RESPONSE TO NRC
REQUEST FOR ADDITIONAL INFORMATION
RELATED TO LICENSE RENEWAL

FOR THE

UNIVERSITY OF FLORIDA TRAINING REACTOR

LICENSE NO. R-56
DOCKET NO. 50-083

REDACTED VERSION

SECURITY-RELATED INFORMATION REMOVED

Redacted text and figures blacked out

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April 7, 2008

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Washington, DC 20555

UFTR Request for License Renewal

University of Florida Training Reactor, Facility License: R-56, Docket No. 50-83
Responses to NRC Request for Additional Information Dated April 6, 2005

This submittal contains the University of Florida Training Reactor (UFTR) facility official responses to the questions contained in your Request for Additional Information (RAI) dated April 6, 2005 concerning our request for license renewal submitted on July 29, 2002.

As with previous exchanges, the responses in the current submittal frequently reference the earlier submittals made and approved to support conversion of the UFTR from using high-enriched uranium to using low-enriched uranium fuel. Nevertheless, the answers to all questions are contained in the attached set of responses.

This entire submittal consists of one signed original letter of transmittal plus *Attachment I* containing the responses to the USNRC Request for Additional Information.

A020
NRR

We appreciate your consideration of this submittal. Please advise if further information is needed.

Sincerely,

William G. Vernetson
Director of Nuclear Facilities

Email: vernet@ufl.edu
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WGV/dms
Attachment I

cc: Al Adams, NRC Senior Project Manager
Rob Kuntz, NRC Project Manager
Craig Bassett, NRC Inspector
Reactor Safety Review Subcommittee

Sworn and subscribed this 7 day of April 2008.

Diana Dampier
Notary Public

Diana L. Dampier
Commission # DD452982
Expires July 20, 2009
Bonded Troy Fain - Insurance, Inc. 800-385-7019

ATTACHMENT I

UFTR RESPONSES
TO USNRC
REQUEST FOR ADDITIONAL INFORMATION
DATED
APRIL 6, 2005

Responses to the US NRC Request For Additional Information
University of Florida Training Reactor
Docket No. 50-83

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March 2008

1 THE FACILITY

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

The reactor building including the reactor facility gets its electrical power and potable water supply from the adjacent building (Nuclear Science Center). The potable water supply that provides secondary cooling for the reactor has a backflow preventer. There is also a building pneumatic air supply system line for the building. This line runs through the reactor cell. However, it has no connections in the reactor cell itself.

2 SITE CHARACTERISTICS

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

Sections 2.1.1.2 and 2.1.1.3 should be referencing to 10 CFR 20. The words 'exclusion area' should be replaced by 'restricted area'. They should read as follows:

2.1.1.2

"The restricted area for this reactor facility is the reactor cell and the control area since this is a low power training and research reactor. Other adjacent areas, including the west lot, laboratories, and offices, are also protected (as defined in 10 CFR Part 73). These areas can be upgraded to restricted area when required"

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

No.

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

No military installation is near enough to affect the reactor building. The nearest military instillation is Camp Blanding. It is located along SR-26 about 67 miles from the UFTR. It is primarily used as a training facility for performing ground deployment training exercises.

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

Local meteorological data can be obtained from the University of Florida Physics Department. The weather sensors are located on the roof of the physics building about 1435 feet from the UFTR and real time meteorological data can be obtained from Local Physics Weather Station webpage at http://www.phys.ufl.edu/weather/.

In addition, local gridded weather data may be obtained from the National Oceanic and Atmospheric Administration (NOAA). The daily or monthly climate report website for the Gainesville area is:

http://www.weather.gov/climate/index.php?wfo=jax .

Gainesville Regional Area should be chosen for the location. It shows current and archived temperature, precipitation, snowfall, heating, cooling, wind and sky cover data.

Gainesville's climate is defined as humid subtropical. The major weather concern would be hurricanes and tropical storms. However, historically, the area is rarely struck seriously by hurricanes due to its inland location. Over the last 50 years, the most severe hurricane condition occurred in the 2004 Atlantic hurricane season. Two Category 3 hurricanes caused some power loss in some areas. However, other than power surges, they have no direct impact on the reactor building as they were downgraded to Category I or less on arriving in the area. Tornados are also a possibility in the area, though tornados cover much smaller areas in Florida than in the Midwest. With the location of the core in the reactor building and the building location surrounded by taller buildings, significant effects from tornados are unlikely.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

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

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

When the reactor building was built in 1958, the state required no specific building code. The building would not have been designed to specific seismic limits. However, the building would likely have been designed for 100 mph wind loads (x1.15 safety factor) which it has probably already withstood on occasion over the 50 years of its existence.
Starting in 1999 all campus constructions including reactor building modifications would have had to meet the standard Building Code from 1997 since Florida had no code. Currently all such modifications have to meet the 2004 Florida Building Codes.

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

Administrative controls for use of the overhead crane during reactor operations are addressed in UFTR SOP-A.2 REV 13, "Reactor Startup" as follows:

1. The 3-ton bridge crane shall not be used in a manner that could damage the control system or prevent the system from performing its intended function.
2. Loads over 500 pounds shall not be lifted over the control blade mechanisms unless the control blades are fully inserted.
3. Only a licensed reactor operator shall operate the crane during reactor operations.

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

The stack vent release point, is 29 ft., 8 in. from the ground. A Core Vent Fan located in the reactor equipment pit (pit A) draws air from the UFTR core and the air travels to a second pit (pit B). A stack dilution fan, situated in pit B, draws outside air from the west lot. This intake is pulled through a series of filters located on the west wall of the reactor building. The air from the core vent and the stack dilution fan are combined in pit B and pushed through the stack vent release point. A schematic of the UFTR Core Vent System can be found in UFTR FSAR Chapter 9, Fig. 9-2.

4 REACTOR DESCRIPTION

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

No significant material degradation was noted in the HEU fuel removed after decades of operation in August 2006. Beginning in early 2006 the pH has been measured weekly from a grab sample. With the conversion to LEU fuel, the pH value of the reactor coolant is required to be below 7.0 per the Technical Specification requirements. The weekly grab samples indicate the pH has been well below 7.0 in all measurements.

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

To prevent fuel cladding corrosion, primary coolant resistivity is controlled, and its pH value is monitored. The effect of hydraulic forces on the fuel integrity is negligible. The UFTR fuel design is approved in the HEU to LEU Fuel Conversion SAR. Since unexpected corrosion was found in another Argonaut reactor (UTR-10 at Iowa State University) with similar type of LEU fuel plate as UFTR, the manufacturer (BWXT) of LEU fuel plates applied a surface treatment resulting in a protective boehmite layer on the surface of the cladding. A Tech Spec was added to assure the primary coolant pH value is less than 7.0. It is concluded that unexpected corrosion should not occur in the UFTR core. Radiation effects should not cause serious damage to the fuel during its lifetime expected to be at least 20 years.

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

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

Control blade drop times are typically 0.5~0.6 sec. In some cases, maintenance has been necessary to replace bushings or better align the shafts. Extending the limit to 1.5 sec increases the margin. The change is also in the interest to meet ALARA requirement to avoid unnecessary access to the core area.

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

There is negligible impact on the reactor safety margins as shown in Tables 13-3 and 13-5 in the HEU to LEU Fuel Conversion SAR.

Table 13-3 RELAP5-3D Results for Step Insertion of 0.6% Δk/k in UFTR.

