS 2.2, provides a safety system setting for low reactor coolant flow that is not a condition for and is inconsistent with the modes defined by Table 1.1-1. Modify the UFTR TSs to ensure consistency between Table 1.1-1 and Table 2.2-1 or justify why a change is not needed.

Table 1.1-1 has been removed and replaced with word definitions describing the modes of operation. Specifying the reactor coolant flow condition within these definitions isn't necessary to define UFTR modes of operation. The water moderator condition however is referenced in the new "reactor secured" definition.

(o) The NRC staff finds the definitions given in ANSI/ANS-15.1-2007, generally acceptable and the guidance in NUREG-1537, Part 1, Appendix 14.1, states that those definitions applicable to a particular facility should be included verbatim. The UFTR TSs make use of a number of terms that are not defined in the TSs. Update the UFTR TSs to incorporate, the following ANSI/ANS-15.1-2007, definitions, as modified by NUREG-1537, Part 1, Appendix 14.1, where applicable:

i. operator, licensed operator, senior reactor operator, and facility operators;

The terms operator, senior operator, and facility licensee are defined in 10 CFR 55.4. The term licensed is synonymous with the term licensee according to ANSI/ANS-15.1-2007. The term licensee is also defined in 10 CFR 55.4. This is a case where verbatim repetition of the guidance definitions sets up a conflict with definitions imposed by the higher order regulation. To prevent this and ensure consistency with 10 CFR 55.4, the Technical Specification term senior reactor operator has been shortened to senior operator, the term licensed operator has been shortened to operator, and the term facility operators has been changed to facility licensee.

ii. license, licensee, and licensed;

The terms license and licensee are already defined in 10 CFR 50.2. The term licensed is synonymous with the term licensee according to ANSI/ANS-15.1-2007. The term licensee is also defined in 10 CFR 55.4. Verbatim repetition of guidance definitions within the Technical Specifications in this case is unnecessary and sets up a conflict with definitions imposed by the higher order regulation(s).

iii. unscheduled shutdowns

A definition for "unscheduled shutdown" has been added.

(p) UFTR TS 1.2, establishes the surveillance intervals for each surveillance requirement

or audit. The specified frequency for "Daily," is stated to be "Daily - interval not to exceed 36 hours." ANSI/ANS-15.1-2007, recommends a maximum interval for daily surveillances of (i) during the calendar day; (ii) within a shift, and, (iii) prior to the first reactor startup of the day. Modify the UFTR "Daily," surveillance interval to be consistent with the guidance of ANSI/ANS-15.1-2007, or provide an analysis to justify the 36 hour interval.

The "Daily" interval has been revised to read "…not to exceed 24 hours" to be equivalent to the length of a calendar day. UFTR TS 1.2 has also been revised to simplify determination of maximum allowable interval lengths by eliminating reference to the 1.25 multiplier and adding allowable intervals for biennial, annual, and semiannual.

2. *Safety limits, limiting safety system settings, and limiting control settings are defined in 10 CFR 50.36(c). Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. The following questions pertain to Section 2.0, Safety limits and Limiting Control Settings of the proposed UFTR TSs. Provide a response that addresses each issue identified, or propose a suitable alternative in your response.*

   a. *UFTR TS 2.1 and UFTR TS 2.2 provide an applicability statement referencing the "Modes of Operation," listed in Table 1.1-1 of the UFTR TSs. The ANSI/ANS-15.1-2007, guidance states the applicability information should be "a statement that indicates which components are involved and when they are involved." Reference to Modes from Table 1.1-1 provides the "when," but does not provide the "which." Modify the Applicability statement for UFTR TS 2.1, and TS 2.2, to indicate the components to which the specification applies or justify why this information is not needed.*

   The component (or parameter or variable) to which the specification applies is within the specification itself.

   b. *UFTR TS 2.1, "Fuel Temperature Safety Limit." Revise UFTR TS 2.1, by removing the phrase "Fuel Temperature," to be consistent with the term "Safety Limit," as formally referenced in 10 CFR 50.36(c)(1).*

   The phrase "Fuel Temperature' has been removed from the TS 2.1 title line.

   c. *UFTR TS 2.1, states that the SL is applicable in MODE 1. However, 10 CFR 50.36(c)(i)(A), states that the SL for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. Such limits apply under all operating modes and conditions. Revise UFTR TS 2.1, to be consistent with the regulations.*

UFTR TS 2.1 applicability has been revised to include both Modes 1 and 2 (all operating modes).

d. *UFTR TS 2.2 is titled "Limiting Control Settings." The regulation in 10 CFR 50.36, provide that the accepted term is "Limiting Safety System Settings," for nuclear reactors (limiting control settings apply only to fuel reprocessing plants). Revise UFTR TS 2.2, including the title for Table 2.2-1 and the header in UFTR TS 2.0, to be consistent with 10 CFR 50.36(c)(1)(ii)(A) by renaming "Limiting Control Settings" to "Limiting Safety System Settings."*

The term "Limiting Control Settings" has been replaced with "Limiting Safety System Settings".

e. *The objective for UFTR TS 2.2, is stated to be "[t]o ensure automatic action terminates the off-normal situation." The regulations in 10 CFR 50.36(c)(ii)(A) states "automatic protective action will correct the abnormal situation before a safety limit is exceeded." Modify the objective for UFTR TS 2.2, to be consistent with 10 CFR 50.36.*

The objective has been revised.

f. *UFTR TS 2.2, states that the Limiting Control Settings (which, per the regulations, should be LSSS for a nuclear reactor) is applicable in MODE 1. However, 10 CFR 50.36(c)(ii)(A) states that the LSSS pertains to "settings for automatic protective devices related to those variables having significant safety functions." The settings in the revised Table 1.1-1 are such settings and thus the LSSS is applicable during operation (i.e., to both MODES 1 and 2 as referenced in Table 1.1-1). Revise UFTR TS 2.2, to be consistent with 10 CFR 50.36.*

UFTR TS 2.2 applicability has been revised to include both Modes 1 and 2.

g. *The current approved UFTR TS contain 12 LSSS specifications. The proposed UFTR TS 2.2 has three specifications. Revise UFTR TS 2.2 to include any missing specifications and provide an analysis justifying those that are removed or justify why this information is not needed.*

Since the time of the previous license renewal, well prior to the advent of ANSI/ANS-15.1 and NUREG-1537, the UFTR Technical Specifications have effectively been a cut-n-paste of all RCS/RPS functions described in the SAR. They exist in the Technical Specifications simply because the equipment functions as installed and described in the SAR at that time were copied over into the Technical Specifications.

Since then the Commission and staff have stated the intent of license renewal now is to ensure a single, complete, accurate, and contemporary licensing basis meeting current requirements and guidance. The UFTR has made every feasible effort to meet the

requirements and guidance expectations including re-performance of the core and accident analyses and a complete reconstitution of all licensing basis documents derived from those analyses including the SAR, Emergency Plan, and the Technical Specifications

Proposed TS 2.2 meets the requirements of 10 CFR 50.36 as well as the guidance of ANSI/ANS 15.1-2007 Sections 1.2.1 & 2.2 and NUREG-1537 Section 2.2. By keeping allowable excess reactivity low, the UFTR is able to demonstrate that postulated reactivity insertion events require no automatic control or safety functions to prevent reaching the Safety Limit. This conclusion is consistent with the NRC's own findings for low power RTRs like the UFTR (Refs: SECY-15-0081 and NUREG-2150). Therefore, the three LSSS specifications proposed are non-safety related trip set points that were conservatively chosen to provide defense-in-depth by ensuring normal operation remains bounded by the normal thermal hydraulic analysis. The typical low or medium power RTR only has one high power or high fuel temperature LSSS.

Currently, the twelve LSSSs are as follows:

- LSSS 2.2(1) Power level at any flow rate shall not exceed 119 kW.
- LSSS 2.2(2) The primary coolant flow rate shall be
  (a) greater than 36 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is 15 mils.
  (b) greater than 41 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is 20 mils.
- LSSS 2.2(3) The average primary coolant
  (a) inlet temperature shall not exceed 109F when the fuel coolant channel spacing tolerance is 10 mils.
  (b) inlet temperature shall not exceed 99F when the fuel coolant channel spacing tolerance is 20 mils.
  (c) outlet temperature shall not exceed 155F when measured at any fuel box outlet.
- LSSS 2.2(4) The reactor period shall not be faster than 3 sec.
- LSSS 2.2(5) The high voltage applied to Safety Channels 1 and 2 neutron chambers shall be 90% or more of the established normal value.
- LSSS 2.2(6) The primary coolant pump shall be energized during reactor operations.
- LSSS 2.2(7) The primary coolant flow rate shall be monitored at the return line.
- LSSS 2.2(8) The primary coolant core level shall be at least 2 in. above the fuel.
- LSSS 2.2(9) The secondary coolant flow shall satisfy the following conditions when the reactor is being operated at power levels equal to or larger than 1 kW:
  (a) Power shall be provided to the well pump and the well water flow rate shall be larger than 60 gpm when using the well system for secondary cooling.
  (b) The water flow rate shall be larger than 8 gpm when using the city water system for secondary cooling.
- LSSS 2.2(10) The reactor shall be shut down when the main alternating current (ac) power is not operating.
- LSSS 2.2(11) The reactor vent system shall be operating during reactor

operations.

- LSSS 2.2(12) The water level in the shield tank shall not be reduced 6 in. below the established normal level.

As stated previously, by keeping allowable excess reactivity low, the UFTR is able to demonstrate that postulated reactivity insertion events require no automatic control or safety functions to prevent reaching the Safety Limit and therefore no LSSSs are required. However, to support the defense-in-depth concept and support generation of a reasonable set of Technical Specifications, the fundamental reactor parameters of power, temperature, and flow were conservatively chosen for incorporation into LSSSs. This also ensures normal reactor operation remains bounded by the normal thermal hydraulic analysis. Therefore, for license renewal the:

- Current LSSS 2.2(1) for reactor power has been lowered from 119 kW to 110 kW. This reactor power LSSS was conservatively chosen to provide defense-in-depth by ensuring normal operation remains bounded by the normal thermal hydraulic analysis (i.e to keep ONBR >1). However, the reactivity insertion analyses demonstrate that automatic protective actions related to reactor power are not required for protection of the Safety Limit even for the hypothetical event that results in coolant boiling due to the self-limiting design and minimal decay heat generation.
- Current LSSS 2.2(2) is reduced to a single LSSS for primary flow of 41 gpm that incorporates the limiting water channel spacing tolerance of 20 mils (Ref. ML062440101). This flow LSSS was conservatively chosen to provide defense-in-depth by ensuring normal operation remains bounded by the normal thermal hydraulic analysis (i.e to keep ONBR >1). The reactivity insertion analyses demonstrate that automatic protective actions related to primary flow are not required for protection of the Safety Limit. Also reference response for Technical RAI #5.
- Current LSSS 2.2(3) is reduced to a single LSSS for primary inlet temperature of 102F that incorporates the limiting water channel spacing tolerance of 20 mils. This temperature LSSS was conservatively chosen to provide defense-in-depth by ensuring normal operation remains bounded by the normal thermal hydraulic analysis (i.e to keep ONBR >1). The reactivity insertion analyses demonstrate that automatic protective actions related to primary temperature are not required for protection of the Safety Limit.
- Current LSSS 2.2(4) is removed since the reactivity insertion analyses demonstrate that automatic protective actions related to reactor period are not required for protection of the Safety Limit. A fast period trip is conservatively required however in TS 3.2.2 as further defense-in-depth to ensure early termination of a reactivity insertion event originating from low power levels.
- Current LSSS 2.2(5) is removed since the reactivity insertion analyses demonstrate that automatic protective actions for loss of detector high voltage are not a requirement for protection of the Safety Limit. Proper high voltage to the nuclear instruments is required however for nuclear instrument operability. This operability is ensured by the combination of LCOs and SRs directly relevant to reactor power channel operability. These include: TSs 3.0.1, 3.0.2, 3.2.2, 3.2.3,

6.2.3, 6.4; and SRs 3.2.2.1, 3.2.3.2, and 3.2.3.3. The loss of high voltage function described in SAR Section 7.3.1 is verified quarterly by SR 3.2.2.2. Calibration of the power channels is performed annually as required by SR 3.2.3.3 which includes neutron detector "plateau" determinations and high voltage checks to ensure the detectors are operating correctly and in the proper detection region. This same annual SR also includes calibration of detector indication to calculated thermal power (i.e. calorimetric calibration). A significant and unexpected loss of detector high voltage at power that didn't cause a trip would be detected by the operator when cross checking and logging the multiple indications of power. This includes verification of reactor coolant flow and temperature difference across the core for indication of calorimetric power (actual power). These checks are required at a minimum of every twenty minutes when steady state and also following any change in power level by procedure SOP-A.3, Operation at Power.

- Current LSSSs 2.2(6) and 2.2(7) are effectively redundant to current LSSS 2.2(2) and were removed since the reactivity insertion analyses demonstrate that automatic protective actions related to primary flow are not required for protection of the Safety Limit. See primary flow related bullet above also.
- Current LSSS 2.2(8) is removed since the reactivity insertion analyses demonstrate that automatic protective actions related to coolant level are not required for protection of the Safety Limit. A low coolant level trip is conservatively required by TS 3.2.2(5) to provide defense-in-depth to ensure control blades stay fully inserted (i.e. de-energized and in the tripped condition) until the minimum core water level is reached. This control blade inhibit function also provides additional defense-in-depth in the form of redundancy for TS 3.2.2(2) since a loss of coolant flow also causes a loss of coolant level.
- Current LSSS 2.2(9) is effectively redundant to current LSSS 2.2(3). This LSSS has been removed since the reactivity insertion analyses demonstrate that automatic protective actions related to secondary flow are not required for protection of the Safety Limit. Depending on reactor power level, a loss of secondary flow may result in an increased primary inlet temperature which is covered under LSSS 2.2(3) above.
- Current LSSS 2.2(10) is removed since the reactivity insertion analyses demonstrate that automatic protective actions related to ac power are not required for protection of the Safety Limit. The source of power also has no bearing on safe operation. Note however that a loss of ac power will result in reactor shutdown due to gravity insertion of the control blades and gravity dumping of the water moderator.
- Current LSSS 2.2(11) is removed since the reactivity insertion analyses demonstrate that automatic protective actions related to the reactor vent system are not required for protection of the Safety Limit. Reactor vent system operability is ensured by TS 3.5 and SRs 3.5.1 and 3.5.2.
- Current LSSS 2.2(12) is removed since the reactivity insertion analyses demonstrate that automatic protective actions related to the shield tank level are not required for protection of the Safety Limit. Proper shield tank level is ensured by TS 3.9.1 and SR 3.9.1.