Case	**HEU Core**			**LEU Core**		
	No SCRAM	**SCRAM (1.0s drop)**	**SCRAM (1.5s drop)**	**No SCRAM**	**SCRAM (1.0s drop)**	**SCRAM (1.5s drop)**
P_o (kW)	100			100		
Reactivity Insertion	0.60%			0.60%		
	$0.76			$0.78		
Length of Transient Modeled (s)	300.0	30.0	30.0	21.2	30.0	30.0
Time to Peak Power (s)	2.57	0.17	0.17	2.46	0.14	0.14
Peak Power (MW)	1.30	0.29	0.30	1.25	0.32	0.32
$T_{fuel,max}$ (°C) (at Peak Power)	89	54	54	96	52	52
$T_{fuel,max}$ (°C)	108	54	55	107	52	52
$T_{clad,max}$ (°C)	108	54	55	107	52	52
$T_{cool,max}$ (°C)	101	44	44	101	45	45

Table 13-5 Transient Response from a Sudden Reactivity Insertion

	SPERT-I	**Argonaut-UTR**[2]	**UFTR HEU Core**		**UFTR LEU Core**
ρ (% Δk/k)	2.63%	2.6%	2.6%	2.3%	2.3%
ℓ (μs)	60	140	187.4	187.4	177.5
β_{eff}	0.0075	0.0065	0.00791	0.00791	0.00771
α (s^{-1})	313	138	96.5	80.5	86.1
E_{total} (MWs)	30.7	12	9.3	8.5	<8.8
T_{max} (°C)	1500[1]	586	453	415	<428

[1] Estimated maximum temperature; from NUREG/CR-2079 (Ref. 17).
[2] From analysis summarized in NUREG/CR-2079.

c. *Section 4.2.2.1, Table 4-1: Table 4-1 states that the worth of the three safety shim arms are as follows: #1, 122(sic)% Δk/k; #2, 1.35% Δk/k; and #3, 1.83% Δk/k. However in the technical specifications (TSs) (Section 5.5) the blade worths are stated to be between about 1.3 and 2.0% Δk/k. Please make the SAR and TSs consistent.*

The control blade worths have been re-evaluated in the HEU to LEU Fuel Conversion SAR. The Tech Specs are also updated. The control blade worths in the SAR and TS agree in the Fuel Conversion SAR.

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

The fuel options referenced in Section 4.2.1 (Fuel System Design) and TS 5.3.1 (Reactor Fuel) have become irrelevant and obsolete upon completion of the HEU to LEU Fuel Conversion. The HEU to LEU Fuel Conversion SAR (Section 4.2.1 and Section 4.5) provides the details of the LEU fuel design fabrication specifications, as well as the comparison with the HEU fuel. TS 5.3.1 is also updated as described in the Fuel Conversion SAR (Chapter 14 p. 63).

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

Life expectancy of the fuel assembly and control blades is at least 20 years. The control blades are described in Section 4.2.2 of the HEU to LEU Fuel Conversion SAR. Only a negligible fraction of Cadmium in the blades is activated during life time. Control blade worth is not expected to change significantly with time. The control blade worth change due to fuel burnup is shown in Table 4-13 (Section 4.5.3 of the Fuel Conversion SAR):

Table 4-13 Comparison of Control Blades Worth for the HEU and LEU Cores

Control Blade	**HEU (calculated)**	**HEU (measured)**	**LEU-fresh (calculated)**	**LEU-depleted (calculated)**
Regulating	0.87%	0.82%	0.63%	0.66%
Safety 1	1.35%	1.21%	1.62%	1.65%
Safety 2	1.63%	1.36%	1.77%	1.76%
Safety 3	2.06%	1.88%	1.42%	1.46%

4-6 *Section 4.2.2.1, Control Rods. This section states the usual detailed information is not provided since the 'control rod systems are previously operated systems', however no information on this system is discussed in Section 16.1; Prior Use of*

Reactor Components. Please provide a reference which discusses the operating history of the control rods.

Rework of the control blade clutch was performed on Safety Blade 2 in 2003 as documented on Maintenance Log Page #03-30 and Modification Package #03-06 (Control Blade Safety-2 Clutch Replacement).. The graphite bearings and other components in all control blades were replaced in 1985-1986 reporting year to correct a sticking Safety-3 control blade.

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

The life of the remaining graphite is considerable. Based on visual inspections, the chipped graphite does not require immediate attention or specific procedural changes.

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

Handling the sources is only permitted by personnel specifically trained to do so. Specifically the sources are only allowed to be lowered into or removed from the reactor by properly trained personnel. When not in use, the PuBe source, is stored in a shielded cylindrical container outside the reactor. The SbBe rechargeable source is normally left in the west vertical port.

Persons who handle the neutron sources receive 10 CFR Part 19 Training (Radiation Worker Instructions), and are certified to have received the training which includes use of radiation survey meters, methods of limiting radiation dose, types of radiation and methods of shielding. In addition, such individuals are certified as "Second Persons" at the reactor facility. A "Second Person" is defined as an individual capable to implement the initial stages of emergency response and meets the Tech Spec requirement in Section 6.1.3 for a second person when the reactor is not secured. These individuals receive training specifically oriented to assuring knowledge in responding to radiological emergencies which is documented with a test on which they must receive 80% as a passing grade. In addition, such individuals would receive specific practical instruction in handling the UFTR neutron sources. Finally, insertion of the neutron source into the reactor or removal from the reactor is always performed with reactor operator cognizance and supervision.

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

The remaining life of the support structures is expected to be as long as the facility exists. The low heat generation is conducive to the continued structural integrity, and currently no stress or damage is indicated.

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

Changes in both flow into the core and vessel coolant level is monitored. Leakage from the reactor tank would drain down and collect in the equipment pit. Significant leakage would be noticed by increases in the make-up coolant volume, which is checked weekly. The pit is checked weekly; therefore, a noticeable leak would be detected within a week at most. The pit water alarm would notify of larger leaks above a gallon immediately; this visual and audible alarm is verified to be operable as part of the weekly preoperational checks.

The tank water is checked weekly for radioactivity using a 100 ml sample. Usually no detectable activity exists and all leakage is confined to the building, with no significant public impact.

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

The Tech Spec safety limit of *200°F* has been removed. This question is not applicable. Section 4.3 is updated per the HEU to LEU Fuel Conversion SAR.

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

No, the addition is not planned at this time.

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

Yes, the 3 mR/hr is a measurement taken at the surface of the biological shielding, and the other measurements are taken at a distance of 1 foot from the biological shielding. These values are not expected to change significantly for the LEU fuel.

Note that these exposure values are measured quarterly, and the actual measured values may vary slightly.

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

Ground water and soil activation is not possible. The primary coolant loop from outlet to inlet is a closed loop entirely within the reactor building with no potential risks for external activation. The Radiological Environmental Monitoring Program measures doses and external to the facility. There is a 2 foot thick concrete slab beneath the reactor between the core and the underlying ground below the building. Therefore, there is no potential for ground water or soil activation

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

The 2.5 mR/hr is the area monitor warning setpoint. This setpoint provides an early warning of a potentially unusual radiation level, which should be further investigated depending upon the operation in progress. The warning is in place to assure the 10 mR/hr alarm level is not reached unexpectedly.

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

Because of the low heat generation in the core, no degradation of the biological shielding is indicated. The shielding is staggered to assure no direct pathways for the streaming of radiation. The plugs in the ports are similarly staggered. Outside the reactor building specific areas such as the rabbit system utilizes extra shielding for samples that have been irradiated. Portable shielding is also available for use around open ports as necessary to meet requirements of Radiation Work Permits. Radiation surveys are performed whenever activities in progress provide the potential for increased radiation levels. Such situations would include experiments that involve operation with shielding removed or altered as well as operations involving movement of "permanent" shielding. In such cases, after the shielding is replaced, radiation surveys are performed in steps as power level is raised to full power (or the highest power allowed) to assure no significant changes in shielding effectiveness. Shielding changes are further controlled by SOP E.2 (Alterations to Reactor Shielding and Graphite Configuration).

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

Spent fuel storage is described in Section 9.2.2 of the FSAR. The criticality safety of the spent fuel storage is analyzed in the HEU to LEU Fuel Conversion SAR. Irradiated spent fuel is adequately shielded in the irradiated fuel storage ports. A process is in place to assure adequate radiation surveys are performed upon placement of irradiated fuel in the storage facility and periodically thereafter at least every six (6) months, Historical data including experimental measurements indicates that the fuel storage facility is adequately designed to contain an entire offloaded irradiated core with no radiation level issues.

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

Interlocks and operational limits specified in operating procedures exist to prevent inadvertent addition of positive reactivity. The interlocks are specified in TS 3.2.1(5) and the operational limits are specified in procedures (SOP A.2 and A.3) as well as in TS 3.5(3). Some important interlocks are as follows:

1) Source count rate less than 2 cps
2) Reactor period less than 10 seconds
3) Raising a 2 or more blades when reactor is operating manually. Administrative controls prevent raising 2 or more blades when reactor is operating in automatic.

Limitations are placed on all experiments to assure the Tech Spec limitations on reactivity are met.

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

Section 4.5.1.1 Flux Distribution became irrelevant for the LEU core. The flux distributions for both the LEU core and the HEU core at 100 kW are re-evaluated in the HEU to LEU Fuel Conversion SAR (Appendix A5).

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

This question became irrelevant due to the Fuel Conversion. Flux distribution is calculated with the Monte Carlo approach in Appendix A5 in the Fuel Conversion SAR. The energy group boundaries are given in Table A5-1 in the Fuel Conversion SAR.

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

All the values are re-calculated in Section 4.5.3 and 4.5.4 of the Fuel Conversion SAR for both HEU and LEU cores. Excess reactivity results are compared in Table 4-12 for both the HEU (fresh and depleted) core and LEU (fresh and depleted) core. Control blade worth results are given in Table 4-13 for the depleted HEU core and fresh and depleted LEU core. Shutdown margin results are given in Table 4-16 for depleted HEU core and fresh LEU core.

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

The analysis in Section 4.5.2 became irrelevant for the LEU core. The core physics parameters for both the fresh and depleted HEU and LEU cores are re-calculated and compared in Section 4.5.5 of the Fuel Conversion SAR.

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

Reactor Core Physics Parameters for both the LEU fresh and depleted cores are re-calculated in the Fuel Conversion SAR. These parameters do not change significantly with burn up. The results are given in Table 4-18 in Appendix Q4 in response to Request for Additional Information for UFTR Fuel Conversion (submitted on June 20, 2006) as attached bellow:

Table 4-18 Kinetics Parameters and Reactivity Coefficients for the UFTR LEU Core.

Parameter		Fresh Core	Depleted Core
β_{eff}		0.00771 ± 1%	0.00756 ± 2%
ℓ (μs)		177.5 ± 5%	195.1 ± 6%
C_{void} ($\Delta\rho$/%void)	*(0 to 5% void)*	-1.53E-03 ± 1%	-1.46E-03 ± 2%
	(5 to 10% void)	-1.75E-03 ± 1%	-1.65E-03 ± 2%
C_{water} ($\Delta\rho/^oC$)	*(21 to 127°C)*	-5.68E-05 ± 2%	-5.26E-05 ± 3%
C_{fuel} ($\Delta\rho/^oC$)	*(21 to 127°C)*	-1.76E-05 ± 6%	-1.72E-05 ± 9%
	(21 to 227°C)	-1.65E-05 ± 3%	-1.49E-05 ± 4%

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

The flux distribution in the LEU core is calculated with the Monte Carlo approach in Appendix A5 in the Fuel Conversion SAR.

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

Safety limits and limiting safety system settings can be found in Technical Specifications, Appendix A, Section 2.1 and 2.2. Technical Specifications, Appendix A, Section 3.0 describes the limiting conditions for operation and reactor safety systems. This document also contains the surveillance requirements pertaining to safety limits, limiting safety settings, and limiting conditions for operations in Section 4. The reference for the Tech Spec changes due to the fuel conversion is provided in the Fuel Conversion SAR. The methodologies to determine the applicable LSSSs are also discussed in the Fuel Conversion SAR. A list of LSSS changes and the reference is provided as follows:

1. LSSS on power level is selected to incorporate conservative 5% uncertainty from previous 125 kW long standing LSSS on power level. The remainder of the basis is in the HEU to LEU Fuel Conversion SAR.

2. LSSS on PC Flow Rate is selected based on power/flow calculations, which are summarized in the Addendum 2 of the Fuel Conversion SAR. The calculations are based on normal fuel spacing (110 mils) on the flow channel in the bundles. We are currently utilizing 90 mils as most restrictive assumption based upon measurements on all fuel bundles.

3. LSSSs on inlet temperature and fuel box outlet temperatures are again selected to provide margin to fuel/cladding damage per the Conversion SAR.

4. LSSS of 3 second period is selected as historical but can be based on the Conversion SAR accident analysis.

5. LSSS on High Voltage (10%drop) is based on experience with neutron detectors; at such a voltage drop the detector could no longer be expected to indicate reliably or accurately.

6. This LSSS on the outlet flow being non-zero provides additional conservatism as the outlet flow loss will provide a trip initiation prior to PC flow loss on the inlet meter.

7. This LSSS on the PC Pump is removed from the LSSS section.

8. The LSSS on the primary coolant level assures the fuel is covered adequately to assure cooling. Basically it is quite conservative since the fuel is below the outlet flow line so if there is outlet flow, then the 2" above the fuel is met.

9. Secondary coolant flow is required at power levels above 1 kW as the point-of-adding-heat (POAH) for this 100 kW reactor. The required well flow rate assures adequate cooling on well water flow to maintain low PC system coolant temperatures with large margins to the LSSSs on temperature. The lower allowed flow on the city water secondary cooling system is designed to allow higher operating temperatures on the primary coolant loop. However, these temperatures are still below the LSSS levels to protect the fuel/cladding but now allow higher coolant temperatures when such temperatures are desired for training or other reasons.

10. This LSSS on AC power is removed.

11. The LSSS on the dilution fan rpm indication (versus the previous operation of the vent AND dilution fan) assures adequate dilution of the reactor cell Argon-41 effluent with sufficient margin to provide confidence that accumulation of Argon-41 in the cell is prevented to protect personnel and that effluents are sufficiently diluted to assure meeting 10 CFR Part 20 release limits to unrestricted areas with a large margin.

12. The LSSS on the shield tank water level is set to assure that shield tank shielding effectiveness is maintained for all operations including full power operations with the shield tank top shield block and tank cover removed for experimental access for experiments, training and educational purposes. The periodic radiation surveys are performed assuming that this level is maintained.

5 REACTOR COOLANT SYSTEMS

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

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

The sketches for the primary and secondary systems are given in Figures 5-1 and 5-2 in the SAR, respectively. The entire primary coolant loop is located within the reactor building. The points where the secondary coolant loop penetrates the confinement are shown in the updated figure 5-4 given in Appendix A in this document. There are three points: the well water line inlet on the east wall, the city water line (alternate cooling line) on the south wall, and the secondary outlet drain line in the equipment pit to the storm sewer drain. The city water line that is connected for alternate secondary cooling has a code-required backflow preventer; in addition the city water line enters the reactor cell at about 12 ft elevation above the floor while the connection to the reactor secondary cooling heat exchanger line is at least 2 feet below floor level making backflow with radioactive contamination even less likely.

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

In the event of a piping/component failure outside the reactor core, the primary water is collected inside the equipment pit adjacent to the reactor north side biological shielding. The equipment pit water level monitor would alarm in the control room as described in TS 5.6.1(5). This level monitor alarms for less than 2 cups of water in the pit in a central depression below the pit floor level. This audible and visual alarm is verified operable during each weekly preoperational check.

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

The normal operating pressure applied to the primary coolant system is one atmosphere (14.7 psia), because the primary coolant system is not sealed. However, depending on the height of a given position in the system, the coolant pressure varies. At the graphite rupture disk, the pressure is about 3 psi(g) above atmospheric pressure. The disk is set to burst at 7 psid, which allows at least a pressure transient up to 3 psi.