3. *A limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the criteria provided in 10 CFR 50.36(c)(2)(ii). Regulation 10 CFR 50.36 also requires the inclusion of surveillance requirements that prescribe the frequency and scope of the surveillance necessary to demonstrate the required performance. The following questions pertain to Section 3, "Limiting Conditions for Operation and Surveillance Requirements" of the proposed UFTR TSs. Provide a response that addresses each issue identified, or propose a suitable alternative in your response.*

   a. *The UFTR TS utilize MODES to define when LCOs are applicable. The UFTR TS are not clear regarding what conditions are required for entry into or exit from any given MODE. The NRC staff position is that entry into or exit from a MODE requires that the conditions of that MODE be satisfied prior to the MODE change. Revise UFTR TS 3.0.1, to provide conditions for each MODE that must be satisfied prior to the MODE change or justify why specific conditions are not needed.*

   Conditions (LCOs) that must be met prior to Mode entry are specified within each LCO specification.

   b. *UFTR TS 3.0.1, proposes that the action time for an LCO not being met is "within 15 minutes of discovery." There is no basis provided for the acceptability or determination of the 15 minute time limit. The NRC staff position is that unless justified otherwise, action times for non-compliance with LCOs require corrective action to be taken immediately or "without delay and in a controlled manner." Similarly, since action times other that IMMEDIATELY have not been justified by the licensee, the statement: "Where corrective measures are completed that permit operation in accordance with the LCO, completion of the actions required by LCO 3.0.1 is not required," is also not justified. Revise UFTR TS 3.0.1, and the associated Basis to specify corrective action must be taken immediately, to include an acceptable definition for IMMEDIATELY consistent with NRC staff position, or provide a safety analysis report (SAR) justification for the 15 minutes of discovery of failure to meet the LCO.*

   In the event a SSC becomes inoperable (during a condition when required to be operable) the operator needs sufficient time to make a determination of inoperability, reference the applicable Technical Specifications and/or SOPs, and initiate the appropriate action. Proposed TS 3.0.1 has been revised to include a list of those SSC-related LCOs for which the 15-minute allowable action time is necessary and justified. The SAR provides sufficient justification to demonstrate these SSCs have little to no relationship to safety under any credible scenario. By keeping allowable excess reactivity low, the excess reactivity insertion analyses described in SAR Chapter 13 demonstrates that no automatic controls or safety functions are necessary to prevent a radiological release or Safety Limit violation. This conclusion is consistent with the NRC's own findings for low power RTRs like the UFTR (Refs: SECY-15-0081 and NUREG-2150). Based on this discussion, the proposed 15-minute allowed action time is appropriate, safe, and justified.

c. *The Applicability, Objective, and Specifications for UFTR TS 3.0.2 make reference to "the applicable system, structure, or component (SSC)." The guidance in ANSI/ANS 15-1-2007, states "[l]imiting conditions for operations (LCOs) are those administratively established constraints on equipment and operational characteristics that shall be adhered to during operation of the facility." Further, throughout the 10 CFR regulations, SSCs are referenced as being either safety- and security- related SSCs or as a risk-informed safety class, neither of which appear to apply to UFTR TS 3.0.2. Provide a response that addresses each issue identified for TS 3.0.2, or propose a suitable alternative in your response*

   i. *UFTR TS 3.0.2, "SSC," is not adequately inclusive of properties and functional capabilities, such as stated in UFTR TSs 3.1, UFTR TS 3.7.2, and UFTR TS 3.8. The surveillance requirement applicability (as defined in UFTR TS 3.0.1) is applicable to all items cited by your LCOs. Use another term, consistent with the regulations and the guidance of ANSI/ANS-15.1-2007, or provide an acceptable definition for "SSC," in Section 1.1 of the UFTR TS.*

   A definition of SSC has been added to TS 1.1. UFTR TS 3.0.2 has been revised to include limits on variables.

   ii. *UFTR TS 3.0.2, Specification (1), states "[s]urveillances do not have to be performed on inoperable equipment or variables outside specified limits." The UFTR TS deferral or exception for performance of surveillances is too vague. Provide a response that addresses each issue identified for TS 3.0.2, Specification (1), or propose a suitable alternative in your response*

      (1) *The guidance in ANSI/ANS-15.1-2007, states that "[f]or each surveillance requirement (SR), it should be specified if the surveillance activity can or cannot be deferred during reactor shutdown." It should also be specified for those that can be deferred, which must be performed prior to reactor operations. Revise the UFTR TS to specifically indicate which SR can be deferred and which must be performed prior to reactor operations, consistent with the ANSI/ANS-15.1-2007, guidance.*

      Proposed TS 3.0.2 has been revised to clarify which SRs must be performed prior to entry into the specified Mode or condition and which SRs (by list) require entry into the specified Mode or condition for performance of the SR.

      The statement "Surveillances do not have to be performed on inoperable equipment or variables outside specified limits" has been removed.

      (2) *The term "variables," is not defined in the UFTR TS. Revise UFTR TS 3.0.2, Specification (1), to use defined terms or provide a definition for "variables" in the UFTR TSs.*

      The term "variables" and variations thereof are neither facility specific nor UFTR

specific in any way. The term is used successfully without need for formal definition throughout NUREG-1537, ANSI/ANS-15.1-2007, the Code of Federal Regulations, and other facility Technical Specifications. Adding this definition would add unnecessary clutter to the UFTR TSs.

iii. *UFTR TS 3.0.2, Specification (3), states "[a]ppropriate surveillance testing on any Technical Specification required SSC shall be conducted after replacement, repair, or modification before the SSC is considered OPERABLE except as provided in UFTR TS 3.0.2 (2)." The guidance in ANSI/ANS-15.1-2007, states that any item, equipment, or condition that is controlled by an LCO shall be evaluated using the applicable SR after replacement, repair, or modification before placing such equipment in service. There is no acceptable exception to performing the TS required SR before placing such equipment in service. Modify UFTR TS 3.0.2, Specification (3), to eliminate the exception allowed in UFTR TS 3.0.2, Specification (2), or provide a safety analysis justifying why and when such an exception would be needed and acceptable.*

The exception has been revised to include only those SRs which require entry into the applicable Mode or other specified condition for performance of the SR.

d. *UFTR TS 3.1, is stated to be applicable to MODES 1 and 2. The guidance in NUREG-1537, Part 1, Section 4.5.3 states the applicant should present information on "[the amount of negative reactivity that must be available by control rod action to ensure that the reactor can be shut down safely from any operating condition and maintained in a safe shutdown state." Accordingly, the applicable MODES for UFTR SDM should be MODES 1 through 5. Revise the applicability of UFTR TS 3.1, to be all modes (i.e., Modes 1 through 5) consistent with the Table 1.1-1 definition of MODES.*

The applicability of TS 3.1 has been revised to Modes 1 through 5.

e. *UFTR TS 3.1, is applicable to reactor core reactivity parameters. Section 3.1, of ANSI/ANS-15.1-2007, NUREG-1537, Part 1, Section 4.5, and Appendix 14.1, Section 3.1, provide guidance for TS related to reactivity parameters, core configuration, fuel burnup limits, and fuel inspections. Respond to the following considering this guidance as related to UFTR TS 3.1.*

i. *The applicability of UFTR TS 3.1 is stated to be Modes 1 and 2. Per NRC guidance, the conditions for these reactor core reactivity parameters apply at all times (i.e., UFTR Modes 1 to 5). Modify the applicability statement of UFTR TS 3.1, to include all modes and indicate the components to which the specification applies or justify why this information is not needed.*

The applicability of TS 3.1 has been revised to Modes 1 through 5. UFTR TS 3.1 applies to core reactivity parameters only.

ii. *The basis for UFTR TS 3.1, is not consistent with the basis provided for SDM in ANSI/ANS-15.1-2007, in that it does not include that the reactor will remain subcritical without further operator action. Modify the basis statement of UFTR TS 3.1, to be consistent with the guidance or justify why this information is not needed.*

The definition of SDM itself (provided in Section 1.1) includes the appropriate wording regarding operator action and therefore consistency between the basis statement of TS 3.1 and the ANSI/ANS standard is assured.

iii. *NUREG-1537, Appendix 14.1, Section 1.2.2, TS that use the SAR as a basis should explicitly reference the SAR section. In addition, any other sources used to support the TS should be explicitly referenced. Add a SAR reference for the accident analyses reference in the basis for UFTR TS 3.1, or justify why this information is not needed.*

The SAR reference has been added.

iv. *The UFTR SR for UFTR TS 3.1 includes a footnote indicating reactivity parameters [i.e., SDM and excess reactivity] be verified within limits following changes in core configuration. Core configuration should be controlled by an LCO using the guidance in NUREG 1537, Part 1, Section 4 and Appendix 14.1, Section 3.1. Revise UFTR TS 3.1, specification and associated SR to add an LCO for Core Configuration or explain why these are not required.*

Core configuration requirements are specified by the core design, fuel loading, and fuel design limits imposed in revised TS Section 5.3. TS 5.3.2(1) was added to specify a minimum of 22 full fuel assemblies (the limiting core configuration) in Modes 1 through 3.

f. *The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.2, recommends using control rod operability and maximum insertion rate LCOs/SRs that are not included in the UFTR TS. Revise UFTR TS 3.2.1, to include the following LCOs and SRs or explain why these are not required:*

i. *an LCO and corresponding SRs to specify the minimum number and type of operable control and safety rods and reference the applicable analysis in the UFTR SAR supporting these TS in the bases for the TS.*

The number and type of control blades is controlled by design limitation TS 5.3.1(2). Control blade operability is assured by TS 3.2.1, TS 3.0.2(1), and TS 3.0.1(1).

ii. *an LCO and corresponding SRs to specify the maximum rates of adding positive reactivity and reference the applicable analysis in the UFTR SAR supporting these TS in the bases. The specification should explicitly state if gang or multiple*

blade withdrawal is allowed.

The prompt and ramp reactivity insertion analyses demonstrate that reactor safety is assured regardless of reactivity insertion rate provided that installed excess reactivity remains within TS 3.1(2). However, after discussion with NRC staff, TS 3.2.2 has been revised to include a fast period trip for defense-in-depth to ensure early termination of reactivity insertion events originating from low power levels.

g. *NUREG-1537, Part 1, Sections 4.5.3 and Appendix 14.1, Section 3.2, item (2) provide guidance concerning acceptable reactivity addition rates, including inadvertent addition of ramp reactivity at the maximum rate for the most conservative power, rod position, and reactor conditions to demonstrate acceptability. Revise UFTR TS 3.2.1, or add another TS to address ramp reactivity and also ensure that the SR frequency includes considerations for performing this SR after changes to the core configuration.*

The prompt and ramp reactivity insertion analyses demonstrate that reactor safety is assured regardless of reactivity insertion rate provided that installed excess reactivity remains within TS 3.1(2). However, after discussion with NRC staff, TS 3.2.2 has been revised to include a fast period trip (and associated SR) for defense-in-depth to ensure early termination of reactivity insertion events originating from low power levels.

h. *The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.2, item (4) states "[a] table should specify all required scram channels and setpoints, the minimum number of channels, [and] other functions performed by the channel." Revise UFTR TS 3.2.2 including the Bases to address the following issues or justify why they are not required:*

   i. *Revise UFTR TS 3.2.2, including the Basis, to specify the number of channels available and required for the MODES specified;*

   NUREG-1537, Part 1, Appendix 14.1, Section 3.2 does not require a listing in the Technical Specifications of <u>available</u> scram channels (only required scram channels or specified minimum number of scram channels in the case of ANSI/ANS-15.1-2007). This is demonstrated in part by other statements in NUREG-1537 such as Part 1, Appendix 14.1, Section 3.2, item (1) that "NRC finds it acceptable to shutdown a non-power reactor by intentionally scramming the control and safety rods" as well as statements in ANSI/ANS-15.1-2007, Sections 1.2.1 and 3.2(4). Please review the response to RAI 2.g also.

   ii. *Revise UFTR TS 3.2.2, including the Basis, to include the "Allowable Condition or Value" for both Mode 1 and Mode 2 for scrams 1 through 4 identified in Table 3.2.2-1;*

   The applicability of TS 3.2.2 (and the bases where appropriate) has been revised to Modes 1 and 2 (reactor operation) for all trips.

iii. UFTR Table 3.2.2-1 does not include information pertaining to full trip vs. blade trip described in Section 7 of the UFTR SAR. Revise UFTR TS 3.2.2, including the Basis, to include the type of reactor trip for each "Function," identified in Table 3.2.2-1.

SAR Section 7.3.1 describes that a full-trip is one where the primary water is dumped into the storage tank in conjunction with a blade-trip. For purposes of core and event analyses and related Technical Specification limits, only the reactivity worth of the blade-trip is assumed or credited. As Technical Specifications are intended to be the minimum subset of functions needed for safety (not simply a repeat of all functions described in the SAR), there is no need or requirement to list (and hence require by LCO) the water dump function in the Technical Specifications.

iv. UFTR Table 3.2.2-1 does not list all UFTR reactor trips listed in SAR Section 7.3.1, "Trip Circuits." Add all UFTR reactor trips to UFTR Table 3.2.2-1 or justify why they are not needed.

NUREG-1537, Part 1, Appendix 14.1, Section 3.2 does not require a listing in the Technical Specifications of available reactor trips (only required scram channels or specified minimum number of scram channels). Proposed TS 3.2.2 meets the requirements of 10 CFR 50.36 as well as the guidance of ANSI/ANS 15.1-2007 Sections 1.2.1 & 3.2 and NUREG-1537 Section 3.2. Please review the response to RAI 2.g also.

v. UFTR SR 3.2.2.1 is redundant to UFTR SR 3.2.2.2 except for "Frequency." Verify that both SRs are intended to be for CHANNEL TEST or modify TS SR 3.2.2.1 if it should refer to a CHANNEL CHECK instead.

SR 3.2.2.1 has been changed to Channel Check.

i. The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.2, item (5), states "interlocks that inhibit or prevent control rod withdrawal or reactor startup should be specified by a table." UFTR SAR, Sections 7.1.3.1.1, 7.1.3.1.2, 7.2.2.2, and 9.1.2 all describe the importance of interlocks to UFTR. Revise UFTR TS 3.2, to provide LCOs/SRs for all interlocks described in the SAR or explain why they do not need to be in the TS.

The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.2, item (5), actually states that "<u>Required</u> interlocks that inhibit or prevent control rod withdrawal or reactor startup should be specified by a table." And, the RAI referenced SAR sections actually describe the existence of, and functions of, these interlocks as opposed to their importance to safety. The reactivity insertion analyses demonstrate that automatic control blade withdrawal inhibits and automatic startup inhibits are not required for protection of the Safety Limit. None of these interlocks are credited or relied upon in the SAR accident analysis and therefore none are required in the

Technical Specifications.

Note however that reactor power measuring channels are required to be verified operable by SR under TS 3.2.3(1). The associated surveillance performance includes verification that the control blade and startup inhibits described in the SAR function as designed to support operability of the reactor power measuring channels. Additionally, the core vent and stack dilute systems are required to be verified operable as specified by SR 3.5. Core vent system functions described in the SAR are also verified by surveillance. Surveillances to demonstrate LCO compliance are performed in accordance with approved written procedures as required under TS 6.4(4).

j. *The guidance in NUREG-1537, Appendix 14.1, Section 3.3, item (5), states that the TS "should provide for prompt detection of fission products escaping from the fuel barrier." Modify the UFTR TS to clearly provide an LCO and associated SR to provide for prompt detection of possible fission products escaping from the fuel barrier or justify why these are not needed.*

Early detection of fission products is ensured by TSs 3.3.2, 3.4, and 3.7.1. The Basis statements for these TSs have been revised to clarify this.

k. *UFTR TS 3.4, establishes the requirement that the automatic actuation of the evacuation alarm requires two simultaneous area radiation monitors (ARMs) to alarm high. UFTR SAR Section 11.1.4.1, summarizes the existence of ARMs, but does not provide any information on the location or total number of ARMs. The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.7.1, item (3) states alarm and automatic action setpoints should be specified to ensure that personnel exposures and potential doses remain well below limits of 10 CFR Part 20, and are consistent with the facility ALARA (as low as is reasonably achievable) program. Explain and revise the UFTR SAR, TS 3.4, or both, as needed, to address the following concerns:*

   i. *Explain how many ARM channels are available to perform this function, where they are located, their alarm setpoint, and the basis for that setpoint.*

   The area radiation monitoring system is described in SAR Section 7.6. More detail will be added to SAR Section 7.6 to address the normal area monitor locations and setpoints. There are three area monitors with remote detector assemblies mounted on the North, South, and East walls of the Reactor Cell. Two levels of alarm are provided. A warning alarm (typically set to 2.5 mR/hr) and a high alarm (typically set to 10 mR/hr). These typical setpoints are appropriate for normal full-power operation with all shielding and port plugs installed.

   ii. *The evacuation alarm interlock is credited in the Basis for UFTR TS 3.4, as a function "designed to alert the staff and occupants of a radiological emergency." Add a SAR reference for the analyses supporting this LCO in the basis for UFTR TS 3.4, or justify why they are not needed.*

The basis has been updated to point to SAR Chapter 7 which includes a description of the evacuation alarm interlock function. Inclusion of an evacuation alarm interlock LCO is beyond the requirements of the regulation or the recommendations within the guidance. However, the UFTR proposes to voluntarily include this LCO to ensure an audible alarm is available to warn operators and staff of a potential radiological event during modes or conditions when that potential exists.

l. *UFTR TS 3.5, states "[t]he core vent and stack dilution systems shall be operating and maintaining REACTOR CELL pressure negative with respect to the surrounding environment." The guidance in ANSI/ANS-15.1-2007, Section 4, states, in part, that the "surveillance specification requirement will prescribe the frequency and scope of surveillance to demonstrate [LCO] performance."*

   i. *With regard to the scope of SR 3.5 identify what equipment provides indication of acceptable reactor cell pressure to demonstrate LCO performance.*

The pressure differential can be measured by appropriate hand-held or permanently installed instruments therefore no specific equipment is identified in the LCO. Measurements taken by hand-held differential pressure instrument confirm that cell pressure is negative relative to the surrounding environment when the core vent and stack dilution systems are in operation. Surveillances to demonstrate LCO compliance are described in detail and performed in accordance with approved written procedures as required under TS 6.4(4).

   ii. *Explain how "negative," pressure is measured and quantified and identify the operating pressure, flow rates, and any associated setpoints for the reactor cell and any other systems associated with reactor cell ventilation system.*

Differential pressure instruments have a reference port and a signal port. When pressure at the signal port is less than pressure at the reference port, the instrument will typically indicate a negative value for pressure. Similarly, when Reactor Cell pressure (signal side) is less than that of the surrounding environment (reference side) it is considered to be negative with respect to the surrounding environment. Design and operation of the core vent and stack dilution systems is described in SAR Sections 9.1.2 and 9.1.3.

   iii. *Add a SAR reference for the analyses supporting this LCO in the basis for UFTR TS 3.5, or justify why they are not needed.*

The basis statement has been revised to include a SAR reference.

m. *NUREG-1537, Part 1, Section 3.7, states that the radiation monitoring system and effluent LCOs/SRs should be stated in the TS. Explain and revise the SAR and UFTR TS 3.7.1, or both, as needed, to address the following concerns:*

i. The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.7.1, item (3), states "[a]larm and automatic action setpoints should be specified to ensure that personnel exposures and potential doses remain well below limits of 10 CFR Part 20, and are consistent with the facility ALARA program." Revise UFTR Table 3.7.1-1 in UFTR TS 3.7.1, to provide the allowable levels or alarm setpoints for the radiation monitoring system, provide a reference with this information, or justify why these are not needed.