5-2 *Section 5.3, Secondary Coolant System, page 5-3. The pressure of the secondary water is maintained higher than the primary system to prevent contamination of secondary coolant. What are the normal operating pressures of the primary and secondary coolant system? How are these pressures monitored? If a leak were to develop in the primary/secondary boundary, how would this be detected? Since the secondary water is tested weekly for radiological contamination (Appendix 14.1, Section 4.3, Item(4)), is there any way to identify such contamination that may be occurring in-between the weekly testing period (e.g., on continuous basis)?*

The normal operating pressure applied to the primary coolant system is one atmosphere (14.7 psia), because the primary coolant system is not sealed. However, depending on the height of a given position in the system, the coolant pressure varies from 14.7 psia to ~20 psia. The normal operating pressure for the secondary is not monitored. However, since the secondary flow rate is about 4 times higher than the primary flow rate, the secondary water dynamic pressure is expected to be higher than the primary system pressure. If a significant leak is developed on the primary/secondary boundary, the resistivity of the primary water is expected to change, which is constantly monitored and controlled. Insignificant leakage can be detected by the weekly radiological contamination tests. The heat exchanger is N-stamped to provide higher confidence of integrity.

Any failure of the heat exchanger resulting in leakage during operation would be subject to significant dilution. The leakage from the primary would enter the storm sewer system for further dilution. The most activation in the primary system would be during and after a lengthy operation at full power. The only significant activity seen in the samples of primary coolant analyzed on a weekly basis is sodium-24. Any nitrogen-16 decays too quickly to reach the environment.

With conservative assumptions on sodium in the primary coolant system, irradiation time, neutron flux level, cross section, primary-to-secondary leakage and secondary diluting flow, the following values are determined for a 1 liter/hr undetected leak rate continuing for 1 hour with 1 ppm sodium assumed in the primary coolant system.

Activation for 10 hours yields ~54 mCi Na-24 in the primary coolant tank at a concentration of ~0.0895 uCi/ml before dilution by the secondary flow.

For a 1 liter/hour leak rate undetected for an hour, the concentration assuming 140 gpm well water flow (minimum based on well water flow without flow warning light), the concentration becomes ~2.8E-06 uCi/ml. Public release is allowed at 5E-3 uCi/ml so we conclude that this unlikely event would not be a problem in this regard.

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

The normal flow rate for city water when it is used as a backup to well water varies depending on city water pressure. Typical flow rate is at or above ~35 gpm.

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

The coolant storage tank is located in the reactor equipment pit (i.e. pit A). There is normally a one-foot thick concrete shield above the reactor equipment pit Standard Operating Procedure SOP-A.3 REV 12, "Operation at Power" in Section 4.3 sets 10 kW as the administrative control limit at which the equipment pit shielding must in place. If it is necessary to operate above 10 kW without the shielding in place, then a radiation work permit is required to enhance necessary controls and ALARA considerations.

6 ENGINEERED SAFETY FEATURES

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

The accident analysis in the HEU to LEU Fuel Conversion SAR supports that no Engineered Safety Features are necessary. The Reactor Vent System is not necessary for reactor cooling. TS 3.3.2 specifies the core vent system functions to reduce gas leakage back into the reactor cell. During an emergency situation, it will be closed completely in order to reduce leakage of radioactivity.

7 INSTRUMENTATION & CONTROL

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

On the reactor console, the four control blade position indicators are light emitting diodes (LEDs) as a digital display. The master clock, temperature monitor and recorder system on the reactor console are also digital displays. All other system instruments remain analog, including a backup temperature monitor and recorder system

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

The basis of the 10 second delay is attributed to the inability to purge all air out of the secondary. The delay accounts for the possibility of air bubbles forming in the secondary system since this could cause the reactor to trip. The delay is implemented in order to prevent a spurious trip from occurring due to entrained air. The delay on well water is further justified since the primary coolant system operates at lower temperatures on well water giving a much larger margin to the LSSSs on inlet and outlet temperature..

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

Operational experience proves that average temperatures at 100KW do not approach the required LSSSs. This is the primary reason for the different setpoints. Additionally, these setponts assure reactor shutdown when required while preventing spurious scrams given the disparate, average flow rates in each system. In City Water Mode the time delay, by design, expires shortly after reaching 1KW and provides an immediate trip at any point that the city water flow drops below 8 gpm.

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

The reactor temperature scram function and alarms are dependent on software. The maintenance report indicates that a series of tests were performed by the engineer and technician at the time of construction and system installation. Other than the extensive tests performed prior to initial installation and as part of confirmatory operations tests, no formal testing outside the 50.59 modification process requirements were performed. Additionally, at the time of installation, it is unclear as to what other verification process might have been acceptable as few standards existed at that time. Even so, the temperature monitor does not constitute a common mode point for all reactor protection systems, and therefore does not represent any more of a failure risk than the replaced system which had a single output channel for all possible scram signals. In essence this system describes only one scram channel which does not interact any more than the other relay-based scram functions with the conventional scram chain. Operational experience shows that the software has performed well, as expected and designed for over seven years since installation.

Both the digital and analog system monitors are used in the decision making process by the reactor operator. Temperature monitor data from both the digital and analog system are stored and used to show compliance with license requirements. The digital system data is stored optically while the analog system continues to print paper reports.

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

This switchover is covered in UFTR SOP-A.6, "Operation of Secondary Cooling Water", Sections 7.0 to 7.2.6. When performed, this switchover is required to be documented by a line entry in the Operations Log and it must be performed by a SRO per steps listed in UFTR SOP-A.6. Currently no indication on the front of the reactor control console is available to show which secondary coolant source (and low flow reactor trip setpoint) is in effect. However, it is recorded in the log and in a daily pre-operational check, which is required prior to the reactor startup. There is an indication that the city water is in use, but there is no direct indication which trip mode is in effect other than the log entry.

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

For the return line flow instrument, the setpoint is at zero flow in the primary coolant return flow line. The surveillance frequency is the same as in the flow fill line, quarterly.

7-7 *Many older analog components have become obsolete and are no longer available; 'equivalent' replacement components are not always true replacements. How are replacement electronics components to repair the analog instruments and electronic circuit boards and logic modules, and the equipment in which they are installed, assured to be functionally operable?*

If the manufacturer cannot provide the exact replacement for an electronic component, a qualified substitute component is used. For non-identical components, a 50.59 evaluation is performed. Bench tests are performed prior to installation. And in-system tests are also performed after installation. The substitute component used along with verification of use is documented in the UFTR Maintenance Log. Procedures controlling the installation replacements are located in the UFTR SOP's.

8 ELECTRICAL POWER

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

A drawing is attached in Appendix B of this document. Note that the diesel generator in the figure is not required for any safety functions. (The diesel generator is connected with a dotted line).

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

Yes, the primary coolant system dump valve also drains the primary coolant system.

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

Internal filtering with isolation transformers is common in the electrical wiring. And the electrical wiring is arranged such that the separation between distribution and signal lines is optimized.

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

If required, auxiliary power such as UPS could be provided to any experimental equipment.

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

No diesel generator operation is required. Battery Backup systems are maintained for critical systems such as radiation monitoring systems and emergency lights. Habitability is not considered beyond the evacuation requirement. The diesel generator simply provides additional coverage because it is available.

9 AUXILIARY SYSTEMS

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

The term 'staff' refers to any individual with radiation worker training who is also second-person qualified. The term includes those who provide assistance with various support operations for the fuel movement activities. And it also includes management, radiation safety, and operations personnel. In all cases, fuel handling requires additional training. TS 3.7 and SOP C.1 and C.2 require that any fuel insertion and removal activities can only be performed by a licensed operator or a trainee under direct supervision, with an SRO present.

Persons who handle the neutron sources receive 10 CFR Part 19 Training (Radiation Worker Instructions), and are certified to have received the training which includes use of radiation survey meters, methods of limiting radiation dose, types of radiation and methods of shielding. In addition, such individuals are certified as "Second Persons" at the reactor facility. A "Second Person" is defined as an individual capable to implement the initial stages of emergency response and meets the Tech Spec requirement in Section 6.1.3 for a second person when the reactor is not secured. These individuals receive training specifically oriented to assuring knowledge in responding to radiological emergencies which is documented with a test on which they must receive 80% as a passing grade. In addition, such individuals would receive specific practical instruction in handling the UFTR neutron sources.

All persons directly involved in the fuel handling operations for loading and unloading the reactor core or moving fuel into, out of or from one location to another in the irradiated fuel storage facility receive Fuel Handling Training within 48 hours of performing the activity. This training includes classroom-type instruction as well as practical training with a dummy fuel bundle etc. The fuel handling training is intended to meet the requirements of Tech Specs Section 4.8(2) required to assure
reactor staff is properly qualified to perform fuel handling.

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

All shield blocks at the facility are below the 3-ton crane limit. Further limits are implemented by specific practical training for all who are allowed to use the crane. In the UFTR SOP-A.2 REV 13, the administrative controls for use of the overhead crane during reactor operations are as follows:

1. The 3-ton bridge crane shall not be used in a manner that could damage the control system or prevent the system from performing its intended functions.
2. Loads over 500 pounds shall not be lifted over the control blade mechanisms unless the control blades are fully inserted.
3. Only a licensed reactor operator shall operate the crane during reactor operations.

Annual inspections on the crane are performed by a company licensed to work on large cranes. The fuel transfer cask weighs ~2800 pounds.

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

The requirements of 10 CFR 70.24 are met by the installed UFTR radiation monitoring system with three channels, each of which alarms at 10 mR/hr as per TS 3.4.1. This system together with personnel dosimetry and hand-held survey meters can be used to provide medical personnel with the means of identifying individuals receiving high radiation doses. Supplies for decontamination are located on site and just off site at the Emergency Support Center and treatment facilities are close by at Shands Hospital at the University of Florida.

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

For UFTR operation up to full power, water can be drained from the core as a means of shutdown so it requires no active cooling when not operating. When stored in the Irradiated Fuel Storage Pits, the fuel has already cooled considerably as at least three days are required for the last power operation before accessing fuel in the core. As a result the temperature of the fuel stored in the Irradiated Fuel Storage Pit is near ambient and well below any temperature of concern.

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

This section refers to conditions as originally licensed. The sentence should read as follows:

All liquid effluents are collected from the reactor floor or drain into the equipment pit. They are transferred into one of the indoor hold-up tanks. Subsequently liquid effluents are pumped from the indoor hold-up tank into the above ground waste water hold-up tank in the west lot, from which all releases are monitored.

There are no floor drains which drain liquid effluent directly to the indoor hold-up tank or the external above ground waste water hold-up tank.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

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

The loss of AC power is considered during the experiment approval process. If continuity of electrical power is important for safety or experiment success, then it would be required to be available. In such cases a fail-safe arrangement is required as via an uninterruptible power supply or suitable alternative.

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

Please see attached SOP-A.5 REV 5, "Experiments" in Appendix C of this document.

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

The Rabbit System uses pressurized nitrogen gas to push samples into the core. The system utilizes polyethylene tubing that is roughly 1 inch inner diameter and approximately 1.3 inches outer diameter. The "rabbit" capsule is about 1 inch in diameter. The schematics for the system are attached as Figure 10-1 (10-4 in the July 2002 FSAR), 10-2 (10-5 in the July 2002 FSAR), and 10-3 (10-6 in the July 2002 FSAR)

The system is inspected prior to each use to preclude failures leading to stuck/immovable capsules. An empty test capsule is inserted prior to actual sample insertion to verify proper operation. Operation of the pneumatic sample transfer (Rabbit) system is controlled by the instructions in SOP-A.8, "Pneumatic Rapid Sample Transfer (Rabbit) System" which has been attached to this document in Appendix D

The removal of a stuck capsule may require dismantling of the applicable section of tubing and removal of the sample. There are multiple access points where a stuck capsule can be reached. If the sample becomes stuck inside the biological shield, then the portable shielding around the entry point on the west side of the reactor structure is removed and the entire rabbit system assembly itself removed (if necessary) under a Radiation Work Permit.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

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

The half life of N-16 is 7-seconds. Only exposure to Ar-41 was considered when calculating dose to the public and staff. Doses for staff members and public are re-evaluated based on continuous reactor operation in the process of HEU to LEU fuel conversion. The methodology and results are provided in Appendix E in this document.

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

The 200 to 1 dilution factor is based on the flow rate (1 to 400 cfm) of air withdrawn from the reactor cell and control room that is then required to be diluted by a flow rate of greater than 10000 cfm of outside air. The 200 to 1 atmospheric dilution factor is applied to the plume rising vertically form the stack.

This dilution factor is also recommended for Argonaut reactors in an NRC letter. Further calculations are performed using the Gaussian Plume Model. The results show that the maximum Ar-41 concentration in atmosphere is 2.98E-08 Ci/m^3 at 50m away from the stack under the worst weather condition (based on the weather archive at University of Florida) when reactor is operating at 100 kW. This concentration corresponds to an atmospheric dilution factor of 416. Therefore, the dilution factor (200 to 1) used in the SAR is very conservative. The description of methodology and the detailed calculation results are provided in Appendix E in this document.

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

The Radiation Control Officer does not supervise the actions of the RSR Subcommittee. The organization can be found in TS 6.2.

Thus, the last sentence in the third paragraph of Section 11.1.2.4 should read as follows:

> The Radiation Control Officer, an ex-officio member of the UFTR RSR Subcommittee, ensures that the Radiation Control Program objectives, guidelines and limitations are carried out at the UFTR facility.

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

The current byproduct materials license will not be changed. The special nuclear material license change was addressed by the HEU to LEU conversion SAR and related Tech Specs changes.

12 CONDUCT OF OPERATIONS

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

The originally attached UFTR Organizational Chart is the same as Figure 6.1 in the current UFTR Tech Specs. To clarify Figure 12.2 the required solid lines for reporting and dotted lines for communications are added in the attached updated

Figure 12.2 which also shows reporting responsibilities as requested. The updated Figure 12.2 is included in Appendix E of this document.

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

In Section 12.1.3 and TSs 6.1.3, "direction" is intended to mean active presence of the SRO. In the case of Section 12.1.3C(3) the qualifier in parentheses is intended to allow the SRO not to be present but to provide documented verbal concurrence for recovery from unplanned or unscheduled shutdowns. The only intended use of this qualifier would be for recovery from simple trips from an observed electrical power outage or similarly uncomplicated trip or unscheduled shutdown. Since this interpretation is not used at the UFTR facility, there is no need to include it. Therefore the word "presence" should replace "direction" in both places. In addition the qualifier in parentheses in Section 12.1.3 C(3) should also be removed.

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

The SAR should reference ANSI/ANS 15.4-1988 to agree with the TSs. The disagreement was an oversight.

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

The radiation safety staff as well as the operating staff have the ability and are trained that it is their duty to raise safety issues whenever they occur and then follow the management (reactor or radiation safety) chain up until their concern is addressed. Anyone on the staff can go directly to the Radiation Control Officer with a radiation safety issue as emphasized in training classes and in staff communications.

12-5 *Section12.1.5.1, Reactor Safety Review Subcommittee, and TS 6.2 Review and Audit. A quorum is defined as at least three members. But the membership if defined as at least five members. If there are more then six members a quorum of three would be less than half. The quorum should be at least three and at least half, also with the operating staff not constituting a majority (to meet ANSI/ANS 15.1). Also the Radiation Control Officer is referred to as both a member and an ex-officio member. Please address.*

A quorum for the RSR Subcommittee consists of at least three members. A quorum would be 4 members when the RSRS has 6 or 7 members. The operating staff does not constitute a majority. The RCO is an ex-officio member of the RSRS by his position and is also a voting member.

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

The last paragraph of Tech Spec 6.2.3 requires submission of the audit report to the Chairman of the University Radiation Control Committee. In addition the last paragraph of TS 6.2.4 requires deficiencies uncovered that affect reactor safety to be reported to the Chairman of the URCC and to the Dean of the College of Engineering (Level 1 management). The Chairman of the NRE Department is also at Level 1 and would also receive such reports by virtue of membership on the RSRS.