Specifying radiation monitor setpoints as LCOs is not the norm for RTRs (or any other NRC regulated reactor). This is a case where the NUREG-1537 guidance is inconsistent with 10 CFR 50.36 definition for LCO: "…lowest functional capability or performance levels of equipment required for safe operation of the facility".

The setpoints are chosen to ensure exposures are ALARA taking into account normal background levels and best practices. Current setpoints for the radiation monitors are: three area monitors with warning alarm at 2.5 mR/hr and high alarm at 10 mR/hr; a stack monitor with fixed alarm set at 4,000 cps and an adjustable alarm setpoint based on power level; and, an APD with fixed alarm set to less than or equal to 10,000 cpm. Installed setpoints and their related functions are checked on a weekly basis (new SR 3.7.1.2) in accordance with procedure SOP-A.1, Pre-operational Checks. The setpoints and functions are also checked during calibration (new SR 3.7.1.3) surveillances which are implemented in accordance with procedure SOP-0.5, Quality Assurance Program.

On occasion, and in consultation with the Radiation Control Officer, the setpoints may need to be adjusted to accommodate experiments or other evolutions. Therefore it's important the facility be able to control radiation monitor setpoint limits via SOPs rather than LCOs.

ii. The number of monitoring systems required for operability is indicated in Table 3.7.1-1, but the number typically in service and where they are located is unclear in the TS and is not elaborated in the SAR, Table 11-4. Revise UFTR Table 3.7.1-1 in UFTR TS 3.7.1, or the SAR to provide this information, or justify why these are not needed.

The radiation monitoring system is described in SAR Section 7.6 including the number of installed monitors. The location of the stack monitor detector is described in SAR Chapter 9. More detail will be added to SAR Section 7.6 to address location of the area radiation monitors and air particulate detector (APD).

iii. ANSI/ANS-15.1-2007, recommends that radiation monitoring systems have an LCO for operability including, where possible, source checks. Add an LCO and SR or SAR reference for the analyses in the basis for UFTR TS 3.7, or justify why they are not needed.

The basis statement has been revised to include a SAR reference.

The radiation monitoring system described in SAR Section 7.6 includes description of the built-in Kr-85 check sources included in the area radiation monitor detectors. The stack monitor and APD do not have built-in check sources. Regardless of radiation detector type, however, the use of radioactive sources or check sources to demonstrate LCO compliance is inherent within the definitions of channel test and channel calibration (i.e. the SRs), the details of which are specified by approved written surveillance procedures as required under TS 6.4(4).

n. *NUREG-1537, Part 1, Section 3.7 states that the radiation monitoring system and effluent LCOs/SRs should be stated in the TSs. Explain and revise the SAR, and UFTR TS 3.7.2, or both, as needed, to address the following concerns:*

   i. *The objective of UFTR TS 3.7.2, states Ar-41 emissions will remain below applicable limits. The Specification portion of the TS should state the complete specification and need to reference the Basis. Modify UFTR TS 3.7.2 to clearly indicate the referenced applicable limits for Argon-41 emissions.*

   Proposed TS 3.7.2 has been revised.

   ii. *The Basis statement should provide reference to the SAR analyses that demonstrates the UFTR method adequately controls Argon-41 generation to within regulatory limits. Modify UFTR TS 3.7.2, Basis, to indicate the SAR reference or explain why it is not required.*

   The basis statement has been revised to include the SAR reference.

o. *The guidance in ANSI/ANS-15.1-2007, states, in part, that "a specific [LCO] will establish the minimum performance level, and a companion [SR] will prescribe the frequency and scope of surveillance to demonstrate such performance." In the case of experiments, ANSI/ANS-15.1-2007, Section 4.8, alternately states, in part, that "specific surveillance activities [for experiments] shall be established during the review and approval process as specified [under Administrative controls]." Modify UFTR TSs 3.8.1, UFTR TS 3.8.2, and UFTR TS 3.8.3 to add specific surveillances for the limitations on experiments, provide a reference to the administrative controls providing surveillance, or justify why a SR is not required.*

In its entirety, ANSI/ANS-15.1-2007, Section 4.8, states: "Specific surveillance activities shall be established during the review and approval process as specified in Sec. 6.2.3 and are typically not part of the technical specifications."

This experiment review and approval process is required under proposed TS 6.5. Note that the review function specified in ANSI/ANS-15.1-2007, Section 6.2.3 (3) is redundant to, and a subset of, Section 6.5 (1). Additionally, procedures for administrative control of experiments are required under TS 6.4. Procedure SOP-A.5, Experiments, describes the administrative controls and surveillance details.

p. The guidance in ANSI/ANS-15.1-2007, Section 1.2.2, states that "[t]he basis is a statement that provides the background or reason for the choice of specification or references a particular portion of the Safety Analysis Report." Modify the basis statement for UFTR TS 3.8.1, UFTR TS 3.8.2, and UFTR TS 3.8.3 to reference the corresponding SAR accident analyses or justify why this information is not required.

The basis statements for TS 3.8.1 and TS 3.8.3 have been revised to include the applicable SAR reference. The bases for TS 3.8.2 is the standard itself.

q. NUREG-1537, Part 1, Appendix 14.1, Section 3.8.2, states that potentially corrosive materials shall be doubly encapsulated. UFTR TS 3.8.2(2), states that known "corrosive materials in quantities greater that trace amounts shall be doubly encapsulated." Revise UFTR TS 3.8.2, Specification (2), to incorporate the stated guidance, define and justify by analysis the acceptability of "quantities greater that trace amounts," or explain why these revisions are not required.

UFTR TS 3.8.2(2) has been revised.

r. UFTR TS 3.8.3, uses the term "credible," failure with regard to limiting the quantity and type of fissile material in any experiment. This term does not provide sufficient limits on the occurrence or consequence for failure of an experiment. Revise UFTR TS 3.8.3, to delete the word "credible," from the UFTR TS Objective, UFTR TS 3.8.3, Specifications (1) and (2) or provide a definition for "credible," including a SAR reference to the analysis in the basis.

UFTR TS 3.8.3 has been revised.

s. The guidance in NUREG 1537, Part 1, establishes the principal objective of the shield design to ensure that the projected radiation dose rates and accumulated doses in occupied areas do not exceed the limits of 10 CFR Part 20, and the guidelines of the facility ALARA program. UFTR TS 3.9.1, establishes the requirement for the shield tank water level, but the basis statement does not establish dose limit constraints for "adequate radiation shielding." Revise UFTR TS 3.9.1, basis to identify that the specified shield tank level offers reasonable assurance that the shield tank can successfully prevent exceeding the limits of 10 CFR Part 20, and the guidelines of the UFTR ALARA program, including a SAR reference to the analysis.

The basis statement has been revised.

t. NUREG 1537, Part 1, Appendix 14.1, provides guidance on the periodic visual inspection of the fuel. This specification should be clear and explicit for detecting deterioration, including the intervals and methods of fuel inspection. UFTR TS 3.9.2, provides for a visual inspection on a 10-year periodic basis. However, it is not clear from the sample size of 8 in-core reactor fuel assemblies how the entire core will be inspected in the required frequency. Update UFTR TS 3.9.2, to provide a basis

reference that establishes the 10 year interval with a sample size of 8 in-core fuel assemblies is adequate to detect cladding deterioration that results from erosion, corrosion, or other damage.

With respect to periodic routine fuel inspections, NUREG 1537, Part 1, Appendix 14.1, Section 4.1(6) states: "Inspections for reactors with plate fuels have not been required by the technical specifications except for higher power reactors that refuel frequently". That said, the UFTR inspection interval and sample size were established and deemed adequate under License Amendment 24 (Ref. ML043510043). Based on subsequent discussion with the staff, adding a reference to LAR 24 in the basis statement is deemed unnecessary.

4. *Regulation 10 CFR 50.36 (c)(4) requires the inclusion of those design features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered elsewhere in the specifications. The following questions pertain to Section 5, Design Features (DF) of the proposed UFTR TS. Provide a response that addresses each issue identified, or propose a suitable alternative in your response.*

   a. *The guidance in ANSI/ANS-15.1-2007, Section 5.1, states, in part, that the site and facility description should include "[a] general description of the site and of the facility including location and exclusion or restricted areas." Revise UFTR TS 5.0, "Design Features," to include information to characterize the area and clearly discuss the area that is under the reactor license or justify why this information is not required.*

   A general description is provided in TS 5.1(1) with more details provided in SAR Chapter 2 and the UFTR Emergency Plan and Security Procedures. There are no exclusion areas or continually fixed restricted areas at the UFTR. Restricted areas are established as needed based upon the requirements of 10 CFR Part 20. Areas restricted for security purposes are described in the UFTR Security Procedures.

   b. *The guidance in ANSI/ANS-15.1-2007, states, in part, that "[i]f not included elsewhere in the technical specifications, features of the reactor room, such as ventilation system minimum free air volume, height of effluent release, etc., that are important to radiological safety and monitoring [should] be presented also. Revise UFTR TS 5.1, "Reactor Cell," DFs to include the free volume of the Reactor Cell and the ventilation system operational parameters (e.g., volumetric exhaust rate) that are not supplied elsewhere or justify why this information is not required.*

   UFTR TS 5.1 has been revised to specify a minimum reactor cell free volume. The value chosen assumes the reactor cell is effectively solid up to about 10' above floor level so the simplified minimum free volume becomes 30'x60'x20' = 36,000 cubic feet. Actual minimum free volume is closer to approx. 53,000 cubic feet. Ventilation system operational parameters (e.g., volumetric exhaust rate) are not relied upon for safety and therefore are not included in the Technical Specifications.

c. *NUREG-1537, Part 1, Appendix 14.1, Section 3.1 item (6)(c), "Materials Testing Reactor (MTR)-Type Fuel," states, in part, that to prevent fuel swelling there should be burnup limitations on the fuel specified for materials testing reactor (MTR)-type fuel. UFTR TS 5.3.2, does not include such a DF limiting fuel burnup. Revise UFTR TS 5.3.2, to provide a limit on uranium-235 burnup or fission density consistent with the SAR, which accounts for all relevant thermal-hydraulic and metallurgical considerations or justify why this information is not required.*

TS 5.3 was revised to eliminate nominal fuel specifications (per NRC request) and to limit burnup to 50% of the initial U-235 content. NUREG-1313 describes experiments showing the fuel can safely be used up to 98% burnup and that full burnup appears feasible as well.

5. *The following questions pertain to Section 6, Administrative Controls (AC) of the proposed UFTR TS. Regulation 10 CFR 50.36 (c)(5) states "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner." Provide a response that addresses each issue identified, or propose a suitable alternative in your response*

a. *UFTR TS 6.1.1, "Structure," references Figure 6-1, UFTR Organizational Chart. The proposed figure identifies the responsible Level 1 individuals as the "VP. Business Affairs," and the "Dean, College of Engineering." The current UFTR organizational chart lists the Level 1 (Figure 6.1) as the "UF President, Dean, College of Engineering, Chairman, Nuclear & Radiological Engineering Department." Modify Figure 6-1 to be consistent with the prior approved Level 1 individuals or provide a detailed explanation of how the Level 1 individuals identified in proposed Figure 6-1 will be responsible for the reactor facility's licenses, charter, and site administration.*

Figure 6-1 has been revised.

b. *UFTR TS 6.1.3, staffing has duplicate numbering for (1), (2), and (3). Revise UFTR TS 6.1.3, to eliminate duplicate numbering, such that each TS can be uniquely identified.*

The TS 6.1.3 and its numbering have been revised.

c. *UFTR TS 6.1.3(1) uses the term "licensed operator," but does not define what constitutes a "licensed operator." ANSI/ANS-15.1-2007, includes a definition for "licensed," and "operator," neither of which appropriately apply to the UFTR use of the term "licensed operator." Modify UFTR TS 6.1.3(1), and add the definition in Section 1.1 to incorporate the use of "reactor operator," from ANSI/ANS-15.1-2007, or provide a facility-specific definition for "licensed operator."*

TS 6.1.3 has been revised. See response to RAI 1(o)(i).

d. For "a designated second person," under Section 6.1.3, "Staffing," in ANSI/ANS-15.1- 2007, the guidance states "[u]nexpected absence for as long as 2 hours to accommodate a personal emergency may be acceptable provided immediate action is taken to obtain a replacement." UFTR TS 6.1.3(2), states unexpected absence is acceptable, but does not constrain the acceptability to an emergency situation. Modify UFTR TS 6.1.3(2), to be consistent with the guidance or explain why it is not necessary.

The guidance provided in ANSI/ANS-15.1-2007, Section 6.1.3(b) was clearly not intended to be copied verbatim into facility Technical Specifications. TS 6.1.3 has been revised to eliminate the 2-hr exception.

e. ANSI/ANS-15.1-2007, Section 6.1.3(c)(iii) provides guidance for parameters defining "reasonable time," (e.g., 30 minutes or within a 15-mile radius). The UFTR TS 6.1.3(3)c., also uses the term "reasonable time," but does not define the term. Revise UFTR TS 6.1.3, to incorporate the cited guidance, define "reasonable time," or explain why this information is not required.

TS 6.1.3 has been revised to eliminate the term "reasonable time".

f. The guidance in ANSI/ANS-15.1-2007, Section 6.1.3(3)(a), states, in part, that (a) initial startup and approach to power requires the presence at the facility of the senior reactor operator. Revise UFTR TS 6.1.3, to incorporate the cited guidance or explain why this information is not required.

The cited guidance has been incorporated.

g. UFTR TS 6.1.4, references the American National Standard, ANSI/ANS-15.4-2007, for selection and training of personnel. However, the UFTR requalification plan references, ANSI/ANS-15.4-1988. Revise TS 6.1.4 to be consistent with the requalification plan, reference the requalification plan, or explain why the references are, and need to be, different. Additionally, the NRC staff noted that ANSI/ANS-15.4 has been recently revised (ANSI/ANS-15.4-2016) and consideration should be given to referencing the latest version.

TS 6.1.4 has been revised consistent with the current requalification plan.

h. The term "SSC important to safety" is used in UFTR TS 6.2.3(3), however, it is not defined. Provide a suitable definition for "SSC important to safety," for the purposes of the UFTR TS, justify an alternative definition, or explain why one is not needed.