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

The requirements of 10 CFR 50.54(i) are met by the exceptions referenced in 10 CFR 55.13 to include: (1) using the UFTR as part of the education/training of students and (2) using the UFTR as part of training for individuals in the UFTR training program to qualify as operators under 10 CFR 55.

The requirements of 10 CFR 50.54(l) are met as follows; all NRC licensed senior reactor operators (SROs) are designated to be responsible for directing the licensed activities of licensed operators as part of their SRO duties. In addition all NRC licensed operators, in their enabling letter following licensing, are designated to be responsible for directing the operations activities of unlicensed individuals allowed to manipulate the reactor controls per the exemption allowed under 10 CFR 55.13.

13 ACCIDENT ANALYSIS

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

BORAX I and II results became obsolete and irrelevant, because new accident analysis is based on the SPERT-I reactor for the LEU core. Detailed analysis of the Sudden Insertion of the Maximum Allowed Excess Reactivity is provided in Section 13.1.3 p. 49 of the Fuel Conversion SAR. The comparison result is given in Table 13-5. Note that the excess reactivity allowed for the LEU core is 1.4% $\Delta k/k$ in the LEU core. For a step insertion of 1.4% $\Delta k/k$, the excursion energy would be less 6.1 MW-sec, and the cladding temperature would be lower than 300 *°C.* Both are well below the cladding melting temperature (Al-1100 cladding 660 *°C,* Al-6061 cladding 582 *°C*). They are also well below the current safety limit of fuel and cladding temperature 986 *°F* (*530 °C*). The analysis methodology is based on predicting the total energy release that results from the initial power spike before the transient is suppressed by the reactivity feedback effects. The feedback coefficient *b* in Equation 13.1 is better described as an 'energy conversion coefficient'. It is used to estimate the total release energy, and is determined from the SPERT-I test based on the inverse period of UFTR-LEU (Figure 13-3 in the Fuel Conversion SAR). Nevertheless, the LEU fuel has a much larger prompt fuel temperature (Doppler) feedback coefficient than the HEU fuel.

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

A core crushing accident (MHA) is considered in which the core is assumed to be severely crushed in either the horizontal or vertical direction by postulating that a [redacted]. The detailed analysis of the MHA is given in the Fuel Conversion SAR (p. 57). The dose calculation methodology and results are provided in Section 13.4.2 and Section 13.4.3 (Tables 13-12, 13-13 and 13-14) respectively.

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

The urban boundary and the public exposure have been evaluated for the Fuel Handling Accident (FHA) and the Maximum Hypothetical Accident (MHA). The urban boundary for the UFTR is set at 400m from the outside wall of the reactor. Exposures at a number of points of special interests are evaluated, including East Hall Housing (190m), Ben Griffin Stadium (230m), Weaver Hall Housing (250m), Riker Hall Housing (275m),O'Connel Center (300m), and Tolbert Hall Housing(310m). The thyroid and whole body doses at these points are given in Table 13-10 in the HEU to LEU Fuel Conversion SAR. The details of the analysis are included in Section 13.3 and 13.4 of the HEU to LEU Fuel Conversion SAR, respectively.

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

The accident analysis based on BORAX reactor is not relevant for the LEU core. The excursion energy is re-evaluated in Section 13.1.3 of the HEU to LEU Fuel Conversion SAR for the LEU fuel. Note that the actual maximum excess reactivity is set at 1.4%$\Delta k/k$ for the LEU core. And for a step insertion of 1.4%$\Delta k/k$ in the LEU core, the excursion energy would be less than 6.1 MW-sec. (Also see response to Question 13.1)

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

The statement applies to the original licensed power of 10 kW. The statement is obsolete since the current normal power level is 100kW. So it will be removed from the SAR,

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

Appendix 13-B became irrelevant. The effect of step insertion of *0.6%* $\Delta k/k$ reactivity has been re-evaluated for the LEU core. The analysis is given in Section

13.1.1 of the Fuel Conversion SAR.

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

The effect of a step insertion of *0.6%* $\Delta k/k$ reactivity has been re-evaluated using the RELAPS5-3D code. The analysis is given in Section 13.1.1 of the Fuel Conversion SAR.

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

Figure 13B-1 is from the BORAX experiment results. However, the accident analysis based on BORAX reactor became irrelevant for the LEU core. In the Fuel Conversion SAR, further analysis of the accident involving Sudden Insertion of the Maximum Allowed Excess Reactivity (Section 13.1.3, p49) is provided based on the SPERT I experiments because of the concern about the applicability of the BORAX experiments. (also see responses to questions 13.1 and 13.4)

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

Yes, they are defined in Table 13D-1. a_1 and a_2 are experimentally determined constants.

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

The Appendix is removed. Thermal hydraulics considerations are addressed in the HEU to LEU Fuel Conversion SAR.

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

See answer to Question 13-10

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

The graphite temperature is assumed to be equal to the average coolant temperature (307.8 K) in the LEU core. The validation of this assumption is included in Appendix A1 (p. 69) of the Fuel Conversion SAR.

14 TECHNICAL SPECIFICATIONS

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

TS 3.2.1(4) states that the control-blade-drop time shall not exceed 1.5 sec from initiation of blade drop to full insertion. This is required to be met as a limiting condition for operation prior to startup based upon the last measurement of the parameter in question within the TS required interval. This is the case for any of the LCOs, which are not specially indicated to be verified prior to starting up the reactor as part of the pre-operational check.

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

All the definitions listed in the Tech Specs are used in the documents.

14-3 Section 2.0, Safety Limits and LSSS. As noted in the guidelines contained in NUREG-1537, Part1, Appendix 14.1, Section 2.1.3, "... For plate type fuel... the applicant should determine a fuel cladding temperature below which cladding damage (softening or blistering) can be precluded. The applicant should then establish a corresponding power level, reactor conditions, and uncertainties that limit cladding temperature below the damage limit."

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

The existing safety limit on fuel and cladding temperature below 986 *°F* is justified in the HEU to LEU Fuel Conversion SAR to provide adequate margin to protect the fuel and the cladding. Therefore, no safety limit is needed on the height of water above the fuel elements.

Section 2.1, Safety limits, states the following specifications to ensure fuel cladding integrity:

(1) The steady-state power level shall not exceed 100 kWth.
(2) The primary coolant flow rate shall be greater than 18 gpm at all power level greater than 1 watt.
(3) The primary coolant outlet temperature from any fuel box shall not exceed 200 *°F*
(4) The specific resistivity of the primary coolant water shall not be less than 0.4 megohm-cm for periods of reactor operations over 4 hours.

Specifications (1), (2) and (3) were changed as part of analysis for the HEU to LEU conversion (Section 4.7). They are replaced by one specification in the current TS:

(1) The fuel and cladding temperatures shall not exceed 986 *°F*.

This specification explicitly requires the fuel temperature to be below 986 *°F* (*530 °C*) as justified and accepted by NRC in the HEU to LEU conversion SAR for the silicide plate type fuel. Since exceeding the limit on primary coolant resistivity does not lead to immediate fuel clad damage, Specification (4), the second safety limit in the current TS, is removed as requested. The existing limit on the primary coolant resistivity will continue to be controlled as a limiting condition for operation.

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

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

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

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

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

LSSS 2.2(6) The primary coolant pump shall be energized during reactor operation.

LSSS 2.2(7) The primary coolant flow rate shall be monitored at the return line

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 operation

The LSSS 2.2 (6) is moved to LCO section. 3.2.1 (8)

LSSS 2.2(7) is changed as follows to provide a faster and more conservative trip, before LSSS 2.2 (2) triggers a trip for loss of normal flow rate.

The primary coolant flow rate at the return line shall be greater than zero.

LSSS 2.2(10) is moved to LCO section 3.2.1 (7).

LSSS 2.2(11) is changed as follows to assure monitoring a process variable. And it will remain in the LSSS section as LSSS 2.2(9).

The dilution fan rpm indication shall be 95% or more of the established normal value.

Note that currently approved LSSS 2.2(11) causes the reactor to trip when the evacuation siren secures the reactor vent system. This trip was installed in response to an NRC request following a reportable event involving concern for release of radioactive material in the reactor cell and potentially to the environment.

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

The excess reactivity limit has been changed to *1.4% Δk/k (See UFTR HEU to LEU conversion SAR, Chapter 14)* as approved by NRC.

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

All of the safety system operability tests are conducted. Many of them are conducted by temporarily increasing the setpoint of the trips, then testing the trip function under the increased setpoint. Others are checked by verifying the conditions required to yield the requisite trips. Loss of electrical signal does not affect the safety system operability as the loss of electrical signal results in a trip in all cases. The trips are all checked as part of the pre-operational checks or as part of the Q-1 (quarterly) scram checks. The current Q-1 surveillance data sheets for documenting the quarterly checks are included as Appendix F for future reference.

How does loss of primary coolant flow show operability of the low inlet water flow?

UFTR Quarterly (Q-1) Surveillance Step 5.a is applicable. In this quarterly test, the operability of primary coolant flow loss trip is verified to produce a trip when the scram setpoint is raised to the coolant flow point on the primary coolant flow meter.

How does loss of primary coolant level show operability of the low water level in core safety system trip?

UFTR Quarterly (Q-1) Surveillance Step 4 is applicable. This safety system operability is verified by a quarterly test. In this test, the primary coolant pump is shut down (the corresponding trip is bypassed). Then, the operability of the low level trip is tested by opening the dump valve to permit the primary coolant system to drain down and recording the level at which the trip occurs.

How does the loss of shield tank water level show shield tank low water level?
UFTR Quarterly (Q-1) Surveillance Step 7 is applicable. This operability test is conducted quarterly. In this test, the shield tank cover is removed so that the water level can be measured directly. The shield tank low water level trip is tested by

slowly raising the level detector to the setpoint and checking the measured water level with the level indicated on the detector.

These scram checks are required on a quarterly basis. The detailed step by step test procedure can be found in UFTR Quarterly (Q-1) Surveillance data sheets, which are attached in Appendix F in this document.

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

The 6-hour time period is a typical work shift time. This period limitation applies even if the reactor is secured. These tests confirm the requisite trip function of the control blades as well as the applicable LCOs as part of a satisfactory completion of the daily pre-operational checkout.

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

LCO 3.8 (2) states:
Primary coolant water shall be demineralized, light water with a specific resistivity of not less than 0.5 megohm-cm after the reactor is operated for more than 6 hr.

The 6 hr period is used to account for the effects while the reactor is in a transient state, and allow for expected transient changes in the water resistivity. Following maintenance entries into the primary coolant system, temperature transients upon first reaching high temperature during power runs can cause temporary increases in resistivity. Under normal operating conditions, the resistivity is usually above 1.0 megohm-cm.
The coolant pH value is measured to meet the current LCO 3.8 (5), which states:

Primary water pH value shall be <7.0.

This LCO was incorporated as part of the HEU to LEU conversion analysis. The pH value is verified as part of the weekly pre-operational checkout. No controls have been necessary due to the nature of the makeup water source and the primary coolant system. Measurements have always shown pH values are well below 7.0 for nearly 2 years.

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

The count rate of 4000 cps represents a minimal hazard, but it indicates an abnormal condition of the reactor and/or reactor systems including the cell air volume. The normal equilibrium count rate at full power (100 kW) is around 2500-3000 cps. The equilibrium value of this count rate is proportional to the reactor power. The allowable release of activated Argon-41 is controlled to assure to meet 10 CFR Part 20 release limits to unrestricted areas. The periodic measurement of the Argon-41 release level at full power is based upon the normal equilibrium count rate at full power. Therefore, the 4000 cps level would indicate a significant change in the level measured. The calibration of the stack monitor is checked periodically to assure accounting for any changes in the detector efficiency.

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

Section 3.6 (4) has been updated:

(4) Explosive Materials
Explosive materials shall not be irradiated in the core unless the irradiation container has the ability to contain the energy released by 200% if the explosive is detonated. Following an adequate safety analysis, explosive materials may be irradiated using beams extracted from the core as well.

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

The fission product inventory is re-evaluated based on the Fuel Handling Accident (FHA) in the HEU to LEU Fuel Conversion SAR (Section 13.3 p. 52). The calculated FHA radionuclide inventories are listed in Table 13-6 of the Fuel Conversion SAR. The dose calculation results are provided in Section 13.3.3. Based on these analyses, the FHA yields very low occupational and public dose exposure. The same results would be true for fueled experiments. And the upper limit on the allowable fission product inventory in a fueled experiment shall be the same.

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

Besides the primary water radioactivity weekly test, the primary coolant resistivity meter alarms if the water resistivity goes below the setpoint on the demineralizer system inlet from the primary coolant system. Therefore, the demineralizer can be used to indicate fuel failure too, since fuel failure would be expected to cause unexpected lowered resistivity. Further tests can then be conducted on primary coolant water to confirm or eliminate the presence of fuel failure. Specific failed fuel bundles can only be identified by removing the reactor shielding to access the core area. At that point, individual fuel bundles can be removed and examined visually for defects. Individual bundles could also be placed in water to test for fission products indicating fuel failure. The identification of a specific failed fuel bundle would be a time-consuming process, especially if there was no visible damage indicated on the fuel bundle.

The air particulate detector (APD)/continuous air monitor (CAM) system in the reactor cell would detect significant releases of fission products from the fuel at the time of occurrence before release to the environment. The same would be true of the stack monitor system. However, it should be noted that it is more difficult for such fission products to get out of the primary coolant system piping and tank arrangement than in a pool reactor.

Appendix A: Schematic of UFTR Secondary Water Cooling System. (in response to Q 5-1)