A definition of SSC has been added to TS 1.1. The phrase "important to safety" is changed to "having safety significance" to be consistent with the guidance.

i. *The UFTR TS 6.3, states "Radiation Control Officer shall be responsible for implementation of the radiation protection program," but does not state the recommended provision from ANSI/ANS-15.1-2007, Section 6.3, that this individual shall report to Level 1 or Level 2. Figure 6-1 in the UFTR TS shows the reporting responsibility for the Radiation Control Officer, but does not show the recommended provision from ANSI/ANS-15.1-2007, Section 6.3, that this individual shall report to Level 1 or Level 2. Revise UFTR TS 6.3, and Figure 6-1 to incorporate the cited guidance or explain why this change is not required.*

TS 6.3 has been revised. Figure 6-1 is consistent with the guidance.

j. *The last paragraph of UFTR TS 6.4, "Procedures," discusses changes to or temporary deviations from the procedures listed in (1) through (8). In part, the proposed TS states, (i) substantive changes will only be made only after review by the RSRS and approval by the Facility Director, (ii) minor modifications to procedures may be made by the Reactor Manager but must be approved by the Facility Director within 14 days. Explain how these allowable changes are performed, such that they are consistent with and meet the requirements of the regulations under 10 CFR 50.59, "Changes, tests and experiments," or modify UFTR TS 6.4, to specify approval of changes must be documented by conducting a 50.59 evaluation.*

TS 6.4 has been revised to eliminate discussion of substantive, minor, or temporary deviations, and to clarify that procedure changes require RSRS review and Facility Director approval. The procedure change process is performed in accordance with UFTR SOP-0.1, Operating Document Controls, to ensure procedure changes are reviewed against 10 CFR 50.59 and properly documented.

6. *The requirements for TS Bases are established in 10 CFR 50.36(a)(1), which states, in part, that "[a] summary statement of the bases or reasons for such specification, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications." ANSI/ANS-15.1-2007, states that the bases are statements that provide the background or reason for the choice of TS, or references a particular portion of the SAR that does. Provide a response that addresses each issue identified, or propose a suitable alternative in your response.*

   a. *The Basis for UFTR TS 3.1, states what the SHUTDOWN MARGIN and EXCESS REACTIVITY specifications represent, not why they are acceptable. It does not cite the restrictions imposed by the limiting core configuration nor the analysis that demonstrates the acceptability of that limiting core configuration in terms of the safety limits and NRC guidance or regulations. Revise the UFTR TS 3.1, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting these LCOs.*

   See previous responses to RAIs 3(e)(i), 3(e)(iii) and 3(e)(iv).

   b. *The Basis for UFTR TS 3.2.1, does not restate or reference the SAR analysis that*

demonstrates the acceptability of the control blade drop time specified consistent with NUREG-1537, Part 1, Section 4.5.3. Revise the UFTR TS 3.2.1, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.

The basis statement has been revised to include the SAR reference.

c. *The Basis for UFTR TS 3.2.3, does not restate or reference the SAR analysis that demonstrates the acceptability of the measuring channel information stated in Table 3.2.3-1. The Basis also does not address interlocks and their important attributes that are discussed in several places in the SAR. Revise the UFTR TS 3.2.3, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.*

As stated in the TS 3.2.3 basis, this Reactor Measuring Channel TS is only intended to provide the operator with the minimum required indication channels for adequate monitoring of steady state and transient conditions. The list of indicating channels is consistent with ANSI/ANS-15.1-2007 Section 3.2(5) as well as other Reactor Measuring Channel LCO's found in recently approved license renewal Technical Specifications at other RTR facilities.

d. *The Basis for UFTR TS 3.3.1, states why maintaining core water level is important, not why the value cited is acceptable. It does not cite the portions of the SAR that demonstrates the acceptability of the LCO in terms of the safety limits or relevant NRC guidance or regulations. Revise UFTR TS 3.3.1, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.*

The core water level LCO has been revised to require a trip function and moved to TS 3.2.2. Two inches above the fuel is the historical setpoint that's been demonstrated and deemed adequate to ensure the reactor core is fully moderated. This brings keff close enough to 1 so that the reactivity worth of control blades can then be used for normal reactor operation. The Low Reactor Coolant Level trip also provides redundancy to the Low Reactor Coolant Flow trip and acts as a blade withdrawal inhibit until the minimum core water level is reached. The basis statement for TS 3.2.2 has been revised accordingly.

For comparison purposes, a BOL 24-bundle core was evaluated for SDM with all water moderator drained and Control Blades 1 and 2 and the Regulating Blade fully inserted with Control Blade 3 fully withdrawn. The MCNP predicted SDM for this case is negative 51,652 pcm. The fully moderated BOL 22-bundle core with the same blade positions results in a SDM of negative 3,503 pcm.

e. *The Basis for UFTR TS 3.3.2, states the purpose of the water level sensor, not why the value cited is acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits or relevant NRC guidance or regulations. Revise UFTR TS 3.3.2, Basis, so that it summarizes or*

restates the analysis or reasoning presented in the SAR supporting this LCO.

One inch above floor level is the historical value that's been demonstrated and deemed adequate as an alarm setpoint to alert the operator of any substantial water leakage into the equipment pit. The basis statement has been revised.

f. The Basis for UFTR TS 3.3.3, states the purpose of the specifications, not why the values cited are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise the UFTR TS 3.3.3, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.

Based on discussions with staff this TS has been significantly revised. The resistivity limit is a historical value shown to minimize fuel assembly corrosion. A footnote was added stating that: "Normal transients and experiments can cause Reactor Coolant System water electrical resistivity to drop below 0.5 MΩ-cm for short periods of time. For these expected occurrences, reactor operations with electrical resistivity less than 0.5 MΩ-cm may continue for periods not to exceed 4 hours provided that continuous control room indication of reactor coolant resistivity is utilized and trended during that period."

The time limit of four hours was chosen since this is the historical time limit for this parameter. Additionally, four hours is the approximate length of time it takes to run the entire 200-gallon capacity of the reactor coolant system through the demineralizer portion of the system once.

g. The Basis for UFTR TS 3.4, states the purpose of the specifications, not why the actions and restrictions cited are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise UFTR TS 3.4, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.

See previous RAI responses related to TS 3.4.

h. The Basis for UFTR TS 3.5, states the importance of the specifications, not what the accepted operational parameters are and why they are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits or relevant NRC guidance or regulations. Revise the UFTR TS 3.5, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.

See previous RAI responses related to TS 3.5.

i. The Basis for UFTR TS 3.7.1, states what the specifications accomplish, not what the accepted setpoints are and why they are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits or relevant NRC guidance or regulations. Revise UFTR TS 3.7.1, Basis,

so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.

See previous RAI responses related to TS 3.7.1.

j. *The Basis for UFTR TS 3.7.2, states the applicable regulations and invokes administrative requirements to satisfy them - not the methodology employed, how it is translated into operating limitations, and why they are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise UFTR TS 3.7.2, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.*

See previous RAI responses related to TS 3.7.2.

k. *The Basis for TS 3.8.1, states the importance of the establishing the reactivity worths, but not why the stated values are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise UFTR TS 3.8.1, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.*

See previous RAI responses related to TS 3.8.1.

l. *The Basis for UFTR TS 3.8.2, states the importance of the establishing the reactivity worths, but not why the stated values are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise UFTR TS 3.8.2, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.*

TS 3.8.2 does not address reactivity worths. See previous RAI responses related to TS 3.8.2.

m. *The Basis for TS 3.8.3 states the importance of the establishing the experiment limits on fissile material and credible failure, but not the criteria that are applicable to the limitations or why they are acceptable. It does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise UFTR TS 3.8.3, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.*

See previous RAI responses related to TS 3.8.3.

n. *The Basis for UFTR TS 3.9.1, states the importance of the ensuring adequate shielding, but not why the value cited is acceptable. It does not cite the portions of the*

SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise UFTR TS 3.9.1, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.

See previous RAI responses related to TS 3.9.1.

o. The Basis for UFTR TS 3.9.2, states that it is important to operate with undamaged fuel because it ensures that the accident analysis is then bounding. The staff does not understand the assertion. It also states that limiting access as stated preserves the assumptions regarding fission product inventory. The staff does not understand this assertion. The Basis does not cite the portions of the SAR submitted that demonstrates the acceptability of the LCO in terms of the safety limits and NRC guidance or regulations. Revise UFTR TS 3.9.1, Basis, so that it summarizes or restates the analysis or reasoning presented in the SAR supporting this LCO.

The basis statement has been revised.

p. The regulation in 10 CFR 50.36(a)(1), requires that DFs shall have bases. The UFTR TS do not provide bases for the DFs. Revise the UFTR TS to provide acceptable bases for all UFTR DFs.

At the direction of the NRC staff, Section 5 has been revised to include the same sections as Section 3 except for SRs.

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).*

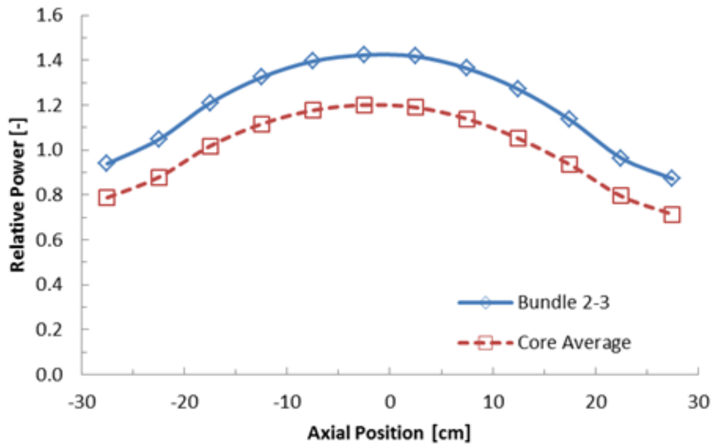
   NUREG-1537, Part 2, Section 4.6, "Thermal-Hydraulic Design," states, in part, that the SLs should include conservative consideration of the effects of uncertainties. Provide the peaking factors, or power shapes, used in the neutronics analysis, as well as any uncertainties, or justify why no additional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

   Power distributions were calculated axially per-bundle and radially across the core. Radial power distributions (F_Q) and axial power distributions (F_{dH}) were calculated as;

$$F_Q = \frac{P_i}{P_{core,average}}$$

$$F_{dH}(z) = \frac{P_i(z)}{P_{i,average}}$$

   Axial power distributions at the most limiting core condition (skewed blades) for the hottest bundle and the core average is shown below. Radial power distribution (FQ) is shown below as well.



0.91	1.01	1.04	0.96	0.80	
1.00	1.12	1.154	1.07	0.87	0.79
1.03	1.15	1.198	1.14	0.94	0.83
0.94	1.05	1.09	1.04	0.87	

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.

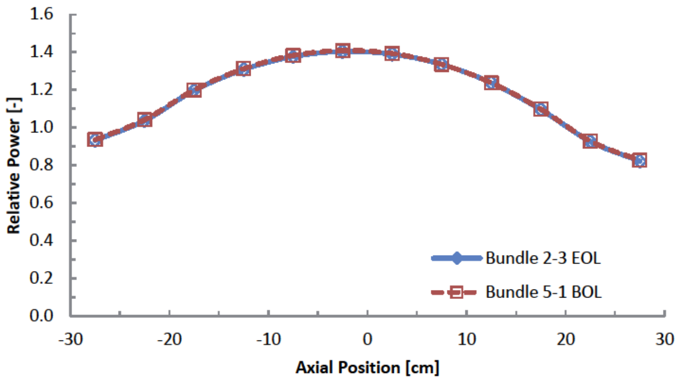
To analyze the effect of burnup on the DNBR analysis, a depletion calculation using the existing MCNP model was performed. The analysis demonstrated the effect of fuel depletion on the radial and axial power distributions in the core. To perform the required depletion calculations for the core, the BURN function in MCNP6 was used. Each fuel bundle isotopic composition was tracked independently with a unique material number identifier. The core was modeled at the licensed steady-state power limit of 100kW in different time steps until k_{eff} is within three standard deviations (±15 pcm) of a critical state (k_{eff} =1), or in other words, until there is little to no excess reactivity left.

The analysis for normal operation burn-up effects on the radial peaking factors for the UFTR show that, as expected, radial peaking decreases as a function of burn-up and that the most limiting condition is at BOL of the 22 bundle core (LCC) with a skewed flux profile. This value (5.44kW +/-3%) is consistent with the analysis in the docketed documents.

The table below shows the bundle powers at BOL and EOL for the UFTR. As the figure below shows, the effect on axial peaking due to burnup is minimal.

Table 1) Bundle Powers (+/-3%) for BOL and EOL at Limiting Critical Position

Bundle	BOL [kW]	EOL [kW]	% Difference
1-1	4.25	4.13	2.85
1-2	4.77	4.62	3.41
1-3	4.66	4.56	1.98
1-4	5.24	5.12	2.22
2-1	4.95	4.78	3.82
2-2	4.71	4.56	3.69
2-3	5.44	5.32	2.25
2-4	5.19	5.09	2.11
3-1	3.94	3.83	3.19
3-2	0	0	-
3-3	4.27	4.2	1.72
3-4	3.77	3.73	1.27
4-1	4.55	4.56	-0.61
4-2	5.09	5.11	-0.92
4-3	4.12	4.12	-0.70
4-4	4.60	4.6	-0.81
5-1	5.24	5.3	-1.73
5-2	4.84	5.04	-4.57
5-3	4.73	4.77	-1.51
5-4	4.37	4.51	-3.79
6-1	3.95	4.16	-5.50
6-2	3.60	3.7	-3.20
6-3	3.63	3.78	-4.74
6-4	0	0	-



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.

   The models and calculational methods are consistent throughout the analyses. The MCNP models used for neutronic analysis of the LCC were also used to provide the input parameters for the thermal hydraulic analyses. Revised SAR Chapters will be provided when completed.

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 worth's 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.

   Using RELAP5-3D (RELAP) an uncontrolled rod withdrawal (URW) analysis was performed for both protected and non-protected transients.

   The MCNP model for the LCC was used to calculate the reactor kinetics parameters, feedback coefficients, and power profiles used in the RELAP model. The table below lists a summary of the parameters for BOL and EOL used for the analysis.

Parameters for 22-bundle LCC		BOL			EOL		
β_{eff} (pcm)		741	±	10	739	±	10
l^* (μs)		198.5	±	0.1	203.4	±	0.1
α_{void} (pcm/%void)	(0 to 5% void)	-125	±	4	-94	±	4
	(5 to 10% void)	-140	±	4	-106	±	4
α_{water} (pcm/^{o}C)	(21 to 99^{o}C)	-6.7	±	0.3	-4.8	±	0.3
α_{fuel} (pcm/^{o}C)	(21 to 127^{o}C)	-1.9	±	0.2	-1.7	±	0.2
	(21 to 227^{o}C)	-1.7	±	0.1	-1.6	±	0.1

To adequately quantify the effects of a URW event, a bounding analysis was performed at BOL and EOL including initial conditions that encompass a wide range of operating regimes. To demonstrate the inherent safety of the design of the UFTR, no protective actions, i.e. trips, were assumed for the transients.

Tables 1 – 2 demonstrate a URW event in which the most reactive blade is withdrawn over a 100second interval. The highest peak power reached was for Case 9 (EOL, 1W, 30gpm, T_{in}=60F). In all cases the maximum Fuel Cladding temperature (~126C) was well below the 530C safety limit.

Tables 3 – 4 demonstrate a URW event with a 74 pcm/sec reactivity insertion rate. No protective actions were used. The highest peak power reached was for Case 30 (EOL, 1W, 50gpm, T_{in}=60F). In all cases the maximum Fuel Cladding temperature (~191C) was well below the 530C safety limit.

In all cases the reactor transients are terminated by the feedback mechanisms, i.e. fuel temperature and void coefficients of reactivity, inherent to the UFTR design before a safety limit is reached.

To demonstrate the effects of the reactor protection system (RPS) on these transients, two conservative trips were added. Tables 5 – 6 show the effects of 4 different reactivity insertion rates (2$, 1480pcm total) at the most limiting conditions at BOL and EOL from Tables 3 – 4. For cases other than the prompt reactivity insertion (0.5 sec), the transients are terminated by either the period or high flux reactor trips well below the 530C safety limit. The prompt reactivity insertion transients are terminated by the feedback mechanisms inherent to the UFTR design before the safety limit is reached. Both the period and high flux trips gravity dump the coolant moderator in addition to gravity dropping of the control blades. The effect on fuel temperature from gravity dumping of the coolant moderator is described by the LOCA analyses is SAR Section 13.2. The LOCA analyses show the upper bound on fuel temperature rise due to decay heat following a coolant dump is 14C. When a 14C increase is added to the RELAP5-3D results the temperatures still remain below the safety limit.

The tables are shown on the following pages.

Table 1 – 100 SEC 2.66$ insertion

Core	BOL – 22 bundle							
	1	**2**	**3**	**4**	**5**	**6**	**7**	**8**
P_o (kW) (at max decay heat)	1.00E-05	1.00E-05	1.00E-05	1.00E-05	100	100	100	100
Initial Condition	50 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=160^{o}F	50 gpm, T_{in}=160^{o}F	50 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=160^{o}F	50 gpm, T_{in}=160^{o}F
Blade Trip Setpoint (kW)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Blade Drop Time (s)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Time to Peak Power (s)	25	25.1	24.8	24.8	43	54.4	20.1	67.5
Peak Power (kW)	3088.9	3093.8	2500.7	2501.2	1778.1	798.6	439.3	598.7
$T_{fuel,max}$ at Peak Power (^{o}C)	118.9	119.3	116.6	116.7	112.2	109.6	106.5	107.1
$T_{fuel,max}$ (^{o}C)	123.4	121.5	126.2	125.0	112.9	110.6	106.6	107.3
$T_{clad,max}$ (^{o}C)	123.4	121.5	126.1	124.7	112.8	110.6	106.6	107.3
$T_{cool,max}$ (^{o}C)	110.4	111.3	114.9	115.4	108.6	107.6	104.4	103.7

Table 2 – 100 SEC 2.66$ insertion

Core	EOL – 22 bundle							
	9	**10**	**11**	**12**	**13**	**14**	**15**	**16**
P_o (kW) (at max decay heat)	1.00E-05	1.00E-05	1.00E-05	1.00E-05	100	100	100	100
Initial Condition	50 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=160^{o}F	50 gpm, T_{in}=160^{o}F	50 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=60^{o}F	30 gpm, T_{in}=160^{o}F	50 gpm, T_{in}=160^{o}F
Blade Trip Setpoint (kW)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Blade Drop Time (s)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Time to Peak Power (s)	25.1	25.1	24.9	24.9	43.1	53.6	20	67.7
Peak Power (kW)	3107.0	3097.7	2499.0	2506.4	1698.7	797.4	442.4	591.0
$T_{fuel,max}$ at Peak Power (^{o}C)	119.3	119.2	116.8	116.8	111.9	110.0	106.6	107.0
$T_{fuel,max}$ (^{o}C)	123.6	137.1	128.3	123.9	112.9	110.6	106.6	107.3
$T_{clad,max}$ (^{o}C)	123.5	137.0	128.3	123.8	112.8	110.6	106.5	107.3
$T_{cool,max}$ (^{o}C)	111.6	111.4	115.7	116.4	108.6	107.6	104.4	103.7

Table 3 – 74pcm/s insertion - 2$ Total

Core	BOL – 22 bundle							
	20	21	22	23	24	25	26	27
P_o (kW) (at max decay heat)	1.00E-05	1.00E-05	1.00E-05	1.00E-05	100	100	100	100
Initial Condition	50 gpm, T_{in}=60°F	30 gpm, T_{in}=60°F	30 gpm, T_{in}=160°F	50 gpm, T_{in}=160°F	50 gpm, T_{in}=60°F	30 gpm, T_{in}=60°F	30 gpm, T_{in}=160°F	50 gpm, T_{in}=160°F
Blade Trip Setpoint (kW)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Blade Drop Time (s)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Time to Peak Power (s)	11.49	11.5	11.42	11.41	10.57	10.59	10.02	9.98
Peak Power (kW)	9753.9	9780.1	8184.2	8155.2	1778.8	1657.0	963.2	1050.8
$T_{fuel,max}$ at Peak Power (°C)	135.0	135.1	128.8	128.6	113.9	113.4	114.1	114.9
$T_{fuel,max}$ (°C)	153.9	135.1	171.8	170.8	129.9	117.6	114.2	114.9
$T_{clad,max}$ (°C)	153.8	133.7	171.8	170.8	129.8	117.4	114.0	114.8
$T_{cool,max}$ (°C)	130.8	116.5	145.7	121.4	113.1	111.1	109.2	109.6

Table 4 – 74pcm/s insertion - 2$ Total

Core	EOL – 22 bundle							
	30	31	32	33	34	35	36	37
P_o (kW) (at max decay heat)	1.00E-05	1.00E-05	1.00E-05	1.00E-05	100	100	100	100
Initial Condition	50 gpm, T_{in}=60°F	30 gpm, T_{in}=60°F	30 gpm, T_{in}=160°F	50 gpm, T_{in}=160°F	50 gpm, T_{in}=60°F	30 gpm, T_{in}=60°F	30 gpm, T_{in}=160°F	50 gpm, T_{in}=160°F
Blade Trip Setpoint (kW)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Blade Drop Time (s)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Time to Peak Power (s)	11.53	11.53	11.45	11.45	10.59	27.09	10	10.03
Peak Power (kW)	9811.6	9805.6	8179.8	8181.2	1775.2	5286.4	967.2	1047.9
$T_{fuel,max}$ at Peak Power (°C)	135.1	135.1	128.6	128.8	113.8	121.4	114.1	114.8
$T_{fuel,max}$ (°C)	191.8	135.1	170.9	171.9	131.7	122.7	114.2	114.9
$T_{clad,max}$ (°C)	191.6	133.7	170.9	171.9	131.4	122.1	114.0	114.8
$T_{cool,max}$ (°C)	132.1	116.4	143.7	141.0	113.2	112.1	109.2	109.7

Core	Table 5 – BOL – 22 bundle							
	40	**41**	**42**	**43**	**44**	**45**	**46**	**47**
P_o (kW) (at max decay heat)	1.00E-05	1.00E-05	1.00E-05	1.00E-05	100	100	100	100
Initial Condition	30 gpm, $T_{in}=60^oF$	30 gpm, $T_{in}=60^oF$	30 gpm, $T_{in}=60^oF$	30 gpm, $T_{in}=60^oF$	30 gpm, $T_{in}=60^oF$	30 gpm, $T_{in}=60^oF$	30 gpm, $T_{in}=60^oF$	30 gpm, $T_{in}=60^oF$
Reactivity Insertion Rate (pcm/s)	37	74	148	Prompt	37	74	148	Prompt
Blade Trip Setpoint (kW)	120	120	120	120	120	120	120	120
Period Trip Setpoint (s)	1	1	1	1	1	1	1	1
Blade Drop Time (s)	3	3	3	3	3	3	3	3
Time to Peak Power (s)	14.71	8.12	4.85	0.837	5.79	4.46	3.78	0.535
Peak Power (kW)	0.096	0.059	0.153	169,481	164	235	677	152,232
$T_{fuel,max}$ at Peak Power (^oC)	15.6	15.6	15.6	345.4	36.0	38.0	51.6	332.6
$T_{fuel,max}$ (^oC)	15.6	15.6	15.6	493.8	36.0	38.1	52.1	487.1
$T_{clad,max}$ (^oC)	15.6	15.6	15.6	491.9	36.0	38.0	52.0	485.6
$T_{cool,max}$ (^oC)	15.6	15.6	15.6	108.7	24.5	24.1	26.8	109.8

Core	Table 6 – EOL – 22 bundle							
	50	**51**	**52**	**53**	**54**	**55**	**56**	**57**
P_o (kW) (at max decay heat)	1.00E-05	1.00E-05	1.00E-05	1.00E-05	100	100	100	100
Initial Condition	50 gpm, $T_{in}=60^oF$	50 gpm, $T_{in}=60^oF$	50 gpm, $T_{in}=60^oF$	50 gpm, $T_{in}=60^oF$	50 gpm, $T_{in}=60^oF$	50 gpm, $T_{in}=60^oF$	50 gpm, $T_{in}=60^oF$	50 gpm, $T_{in}=60^oF$
Reactivity Insertion Rate (pcm/s)	37	74	148	Prompt	37	74	148	Prompt
Blade Trip Setpoint (kW)	120	120	120	120	120	120	120	120
Period Trip Setpoint (s)	1	1	1	1	1	1	1	1
Blade Drop Time (s)	3	3	3	3	3	3	3	3
Time to Peak Power (s)	3.0	3.0	3.0	0.848	3.0	3.0	3.0	0.538
Peak Power (kW)	0.001	0.002	0.009	164,827	134	190	619	148,912
$T_{fuel,max}$ at Peak Power (^oC)	15.6	15.6	15.6	338.8	31.0	33.9	49.8	327.7
$T_{fuel,max}$ (^oC)	15.6	15.6	15.6	487.1	31.0	34.0	50.2	459.7
$T_{clad,max}$ (^oC)	15.6	15.6	15.6	485.6	31.0	34.0	50.1	457.9
$T_{cool,max}$ (^oC)	15.6	15.6	15.6	109.8	21.2	21.9	25.7	107.8

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 set points 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 set points used in the TSs.

   Provide a T-H analysis using the UFTR LSSS set points (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.

   The steady state thermal hydraulic analysis was performed using PLTEMP for the BOL 22-bundle core (LCC) with skewed power profile. The analysis used the same conservative assumptions from the UFTR Conversion SAR section 4.7.3.2 (i.e. decrease in the water channel spacing of 20mils) and then added further conservatism due to incorporation of the skewed critical blade height assumption and the 22-bundle core (versus banked blade heights and 22-bundle with 10-plate partial bundle).

   Multiple cases were run to optimize the LSSS parameters for operating margin and human factoring for analog power indication (i.e. it's easier to identify analog indication of power exactly at 110% than at 119%). For the same true flow of 39 gpm, this optimization results in a reduction of true power to 116 kW and an increase in true inlet temperature to 103.1F at an ONBR of 1.003. Therefore, the new proposed LSSS values are 110 kW, 102F, and 41 gpm (unchanged).

   The nominal operating conditions for the core are listed in the table below.

Nominal Operating Conditions for the UFTR Core

Nominal Condition	
Inlet Temperature, (^{o}C)	27.5 (81^{o}F)
Inlet mass flow rate, (gpm)	46
Power, (kW)	100

The PLTEMP T-H parameters for the LCC at nominal operating conditions are listed in the table below.

Thermal-hydraulics Parameters of the LCC at Nominal Operating Conditions

Parameter	
Max. Fuel Temperature, (^{o}C)	73.6
Max. Clad Temperature, (^{o}C)	73.5
Max. Coolant Channel, outlet temperature, (^{o}C)	71.5
Min. ONBR	1.540
Min. DNBR	463

The operating conditions at LSSS conditions are listed in the table below.

LSSS Operating Conditions for the UFTR Core

LSSS Condition	
Inlet Temperature, (^{o}C)	39.5 (103.1 ^{o}F)
Inlet mass flow rate, (gpm)	39
Power, (kW)	116

The PLTEMP T-H parameters for the LCC at LSSS operating conditions are listed in the table below.

Thermal-hydraulics Parameters of the LCC at LSSS Operating Conditions

Parameter	
Max. Fuel Temperature, (^{o}C)	98.7
Max. Clad Temperature, (^{o}C)	98.6
Max. Coolant Channel, outlet temperature, (^{o}C)	96.3
Min. ONBR	1.003
Min. DNBR	280

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 7.3.1states, "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.

   New reactivity insertion analyses (URW) have been performed including cases with and without trips. Refer to RAI #4 above for more details. Chapter 13 of the SAR will be revised to reflect the new analyses. Additionally, SAR Section 7.3.1 will be revised to state:

   The UFTR facility is provided with two types of reactor trips. These reactor trips are classified into two categories:

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

   The following conditions will initiate a Full-trip when two or more control blades are not at their bottom position;

   - Short Period (3 seconds or less);
   - High Power (110%);
   - Reduction of high voltage to the neutron chambers of 10% or more;
   - Turning off the console magnet power switch;
   - A.C. power failure.

The following conditions will initiate a Blade-trip:

- Loss of power to Stack Dilution fan;
- Loss of power to Core Vent fan/damper;
- Loss of power to the deep well pump when operating at or above 1 kW and using deep well for secondary cooling;
- Secondary flow below 60 gpm when operating at or above 1 kW using the well water system for secondary cooling (10 sec delay);
- Secondary flow below 8 gpm when operating at or above 1 kW using city water for secondary cooling (no delay after initial 10 second time interval);
- Shield tank water level 6" below established normal level;
- Loss of power to primary coolant pump;
- Primary coolant flow below 41 gpm (inlet flowrate);
- Loss of primary coolant flow (no return flow);
- Primary coolant level below 42.5";
- Any primary coolant return temperature above 155°F;
- Primary coolant inlet temperature above 102°F;
- Initiation of the evacuation alarm;
- Manual reactor trip button depressed.

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 set points, by:

(i) identifying the trips excluded from the TSs,

Please refer to the RAI #2g response in the July 2016 set of Technical Specification related RAIs and RAI#6a above.

(ii) explain whether that trip will still be utilized in the operation of UFTR if the requested change is approved, and

Please refer to RAI #6a above for current trip functions of the Reactor Protection System. The UFTR has no current plans to eliminate or disable RPS trip functions.

(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 set points. Alternately, justify why no additional information is needed.

Please refer to the RAI #2g response in the July 2016 set of Technical Specification related RAIs and RAI#6a above.

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,

The guidance in NUREG-1537, Part 1, Appendix 14.1, Section 3.2, item (5), actually states that "Required interlocks that inhibit or prevent control rod withdrawal or reactor startup should be specified by a table." By keeping allowable excess reactivity low, the UFTR is able to demonstrate that postulated reactivity insertion events require no automatic control or safety functions.

Because context is also important, it should be noted that in addition to the SAR Section 3.1 statement quoted in the RAI introduction, SAR Section 3.1 states in part that:

"The UFTR principal physical barrier to fission product release is the fuel cladding. Because of the fuel material and core design, the fuel and moderator temperature reactivity coefficients are negative assuring inherent protection. Safe reactor operation is guaranteed by this inherently safe reactor design and by limiting the installed excess reactivity. Calculations presented in Chapters 4 and 13 demonstrate that the safety limit on the temperature of the fuel will not be exceeded and that residual heat removal is not necessary even under loss of coolant moderator."

"The scenarios analyzed in Chapter 13 conservatively demonstrate that instrumented shutdown actions and building confinement are not necessary to ensure that radiological doses will not exceed 10 CFR Part 20 allowable limits."

"No control or safety system is required to maintain a safe shutdown condition."

The SAR Sections 7.1.3.1.1, 7.1.3.1.2, 7.2.2, and 9.1.2 referenced in the RAI introduction actually describe the existence of, and functions of, these interlocks as opposed to their importance to safety. Additionally, SAR Sections 7.1.3.1.1 and 7.1.3.1.2, which describe Reactor Power Channels 1 and 2 respectively, clearly state that blade withdrawal interlocks are "…activated for specific conditions impacting channel operability". The UFTR has not proposed eliminating the Technical Specification requirement for Reactor Power Channel operability.

Additionally, contrary to the RAI introduction, the Reactor Cell Evacuation Alarm interlock function required by proposed TS 3.4 is discussed in SAR Section 7.6.

Since the time of the previous license renewal, well prior to the advent of ANSI/ANS-15.1 and NUREG-1537, the UFTR Technical Specifications have effectively been a cut-n-paste of all RCS/RPS functions described in the SAR. They exist in the Technical Specifications simply because the equipment functions as installed and described in the SAR at that time were copied over into the Technical Specifications.

Since then the Commission and staff have stated the intent of license renewal now is to ensure a single, complete, accurate, and contemporary licensing basis meeting current requirements and guidance. The UFTR has made every feasible effort to meet the requirements and expectations including re-performance of the core and accident analyses and a complete reconstitution of all licensing basis documents derived from those analyses including the SAR, Emergency Plan, and the Technical Specifications.

By keeping allowable excess reactivity low, the UFTR is able to demonstrate that postulated reactivity insertion events require no automatic control or safety functions to prevent reaching the Safety Limit. This conclusion is consistent with the NRC's own findings for low power RTRs like the UFTR (Refs: SECY-15-0081 and NUREG-2150).

The UFTR has not proposed eliminating the Technical Specification requirement for Reactor Power Channel operability. The reactor power measuring channels are required to be verified operable by SR under TS 3.2.3(1). The associated surveillance performance includes verification that the control blade and startup inhibits described in the SAR function as designed to support operability of the reactor power measuring channels. Additionally, the core vent and stack dilute systems are required to be verified operable as specified by SR 3.5. Core vent system functions described in the SAR are

also verified by surveillance. Surveillances to demonstrate LCO compliance are performed in accordance with approved written procedures as required under TS 6.4(4).

Additionally, the proposed fast period trip in TS 3.2.2 is specifically included to ensure early termination of a reactivity insertion event originating from low power and the high power trip in TS 3.2.2 will terminate reactivity insertion events that begin closer to rated power.

Therefore, the requirements of 10 CFR 50.36 as well as the guidance of ANSI/ANS 15.1-2007 Sections 3.2(6) & 4.2(5) and NUREG-1537, Part 1, Appendix 14.1, Section 3.2, are met.

(ii) explains whether that interlock will still be utilized in the operation of UFTR if the requested change is approved, and

The UFTR has no current plans to eliminate or disable interlocks with the exception of the evacuation alarm interlock with the core vent system (current SR 4.2.3(2) and DF 5.7.3) as previously discussed with the staff. Disabling the evacuation alarm interlock with the core vent system ensures confinement pressure remains negative so in the event of potential radioactive gas accumulation any release from confinement continues to be routed through the stack where it is filtered, monitored, diluted and elevated to minimize any potential dose to the public.

(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.

See responses to RAIs #6ci and 6cii above.

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 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.

      This is an editorial error that will be corrected in the revised SAR Chapter 13. The new value is 5.44 kW.

      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.

      The UFTR calculated an activity release fraction for a break widthwise across the assembly (fuel meat of 5.96 cm). The NRC staff value indicates an assumption the break is lengthwise along the assembly (fuel length of 60 cm).

      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.

The COMPLY code was chosen because it provides a reasonable method for estimating low dose (non-accident) public exposures from airborne emissions from facilities like the UFTR. This is demonstrated by NRC endorsement of its use for comparable purposes in RG-4.20, Constraint On Releases Of Airborne Radioactive Materials To The Environment For Licensees Other Than Power Reactors.

As discussed with staff, the UFTR re-performed the Level 3 COMPLY calculation using the revised ORIGEN results and the following assumptions:

1. Locations outside the reactor building are exposed to a fractional release of the total iodines due to plating out of iodines inside the reactor cell and reactor ventilation components. The portion available for release outside of the reactor cell is conservatively assumed to be 25%.
2. The most exposed member of the public is conservatively assumed to continuously occupy a location 10 meters away from the UFTR stack outside of a building or shelter of any type. This is the approximate ground level distance from the base of the stack to the closest walking path which is located in the unrestricted area just east of the reactor building.
3. The most exposed member of the public is conservatively assumed to get all their meat, milk, and vegetables from on-campus gardens and farms.
4. Following the accident, no credit is taken for radiological decay during holdup in the reactor cell prior to release through the stack.
5. At Level 3 the COMPLY code assumes the wind is in the direction of the receptor 25% of the year. To account for a shorter duration release where the wind is assumed to be in the direction of the receptor 100% of the time, the COMPLY results will be multiplied by 4.

Using the COMPLY computer code the maximum postulated TEDE received by the most exposed member of the general public is estimated. The results of calculating this annual TEDE for the MHA and FHA are given in the following table.

	TEDE (mrem/year)	
Most Exposed Location	MHA	FHA
10 meters from reactor stack	24.0	0.4

Exposures in both scenarios are significantly less than the annual public dose limit of 100 mrem and show that the UFTR has no significant impact on the health and safety of the public even under the worst postulated events.

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.

This is an editorial error (13.2.5.1 should have been 13.2.1.1) that will be corrected in the revised SAR Chapter 13.

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 ofI-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.

Between 3 and 4 days following shutdown the I-132 inventory becomes less than I-131. The I-132 inventory following shutdown is a combination of the undecayed I-132 (t1/2=2.295 hours) produced from the fission process and the I-132 produced by the decay of Te-132 (t1/2=3.204 days). So, in less than a day following shutdown, the only significant source of I-132 becomes the decay of Te-132. Therefore, the inventory of I-132 follows the half-life of Te-132 in the days following shutdown.

The ORIGEN input and output files (including un-decayed inventories) with explanation of the decay chain were provided via email.

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 Table13-1 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.

The TEDE (or Thyroid dose) is defined as the sum of the external dose (DDE) and internal dose (CEDE or CDE). The deep dose equivalent (DDE) for personnel in the reactor cell for a given stay-time for both the thyroid and the whole body is calculated by:

$$DDE_{thy} \text{ or } DDE_{eff} = \sum_i \left[\frac{DCF_{AS,i}\, C_i\, [1 - exp(-\lambda_i t_{ST})]}{\lambda_i} \right]$$

where:

$DCF_{AS,i}$ = dose coefficient for air submersion for isotope *i* tabulated in Federal Guidance Report No. 12 (rem-m^3/Ci-s)

C_i = initial concentration of isotope *i* released into reactor cell (Ci/m^3)

λ_i = decay constant for isotope *i* (s^{-1})

t_{ST} = stay-time of personnel (s).

The CDE and CEDE for personnel in the reactor cell for a given stay-time is calculated by:

$$CDE_{thy} \text{ or } CEDE = \sum_i \left[\frac{DCF_{INH,i}\, C_i\, BR\, [1 - exp(-\lambda_i t_{ST})]}{\lambda_i} \right]$$

where:

$DCF_{INH,i}$ = dose coefficient for inhalation of isotope *i* tabulated in Federal Guidance Report No. 11 (rem/Ci)

BR = breathing rate (3.33E-04m^3/s).

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.

Please refer to the RAI #4b response in the July 2016 set of Technical Specification related RAIs. The new value chosen assumes the reactor cell is effectively solid up to about 10' above floor level so the simplified minimum free volume becomes 30'x60'x20' = 36,000 cubic feet.

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 a625 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.

A copy has been provided by email.

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.

Approximate times are 12 seconds for full-trip (PC pump off and dump valve open) and 75 seconds for PC pump off only.

The LOCA analyses show the upper bound on fuel temperature rise due to decay heat following a coolant dump is 14C. When a 14C increase is added to the RELAP5-3D results the temperatures still remain below the safety limit. See RAI#4 response above.

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 coolant dump valve opening or rupture disk breaking.

See RAI #6a response for full-trip (automatic dump valve open) conditions. The graphite rupture disk is designed to break for any condition that results in approximately 2 psi above the normal operating system pressure as described in SAR Section 5.2.

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 is11.14 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 UFTRSAR calculation was 40,000 psi.

TS 3.8.2 has been revised to eliminate consideration of explosives irradiation (i.e. no irradiation of explosives will be allowed by TS restriction). The corresponding analyses will be removed from the SAR.

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

TS 3.8.2 has been revised to eliminate consideration of explosives irradiation (i.e. no irradiation of explosives will be allowed by TS restriction). The corresponding analyses will be removed from the SAR.

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.

TS 3.8.3 has been revised to include a radioiodine Curie limit to limit fissile material to that which would result in a radioiodine activity equivalent to that released in the FHA analyses.

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 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

The current SNM license limit of "…up to 0.2 kilograms of contained uranium-235 of any enrichment in the form of fission chambers, flux foils and other forms, all used in connection with operation of the reactor" is sufficient.

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 and50% 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.*

   A copy of SAR Reference 13.16 explaining the applicability of the Tong correlation was provided by email.

Appendix A to Facility License No. R-56

Technical Specifications and Bases

University of Florida Training Reactor

Docket No. 50-083

October 31, 2016

TECHNICAL SPECIFICATIONS

1.0 Introduction

1.1 Scope

This document constitutes the Technical Specifications for Facility License No. R-56 as required by 10 CFR 50.36 and supersedes all prior UFTR Technical Specifications. This document includes the "bases" to support the selection and significance of the specifications. Each basis is included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.2 Definitions

CHANNEL: A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

CHANNEL CALIBRATION: Channel calibration shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel measures. The channel calibration shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL TEST.

CHANNEL CHECK: Channel check shall be the qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel indication and status with other independent channels measuring the same parameter.

CHANNEL TEST: A channel test shall be:

a. Analog and bistable channels - the introduction of a signal into the channel for verification that it is OPERABLE.
b. Digital computer channels – the use of diagnostic programs to test digital computer hardware and the introduction of simulated process data into the channel for verification that it is OPERABLE.

CORE ALTERATION: Core alteration shall be the movement of any reactor fuel assemblies, graphite moderator elements, experimental facilities, or control blade assemblies within the reactor core region in MODE 5.

CORE CONFIGURATION: Core configuration shall include the number, type, or arrangement of fuel assemblies, graphite moderator elements, experimental locations, and control blades occupying the core region.

DAMAGED FUEL: A fuel element shall be identified as damaged if the cladding is breached resulting in fission product release or if visual inspection of the fuel indicates cladding blistering, excessive swelling, excessive bulging, excessive deformation, cladding holes, cladding tears, or cladding breaches of any kind.

EXCESS REACTIVITY: Excess reactivity shall be that amount of reactivity that would exist if all control blades were moved to the maximum reactive condition from the point where the reactor is exactly critical (keff = 1). When calculating excess reactivity, no credit shall be taken for negative experiment worth, temperature effects or xenon poisoning.

EXPERIMENT: Any evolution, hardware, or target (excluding devices such as detectors or foils) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within an irradiation facility. Hardware rigidly secured to the core or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.

FUEL DEFECT: A fuel defect shall be any unintended change in the physical as-built condition of the fuel with the exception of normal effects of irradiation (e.g. elongation due to irradiation growth or assembly bow). Examples include unusual pitting, unusual bulging, missing or broken bolts, missing or broken spacers, missing or broken combs, missing or broken welds, or unusual corrosion.

MOVABLE EXPERIMENT: A movable experiment is one where it is intended that all or part of the experiment may be moved into or adjoining the core or into and out of the core while the reactor is in MODES 1 or 2.

OPERABLE - OPERABILITY: A system or component shall be operable or have operability when it is capable of performing its intended function.

RATED THERMAL POWER (RTP): RTP shall be a total reactor core heat transfer rate to the reactor coolant of 100 kWt.

REACTIVITY WORTH OF AN EXPERIMENT: The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

REACTOR CELL: The Reactor Cell is the confinement enclosure around the reactor structure that is designed to limit the release of effluents between the enclosure and its external environment through defined pathways.

REACTOR OPERATING: The reactor is operating whenever it is not in MODES 3, 4, 5 or defueled. Reactor operation at greater than or equal to 1% RTP shall be called MODE 1. Reactor operation at less than 1% RTP shall be called MODE 2.

REACTOR SHUTDOWN: The reactor is shutdown if it is subcritical by at least 760 pcm with the core at ambient temperature with the reactivity worth of xenon equal to zero and with the reactivity worth of all installed experiments included. The reactor shutdown condition shall be called MODE 3.

REACTOR SECURED: The reactor is secured when with fuel present in the reactor there is insufficient water moderator available in the reactor to attain a keff greater than 0.8 or there is insufficient fuel present in the reactor under optimum available conditions of moderation and reflection to attain a keff greater than 0.8 or the reactor is shutdown with all control blades fully inserted; and the following conditions exist:

a. the console key switch is in the OFF position and the key is removed from the switch; and
b. no work is in progress involving fuel, core structure, installed control blades, or control blade drives unless they are physically decoupled from the control blades; and
c. no experiments are being moved or serviced that have, on movement, a reactivity worth exceeding 720 pcm.

The reactor secured condition shall be called MODE 4.

REACTOR OUTAGE: The reactor is in an outage condition anytime less than two layers of concrete block shielding are fully installed over the top of the core area with fuel in the core. The reactor outage condition shall be called MODE 5.

SHALL, SHOULD, and MAY: The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

SHUTDOWN MARGIN: Shutdown margin is the minimum shutdown reactivity necessary to ensure the reactor can be made subcritical by means of the reactor control and trip systems starting from any permissible operating condition with the most reactive blade in its most reactive position and that the reactor will remain subcritical without further operator action. When calculating shutdown margin, no credit shall be taken for negative experiment worth, temperature effects or xenon poisoning.

STRUCTURE, SYSTEM, OR COMPONENT (SSC): A structure is an element, or a collection of elements, to provide support or enclosure, such as a building, free-standing tanks, basins, dikes, or stacks. A system is a collection of components assembled to perform a function, such as piping, cable trays, conduits, or ventilation. A component is an item of mechanical or electrical equipment, such as a pump, valve, or relay, or an element of a larger array, such as a length of pipe, elbow, or reducer.

UNSCHEDULED SHUTDOWN: An unscheduled shutdown is any unplanned shutdown of the reactor caused by actuation of the reactor trip system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

1.3 Surveillance Intervals

The specified frequency for each Surveillance Requirement (SR) or audit is met if performed within the interval specified, as measured from the previous performance or as measured from the time a specified condition of the frequency is met. The permitted intervals are established as follows:

a. 10 years – interval not to exceed 12 years

b. 5 years – interval not to exceed 6 years

c. Biennial – interval not to exceed 30 months

d. Annual – interval not to exceed 15 months

e. Semiannual – interval not to exceed 7.5 months

f. Quarterly – interval not to exceed 4 months

g. Monthly – interval not to exceed 6 weeks

h. Weekly – interval not to exceed 10 days

i. Daily – interval not to exceed 24 hours

2.0 Safety Limit and Limiting Safety System Settings

2.1 Safety Limit

Applicability: MODES 1 and 2.

Objective: To ensure fuel cladding integrity.

Specification: The fuel and cladding temperatures shall not exceed 986°F (530°C).

Basis: The safety limit is based on measurement of first fission product release from the fuel at or above the blister threshold temperature described in NUREG-1313.

2.2 Limiting Safety System Settings

Applicability: MODES 1 and 2.

Objective: To ensure automatic action terminates the abnormal situation before the safety limit is challenged.

Specification: According to Table 2.2-1.

Basis: Due to the inherently safe core design and low EXCESS REACTIVITY, postulated reactivity insertion event analyses indicate no automatic control or safety functions are needed to prevent reaching the Safety Limit (Ref. SAR Section 13.2). Therefore, to allow for generation of a reasonable set of Technical Specifications, and provide defense-in-depth, the fundamental reactor parameters of power, temperature, and flow were conservatively chosen for incorporation as LSSSs. These very conservative settings ensure normal reactor operation remains within the assumptions of the thermal hydraulic analysis for normal operation (ONBR > 1) as described in SAR Section 4.6.

Table 2.2-1
Limiting Safety System Settings

	FUNCTION	ALLOWABLE VALUE
1.	High Reactor Power Trip	≤ 110% RTP
2.	Low Reactor Coolant Flow Trip	≥ 41 gpm
3.	High Average Reactor Coolant Inlet Temperature Trip	≤ 102°F

3.0 Limiting Conditions for Operation and Surveillance Requirements

3.0.1 LCO Applicability

Applicability: Any MODE or specified condition in which the applicable SSC is required to be OPERABLE.

Objective: To ensure timely operator action in the event a SSC is discovered to be inoperable during a MODE or other specified condition in which the SSC is required to be OPERABLE.

Specification:

1. When any of the following LCOs are not met the reactor shall be placed in a MODE or other condition in which the LCO is not applicable. Action shall be initiated within 15 minutes of discovery of failure to meet the LCO. Where corrective measures are completed that permit operation in accordance with the LCO, completion of the actions required by LCO 3.0.1 is not required:
   a. LCO 3.2.1
   b. LCO 3.2.2
   c. LCO 3.2.3
   d. LCO 3.3.1
   e. LCO 3.3.2
   f. LCO 3.4
   g. LCO 3.5
   h. LCO 3.7.1
   i. LCO 3.9.1

2. Suspension of CORE ALTERATIONS, irradiated fuel movement, or irradiated fueled EXPERIMENT movement, shall not preclude completion of movement of an irradiated component to a safe position.

Basis: LCO 3.0.1(1) provides the operator with guidance and an allowed action time upon discovery that the specified LCO is not being met. The 15-minute time limit ensures sufficient time is available to initiate appropriate action while limiting the duration of the LCO outage. LCO 3.0.1(2) provides the operator with prioritization guidance and an exception to the 15-minute allowed action time to allow for safe completion of an irradiated component movement already in-progress.

3.0.2 Surveillance Requirement Applicability

Applicability: Any MODE or specified condition in which the applicable SSC or variable is required to be OPERABLE or within specified limits.

Objective: To confirm SSCs and variable properties required by Technical Specifications are OPERABLE and within specified limits.

Specification:

1. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the associated LCO. Failure to perform a surveillance within the specified frequency shall be failure to meet the LCO except as provided in TS 3.0.2 (2) and TS 3.0.2 (3).

2. SRs may be deferred during MODES or other specified conditions in which a SSC or variable is not required to be OPERABLE or within specified limits; however, they shall be completed prior to entry into a MODE or other specified condition in which the SSC or variable is required to be OPERABLE or within specified limits unless entry into the MODE or other specified condition is required for performance of the surveillance as provided in TS 3.0.2 (3).

3. The following SRs require entry into the applicable MODE or other specified condition for performance of the surveillance. These SRs shall be performed as soon as practicable after entry into the MODE or other specified condition required for performance of the surveillance:
   a. SR 3.1.1
   b. SR 3.1.2
   c. SR 3.2.3.3 for LCO 3.2.3(1)
   d. SR 3.7.2.2
   e. SR 3.7.2.3

4. Appropriate surveillance testing on any Technical Specification required SSC shall be conducted after replacement, repair, or modification before the SSC is considered OPERABLE except as provided in TS 3.0.2 (3).

Basis: These LCOs provide the operator with guidance and restrictions regarding missed SRs, deferred SRs, and post-maintenance testing of Technical Specification required SSCs.

3.1 Reactor Core Reactivity Parameters

Applicability: MODES 1 through 5.

Objective: To ensure the reactor can be made subcritical and to ensure the safety limit shall not be exceeded.

Specification: According to Table 3.1-1.

Basis: The value of SHUTDOWN MARGIN assures the reactor can be made subcritical from any operating condition. The value of EXCESS REACTIVITY allows flexibility to operate the reactor without the need to add fuel on a frequent basis while maintaining the installed core EXCESS REACTIVITY within the bounds of the analysis described in SAR Section 13.2.

Table 3.1-1
Reactor Core Reactivity Parameters

	REACTIVITY PARAMETER	ALLOWABLE VALUE
1.	SHUTDOWN MARGIN	$\geq$ 760 pcm
2.	EXCESS REACTIVITY	$\leq$ 1480 pcm

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.1	Verify SHUTDOWN MARGIN within limits	Annual [(a)]
SR 3.1.2	Verify EXCESS REACTIVITY within limits	Annual [(a)]

(a) These reactivity parameters shall also be verified within limits following changes in CORE CONFIGURATION.

3.2 Reactor Control and Trip Systems

3.2.1 Control Blades

Applicability: MODES 1 and 2.

Objective: To ensure the reactor can be shut down promptly when a trip signal is initiated.

Specification: Individual control blade drop times as measured from the fully withdrawn position for each of the four control blades shall not exceed 2.0 seconds from initiation of blade drop to full insertion.

Basis: This specification ensures that the reactor will be promptly shut down when a trip signal is initiated. The reactivity insertion analyses provided in SAR Section 13.2 demonstrate the acceptability of the control blade drop time.

SURVEILLANCE REQUIREMENT

	SURVEILLANCE	FREQUENCY
SR 3.2.1	Verify each control blade drop time is within limits	Annual

3.2.2 Reactor Trips

Applicability: MODES 1 and 2.

Objective: To specify the minimum required OPERABLE reactor trips.

Specification: According to Table 3.2.2-1.

Basis: LCOs 3.2.2(1), 3.2.2(2), and 3.3.3(3) ensure reactor operation remains bounded by the thermal hydraulic analysis described in SAR Section 4.6. LCO 3.2.2(4) ensures early termination of a reactivity insertion event originating from low power levels. LCO 3.2.2(5) provides redundancy to LCO 3.2.2(2) and acts as a blade withdrawal inhibit until the minimum core water level is reached. The Manual trip allows the operator to quickly shutdown the reactor if an unsafe or abnormal situation occurs.

Table 3.2.2-1
Specifications for Reactor System Trips

	FUNCTION	SURVEILLANCE REQUIREMENTS	ALLOWABLE CONDITION OR VALUE
1.	High Reactor Power	SR 3.2.2.1	≤ 110% RTP
2.	Low Reactor Coolant Flow	SR 3.2.2.2	≥ 41 gpm
3.	High Reactor Coolant Inlet Temperature	SR 3.2.2.1	≤ 102°F
4.	Fast Reactor Period	SR 3.2.2.1	≥ 3 seconds
5.	Low Reactor Coolant Level	SR 3.2.2.2	≥ 2 inches above the fuel
6.	Manual	SR 3.2.2.1	OPERABLE

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.2.1	Perform a CHANNEL CHECK	Daily
SR 3.2.2.2	Perform a CHANNEL TEST	Quarterly

3.2.3 Reactor Measuring Channels

Applicability: MODES 1 and 2.

Objective: To specify the minimum measuring channels required to be OPERABLE.

Specification: According to Table 3.2.3-1.

Basis: To ensure indications of the specified parameters are provided to the operator for adequate monitoring of steady state and transient reactor conditions.

Table 3.2.3-1
Minimum Required Measuring Channels

	CHANNEL	SURVEILLANCE REQUIREMENTS	NUMBER OPERABLE
1.	Reactor Power	SR 3.2.3.2 and SR 3.2.3.3	2
2.	Reactor Period	SR 3.2.3.2 and SR 3.2.3.3	1
3.	Control Blade Position	SR 3.2.3.1	4
4.	Reactor Coolant Flow	SR 3.2.3.1 and SR 3.2.3.3	1
5.	Average Reactor Coolant Inlet Temperature	SR 3.2.3.3	1
6.	Average Reactor Coolant Outlet Temperature	SR 3.2.3.3	1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.3.1	Perform a CHANNEL CHECK	Weekly
SR 3.2.3.2	Perform a CHANNEL TEST	Daily
SR 3.2.3.2	Perform a CHANNEL CALIBRATION	Annual

3.3 Coolant Systems

3.3.1 Leak Detection

Applicability:	MODES 1 and 2.
Objective:	To ensure remote indication of water leakage into the equipment pit.
Specification:	The equipment pit water level sensor shall provide an alarm if water level in the equipment pit is greater than 1 inch above equipment pit floor level.
Basis:	This specification is designed to alert the operator of water leakage into the equipment pit. The setpoint of one inch is based on the design of the equipment pit alarm level sensor.

SURVEILLANCE REQUIREMENT

	SURVEILLANCE	FREQUENCY
SR 3.3.1	Perform a CHANNEL TEST	Weekly

3.3.2 Reactor Coolant System Water

Applicability: When Reactor Coolant System water is in contact with fuel assemblies.

Objective: To specify the electrical resistivity limit for reactor coolant system water in contact with in-core fuel assemblies.

Specification: The electrical resistivity of reactor coolant system water shall be no less than 0.5 MΩ-cm [a].

Basis: The resistivity limit is designed to minimize fuel assembly corrosion. Monitoring reactor coolant resistivity provides for early indication of any potential fission product release.

SURVEILLANCE REQUIREMENT

	SURVEILLANCE	FREQUENCY
SR 3.3.2	Verify resistivity is within the limit	Daily

(a) Normal transients and experiments can cause Reactor Coolant System water electrical resistivity to drop below 0.5 MΩ-cm for short periods of time. For these expected occurrences, reactor operations with electrical resistivity less than 0.5 MΩ-cm may continue for periods not to exceed 4 hours provided that continuous control room indication of reactor coolant resistivity is utilized and trended during that period.

3.4 Reactor Cell Evacuation Alarm Interlock

Applicability:	MODES 1 and 2; during CORE ALTERATIONS, irradiated fuel movement, and irradiated fueled EXPERIMENT movement.
Objective:	Specify requirements for this evacuation alarm system interlock.
Specification:	Two area radiation monitors simultaneously alarming high shall cause an automatic actuation of the evacuation alarm.
Basis:	As described in SAR Chapter 7, the evacuation alarm interlock with the area monitor high alarm function is designed to alert the staff and occupants of potential radiological emergencies including potential fission product release into the REACTOR CELL.

SURVEILLANCE REQUIREMENT

	SURVEILLANCE	FREQUENCY
SR 3.4	Verify proper interlock function	Weekly

3.5 Reactor Cell Ventilation Systems

Applicability: MODES 1 and 2; during CORE ALTERATIONS, irradiated fuel movement, and irradiated fueled EXPERIMENT movement.

Objective: To specify the minimum OPERABILITY requirement for the REACTOR CELL ventilation systems.

Specification: The core vent and stack dilution systems shall be operating and maintaining REACTOR CELL pressure negative with respect to the surrounding environment.

Basis: As described in SAR Chapters 9 and 11, operation of the core vent system ensures REACTOR CELL pressure is maintained negative relative to the surrounding environment and potential gaseous effluents are routed to the reactor stack. Operation of the stack dilution system ensures that gaseous effluents originating from the REACTOR CELL are diluted prior to release.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.1	Verify core vent and stack dilution systems are operating	Daily
SR 3.5.2	Verify REACTOR CELL pressure is negative with respect to the surrounding environment	Quarterly

3.6 Emergency Power – This section intentionally blank

3.7 Radiation Monitoring Systems and Radioactive Effluents

3.7.1 Radiation Monitoring Systems

Applicability: MODES 1 and 2; During CORE ALTERATIONS, irradiated fuel movement, and irradiated fueled EXPERIMENT movement.

Objective: To specify minimum OPERABILITY requirements for the area radiation monitors, air particulate detector, and stack radiation monitor.

Specification: According to Table 3.7.1-1.

Basis: As described in SAR Chapter 7, the radiation monitoring channels inform the operator about the radiological conditions present in the REACTOR CELL and reactor stack and provide early detection of any potential fission product release or radiological abnormality.

Table 3.7.1-1
Minimum Radiation System Requirements

	MONITOR TYPE	SURVEILLANCE REQUIREMENTS	NUMBER REQUIRED OPERABLE [(a)]
1.	Area Radiation Monitor	SR 3.7.1.1, SR 3.7.1.2, and SR 3.7.1.3	3
2.	Air Particulate Detector	SR 3.7.1.1, SR 3.7.1.2, and SR 3.7.1.3	1
3.	Stack Radiation Monitor	SR 3.7.1.1, SR 3.7.1.2, and SR 3.7.1.3	1

(a) When any single required radiation monitoring channel becomes inoperable, portable instruments, surveys, or analysis may be substituted within one hour of discovery for periods not to exceed one week. Maintenance and surveillance interruptions for periods of one hour or less are permissible.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.1.1	Perform a CHANNEL CHECK	Daily
SR 3.7.1.2	Perform a CHANNEL TEST	Weekly
SR 3.7.1.3	Perform a CHANNEL CALIBRATION	Semiannual

3.7.2 Argon-41 Discharge

Applicability: MODES 1 and 2.

Objective: To ensure Argon-41 emissions resulting from licensed UFTR operation remain below applicable limits.

Specification:

1. Ar-41 emissions resulting from licensed UFTR operation shall not exceed the total effective dose limit of 10 CFR 20.1101(d).

2. Energy generation (kW- hours) of the UFTR shall be limited to ensure TS 3.7.2(1) is not exceeded.

Basis: Regulation 10 CFR 20.1101(d) imposes an ALARA constraint of 10 mrem per year total effective dose equivalent on airborne emissions of radioactive material to the environment. To ensure compliance with this annual constraint, the UFTR limits Ar-41 produced by administratively limiting energy generation as described in SAR Chapter 11.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	Verify UFTR energy generation is within the limit	Quarterly
SR 3.7.2.2	Verify the expected total effective dose equivalent to the individual member of the public likely to receive the highest dose from Ar-41 emission is within the limit of 10 CFR 20.1101(d)	Semiannual
SR 3.7.3.2	Determine the UFTR energy generation limit based on measurement of the stack effluent discharge	Semiannual

3.8 Limitations on Experiments

3.8.1 Experiment Reactivity Limits

Applicability: MODES 1 and 2.

Objective: To minimize the likelihood of an inadvertent prompt reactivity excursion and to prevent damage to the fuel and cladding.

Specification:

1. The absolute value of the reactivity worth of any single MOVABLE EXPERIMENT shall be less than or equal to 720 pcm.

2. The sum of the absolute values of the reactivity worths of all EXPERIMENTS shall be less than or equal to l400 pcm.

Basis: The reactivity limit on MOVABLE EXPERIMENTS is less than the effective delayed neutron fraction to prevent an inadvertent prompt reactivity excursion. The total reactivity worth limit is established to prevent a reactivity insertion larger than the stipulated maximum step reactivity insertion in the accident analysis (Ref. SAR Sections 4.1 and 13.2).

3.8.2 Experiment Materials and Malfunctions

Applicability: MODES 1 and 2.

Objective: To prevent damage to reactor components resulting from failure of an EXPERIMENT involving explosive or corrosive materials.

Specification:

1. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, shall not be irradiated in the reactor.

2. EXPERIMENTS known to contain corrosive materials shall be double encapsulated.

3. EXPERIMENTS shall be designed such that they will not contribute to the failure of other EXPERIMENTS, core components, or fuel cladding.

Basis: This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive or corrosive materials (Ref. ANSI/ANS-15.1-2007).

3.8.3 Fueled Experiment Malfunctions

Applicability: MODES 1 and 2.

Objective: To ensure fueled EXPERIMENT malfunctions are bounded by accident analyses.

Specification:

1. Each fueled EXPERIMENT shall be limited such that the total inventory of iodine isotopes 131 through 135 in the EXPERIMENT is not greater than 0.01 curies.

2. Fueled EXPERIMENTS shall be designed such that they will not contribute to the failure of other EXPERIMENTS, core components, or fuel cladding.

Basis: This specification ensures that malfunction of a fueled experiment remains bounded by the accident analyses of SAR Section 13.2 and designed such that they do not contribute to the failure of other experiments and reactor components (Ref. ANSI/ANS-15.1-2007).

3.9 Other Facility Specific Limitations

3.9.1 Shield Tank Level

Applicability:	MODES 1 and 2.
Objective:	To specify the minimum OPERABILITY requirement for the shield tank.
Specification:	Shield tank water level shall be no less than 6 inches below the normal established level.
Basis:	Maintaining shield tank water level within 6 inches of the normal established level ensures sufficient water to adequately shield the west side of the reactor core during full power reactor operation (Ref. SAR Chapters 4, 7, 9, 10 and 11).

SURVEILLANCE REQUIREMENT

	SURVEILLANCE	FREQUENCY
SR 3.9.1	Verify shield tank water level is within the limit	Daily

3.9.2 Fuel and Fuel Handling

Applicability: According to Table 3.9.2-1.

Objective: To establish fuel integrity and fuel handling operations remain bounded by the accident analyses.

Specification: According to Table 3.9.2-1.

Basis: Operation with damage free fuel ensures consequences of accidents involving a fission product release remain bounded by the analysis provided in SAR Section 13.2. Limiting entry into MODE 5 until at three days after shutdown ensures actual fuel fission product inventory remains bounded by the conservative calculated fission product inventory provided in SAR Section 13.2.

Table 3.9.2-1
Fuel and Fuel Handling Limitations

	LIMITING CONDITION	APPLICABLE MODES	SURVEILLANCE REQUIREMENTS
1.	The reactor shall not be operated with DAMAGED FUEL in the core except to locate the damaged in-core fuel	1, 2	SR 3.9.2.1
2.	At least two layers of concrete block shielding shall remain fully installed over the core area until a minimum of three days have passed since the last operation in MODE 1	5	SR 3.9.2.2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.2.1	Reactor coolant water shall be sampled and evaluated for indications of DAMAGED FUEL	Weekly
SR 3.9.2.2	Verify the integrity of in-core reactor fuel assembly cladding by visual inspection of at least 8 in-core reactor fuel assemblies. DAMAGED FUEL assemblies and assemblies with FUEL DEFECTS shall be removed from the core	10 years
SR 3.9.2.3	Verify a minimum of three days have passed since last operation in MODE 1	Prior to MODE 5 entry

4.0 This section intentionally left blank. Surveillances are included in Section 3.0

5.0 Design Features

5.1 Reactor Cell

Applicability: At all times.

Objective: To specify REACTOR CELL features supporting facility radiological assumptions.

Specification:

1. The REACTOR CELL shall be located at the north end of the Reactor Building which is located on the main campus of the University of Florida in the vicinity of the buildings housing the College of Engineering and the College of Journalism.

2. The REACTOR CELL shall be equipped with independent air conditioning and ventilation systems.

3. The REACTOR CELL core ventilation system effluents shall be discharged through a stack at a minimum of 25 feet above ground level.

4. The REACTOR CELL minimum free volume shall be 36,000 cubic feet.

Basis: To ensure changes to specified REACTOR CELL features supporting radiological safety assumptions are not made without prior NRC approval.

5.2 Reactor Coolant System

Applicability:	When Reactor Coolant System water is in contact with fuel assemblies loaded into the reactor core.
Objective:	To specify Reactor Coolant System design features that support gravity draining of the core water moderator.
Specification:	The reactor coolant water flow path shall be from the storage tank located in the equipment pit through the heat exchanger up to the bottom of the fuel boxes, upward past the fuel assemblies to overflow pipes and into a header for gravity driven return to the storage tank.
Basis:	Fuel boxes are elevated above other major Reactor Coolant System components to ensure any event causing a loss of primary coolant flow results in the water moderator gravity draining from the fuel boxes thereby shutting down the reactor (Ref. SAR Section 5.2).

5.3 Reactor Core and Fuel

5.3.1 Reactor Core Design

Applicability: MODES 1 through 5.

Objective: To specify Reactor Core design features which if altered could affect safety.

Specification:

1. The reactor core shall contain six aluminum fuel boxes, containing up to four fuel assemblies each, arranged in two parallel rows of three boxes each, and separated by about 30 cm of graphite.

2. The reactor core shall contain four control blades of swing-arm type consisting of aluminum vanes tipped with cadmium, protected by magnesium shrouds.

3. The reactor core shall contain the surrounding graphite assembly that measures about 5' x 5' x 5'.

4. The reactor core shall contain experimental locations to include three vertical columns and one horizontal throughport.

Basis: This ensures specified reactor core design features remain as analyzed in SAR Chapters 4 and 13.

5.3.2 Reactor Core Fuel Loading

Applicability: MODES 1, 2 and 3.

Objective: To ensure the operational reactor core is loaded as intended and contains no fewer full fuel assemblies than the limiting CORE CONFIGURATION.

Specification:

1. The reactor core shall contain no less than 22 full fuel assemblies and shall be loaded so that all fuel assembly positions are occupied.

2. The reactor core shall contain up to 24 fuel assemblies of 14 plates each. Up to 6 of these assemblies may be replaced with pairs of partial assemblies. Each partial assembly shall be composed of either all dummy or all fueled plates. A full assembly shall be replaced with no fewer than 13 plates in a pair of partial assemblies.

Basis: This ensures the reactor core is loaded as intended and that the operational fuel loading remains bounded by the limiting CORE CONFIGURATION described in SAR Chapters 4 and 13.

5.3.3 Reactor Fuel Design

Applicability: MODES 1 through 5.

Objective: To specify the proper reactor fuel type and burnup limit.

Specification:

1. Fuel assemblies installed in the core shall be of the general MTR type, with thin fuel plates clad with aluminum 6061 and containing uranium silicide-aluminum (U3Si2-Al) fuel meat enriched to no more than about 19.75% U-235.

2. Fuel assembly burnup shall not exceed 50% of its initial U-235 content.

Basis: This ensures the reactor core is loaded with the proper type fuel as analyzed in SAR Chapters 4 and 13 and that fuel burnup is limited to within the evaluation limits of NUREG-1313.

5.4 Fuel Storage

Applicability: At all times.

Objective: To ensure fuel in storage remains subcritical.

Specification: Fuel, including fueled EXPERIMENTS and fueled devices, not in the reactor shall be stored in a geometry that ensures keff is no greater than 0.90 for all conditions of moderation and reflection using light water.

Basis: This ensures fuel in storage remains subcritical as described in SAR Section 9.2.

6.0 Administrative Controls

6.1 Organization

6.1.1 Structure

The organization for the management and operation of the reactor facility shall include the structure indicated in Figure 6-1. Job titles are shown for illustration and may vary. Four levels of authority are provided.

Level 1 - Individuals responsible for the reactor facility's licenses, charter, and site administration.

Level 2 - Individual responsible for reactor facility management.

Level 3 - Individual responsible for reactor operations, and supervision of day-to-day facility activities.

Level 4 - Reactor operations staff.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Figure 6-1. In addition to having responsibility for the policies and operation of the reactor facility, individuals at various management levels shall be responsible for safeguarding the public and facility personnel from undue radiation exposures, and for adhering to all requirements of the operating license and Technical Specifications. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

6.1.3 Staffing

1. The minimum staffing when the reactor is in MODES 1, 2, or 3 shall be:

    a. An operator in the control room;

    b. A designated second person present at the facility complex able to carry out prescribed written instructions; and

    c. A designated senior operator shall be readily available on call. "Readily Available on Call" means an individual who:

        i. has been specifically designated and the designation known to the operator on duty;

        ii. can be rapidly contacted by phone or other means of communication available to the operator on duty; and

        iii. is capable of getting to the reactor facility within 30 minutes under normal conditions.

2. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list shall include:

    a. Management personnel,

    b. Radiation control personnel, and

    c. Other operations personnel.

3. Events requiring the presence at the facility of a senior operator are:

    a. All CORE ALTERATIONS,

    b. Initial startup and approach to power,

    c. Relocation of any EXPERIMENT with reactivity worth greater than 720 pcm, and

    d. Recovery from UNSCHEDULED SHUTDOWN

6.1.4 Selection and Training of Operations Personnel

The selection and training of licensed operations personnel should be in accordance with the American National Standard, ANSI/ANS-15.4-1988, Selection and Training of Personnel for Research Reactors.

6.2 Review and Audit

6.2.1 RSRS Composition and Qualifications

1. The RSRS shall be composed of a minimum of three members with expertise in reactor technology and/or radiological safety.

2. Members of the RSRS shall be appointed by the Chair of the Radiation Control Committee (RCC).

3. Qualified and approved alternates may serve in the absence of regular members.

6.2.2 RSRS Rules

RSRS functions shall be conducted in accordance with the following charter:

1. At least one meeting shall be held annually. Meetings may be held more frequently as circumstances warrant, consistent with the effective monitoring of facility operations as determined by the RSRS Chair;

2. The RSRS Chair shall ensure meeting minutes are reviewed, approved, and submitted in a timely manner; and

3. A quorum shall consist of at least three members where the operating staff does not constitute a majority.

6.2.3 RSRS Review Function

The following items shall be reviewed:

1. Changes performed under 10 CFR 50.59;

2. New procedures and major revisions of existing procedures having safety significance;

3. Proposed changes to a SSC having safety significance;

4. Proposed changes in Technical Specifications or license;

5. Violations of Technical Specifications or license;

6. Violations of procedures having safety significance;

7. Operating abnormalities having safety significance;

8. Reportable occurrences listed in Section 6.7.2; and

9. Audit reports.

6.2.4 RSRS Audit Function

The following items shall be audited:

1. Facility operations for conformance to the Technical Specifications and applicable license conditions, annually;

2. The retraining and requalification program for the operating staff, biennially;

3. The results of action taken to correct deficiencies in reactor SSCs or methods of operations that affect reactor safety, annually; and

4. The emergency plan and emergency implementing procedures, biennially.

A report of audit findings shall be submitted to the Dean of the College of Engineering and RSRS members within three months after the audit has been completed.

6.3 Radiation Safety

The Radiation Control Officer shall be responsible for implementation of the radiation protection program and shall report to Level 2 or higher.

6.4 Procedures

The UFTR facility shall be operated in accordance with approved written procedures. Operating procedures shall be in effect for the following items:

1. Normal startup, operation and shutdown of the reactor;

2. Fuel loading, unloading, and movement within the reactor;

3. Maintenance of major components of systems that could have an effect on reactor safety;

4. Surveillances and inspections required by the Technical Specifications or those that may have an effect on reactor safety;

5. Personnel radiation protection, consistent with applicable regulations. The procedures shall include management commitment to maintain exposures as low as reasonably achievable (ALARA);

6. Administrative controls for operations and maintenance and for the conduct of irradiations and EXPERIMENTS that could affect reactor safety or core reactivity;

7. Implementation of the Emergency Plan and security procedures; and

8. Procedures for the use, receipt, and transfer of by-product material, if appropriate.

Changes to the above procedures shall be made only after review by the RSRS and approval by the Facility Director.

6.5 Experiment Review and Approval

Approved EXPERIMENTS shall be carried out in accordance with established and approved procedures. In addition:

1. All new EXPERIMENTS or class of EXPERIMENTS shall be reviewed by the RSRS and approved in writing by the Facility Director or designated alternates prior to initiation; and

2. Substantive changes to previously approved EXPERIMENTS shall be made only after review by the RSRS and approval in writing by the Facility Director or designated alternates. Minor changes that do not significantly alter the EXPERIMENT may be approved by Reactor Manager or higher.

6.6 Required Actions

6.6.1 Actions to be Taken in the Event of a Safety Limit Violation

1. The reactor shall be shut down, the Facility Director shall be notified, and reactor operations shall not resume until authorized by the NRC;

2. The NRC shall be notified in accordance with Section 6.7.2; and

3. A safety limit violation report shall be prepared. The report shall describe the following:

   a. Applicable circumstances leading to the violation including, when known, the cause and contributing factors;

   b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and

   c. Corrective action to be taken to prevent recurrence.

The report shall be reviewed by the RSRS and any follow-up report shall be submitted to the NRC when authorization is sought to resume operation of the reactor.

6.6.2 Actions to be Taken in the Event of a Reportable Occurrence of the Type Identified in Section 6.7.2(a) Other Than a Safety Limit Violation

1. Reactor conditions shall be returned to normal, or the reactor shall be shut down;

2. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Facility Director or designated alternates;

3. Occurrence shall be reported to the Facility Director or designated alternates and to the NRC as required in Section 6.7.2; and

4. Occurrence shall be reviewed by the RSRS at its next scheduled meeting.

6.7 Reports

6.7.1 Annual Operating Report

An annual report covering the previous calendar year shall be submitted to the NRC by November 1 of each year consisting of:

1. A brief summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both;

2. The UNSCHEDULED SHUTDOWNS including, where applicable, corrective action taken to preclude recurrence;

3. Tabulation of major preventive and corrective maintenance operations having safety significance;

4. A brief description, including a summary of the change evaluation, of changes, tests, and EXPERIMENTS implemented under 10 CFR 50.59;

5. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the facility licensee as determined at, or before, the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient;

6. A summarized result of environmental surveys performed outside the facility; and

7. A summary of exposure received by facility personnel and visitors where such exposures are greater than 25% of that allowed.

6.7.2 Special Reports

a. There shall be a report not later than the following working day by telephone and confirmed in writing by facsimile or similar conveyance to the NRC, to be followed by a written report that describes the circumstances of the event within 14 days of any of the following:

1. Violation of safety limit;

2. Release of radioactivity from the site above allowed limits;

3. MODE 1 or MODE 2 operation with actual trip system settings for required systems less conservative than the limiting safety system settings;

4. MODE 1 or MODE 2 operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in Section 3;

5. A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is discovered during MODES or conditions in which the LCO is not applicable then no report is required;

   Note: Where components or systems are provided in addition to the minimum required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required are capable of performing their intended function.

6. An unanticipated or uncontrolled change in reactivity greater than 720 pcm. Reactor trips resulting from a known cause are excluded;

7. Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary (excluding minor leaks), or REACTOR CELL boundary (excluding minor leaks) where applicable; or

8. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

b. There shall be a written report within 30 days to the NRC of the following:

1. Permanent changes in the facility organization of Level 1 or 2 personnel; and

2. Significant changes in the transient or accident analyses as described in the Safety Analysis Report.

6.8 Records

6.8.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less Than Five Years

1. Normal reactor operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year),

2. Principal maintenance operations,
3. Reportable occurrences,

4. Surveillance activities required by the Technical Specifications,

5. Reactor facility radiation and contamination surveys where required by applicable regulations,

6. EXPERIMENTS performed with the reactor,

7. Fuel inventories, receipts, and shipments,

8. Approved changes in operating procedures, and

9. Records of meetings and audit reports of the RSRS.

6.8.2 Records to be Retained for at Least One Training Cycle

Record of retraining and requalification of operators shall be maintained at all times the individual is employed or until the operators license is renewed.

6.8.3 Records to be Retained for the Lifetime of the Facility

1. Gaseous and liquid radioactive effluents released to the environs,

2. Offsite environmental monitoring surveys,

3. Radiation exposures for all personnel monitored, and

4. Drawings of the reactor facility.

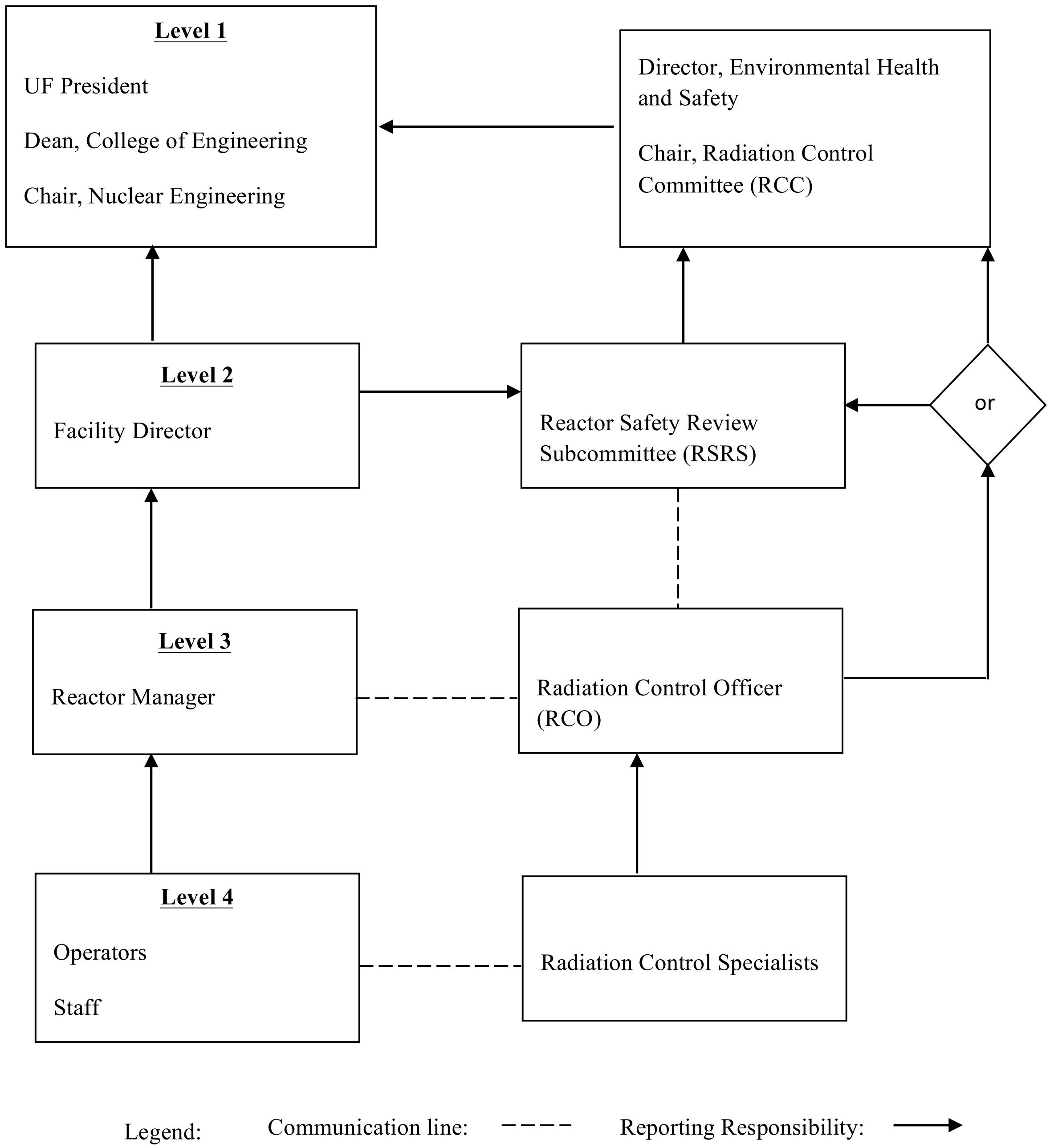


Figure 6-1 UFTR Organizational Chart