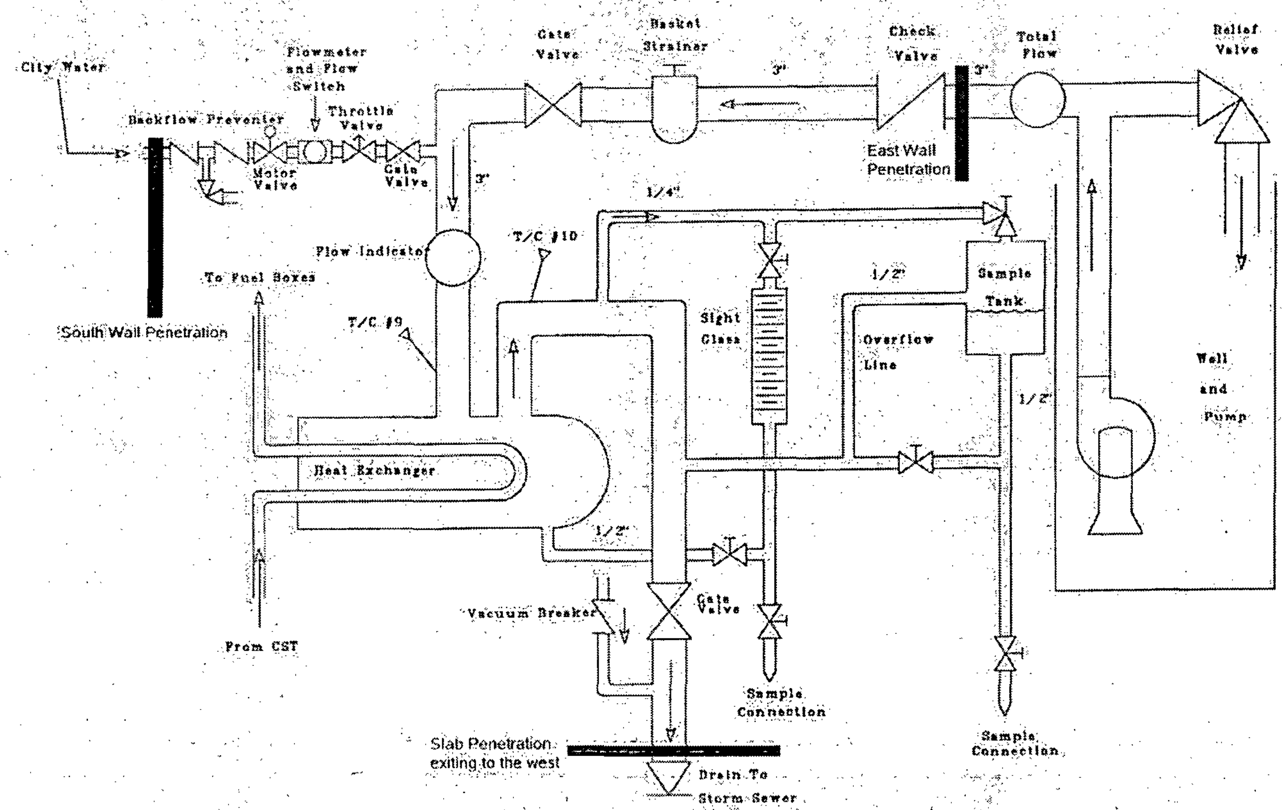
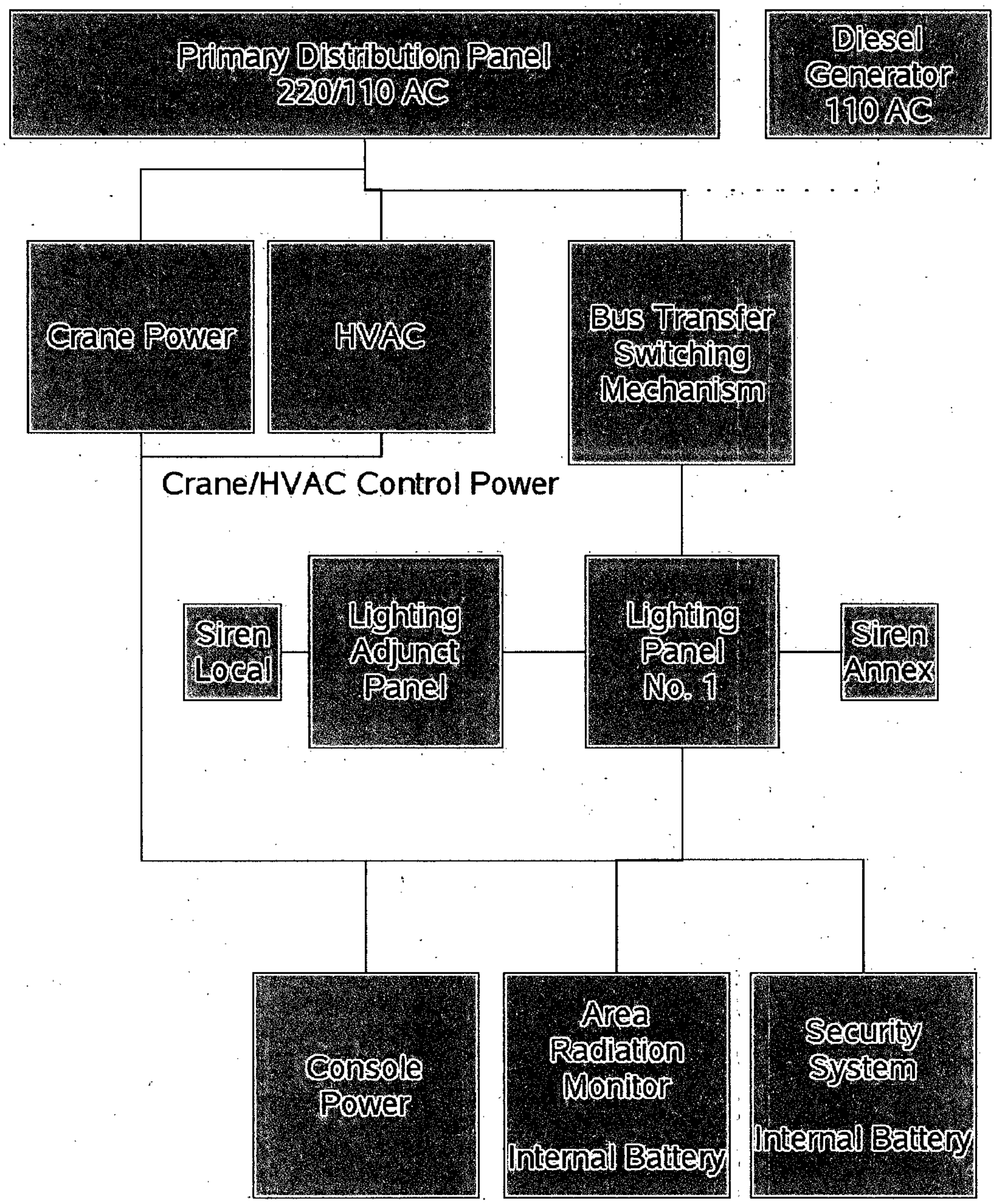


Figure 5-4 Schematic of UFTR Secondary Water Cooling System.

Appendix B: Power distribution drawing. (in response to Q 8-1)



Appendix C SOP-A.5 REV 5, "Experiments (In response to Q 10-2)

UFTR OPERATING PROCEDURE A.5

1.0 Experiments

2.0 Approval

Reactor Safety Review Subcommittee.. ______________________ Date __________

Director, Nuclear Facilities .... ______________________ Date __________

3.0 Purpose and Discussion

3.1 The purpose of this procedure is to assure that experiments receive sufficient review and care in performance to assure operational safety and prevent damage to the reactor facility, reactor fuel, reactor core, and associated equipment; to prevent exceeding the reactor safety limits; and to minimize potential hazards from experimental devices.

3.2 It is recommended that the Principal Investigator or his/her representative discuss the experiment with the reactor staff prior to submitting UFTR Form SOP-A.5A (Request for UFTR Operation). The reactor staff will make recommendations on the feasibility of the experiment and on specific limitations which may apply to the experiment as planned by the investigator.

4.0 Limits and Precautions

4.1 Reactivity Limits

4.1.1 Absolute reactivity of any single moveable or non-secured experiment shall not exceed 0.6% $\Delta k/k$.

4.1.2 Total absolute reactivity worth of all experiments shall not exceed 1.4% $\Delta k/k$.

4.1.3 When determining absolute reactivity of an experiment, no credit shall be taken for temperature coefficients.

4.1.4 An experiment shall not be inserted or removed unless all control blades are inserted or unless its absolute reactivity worth is less than that which would cause positive 20 second stable period.

4.2 Explosive materials shall not be irradiated.

4.3 Experiments shall be designed so that during normal operation or possible failure:

4.3.1 The thermal hydraulic parameters of the core do not exceed the safety limits.

4.3.2 Fuel cladding integrity shall not be compromised by either chemical or blast effects from the experiment.

4.4 A limit should be established on the inventory of fission products in each experiment containing fissile material according to its potential hazard and as determined by the Reactor Safety Review Subcommittee (RSRS).

4.5 For Class III and IV experiments, a limit on the permissible concentration of radioisotopes should be established by the RSRS according to the potential for airborne releases at greater than the allowed legal limits.

4.6 Experimenters handling radioactive materials should have a University of Florida Radioactive Materials User Statement of Training and Experience (RC-1 Form) on file with the Radiation Control Office.

4.7 UFTR SOP-D.4, "Removing Irradiated Samples from UFTR Experimental Ports" shall be used to control removal of irradiated material from the core.

4.8 UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers" shall be used to control radioactive material transfers between the UFTR R-56 License and the University of Florida 356-1 State License.

5.0 References

5.1 UFTR Technical Specifications

5.2 UFTR SOP-B.1, "Radiological Emergency"

5.3 UFTR SOP-D.1, "UFTR Radiation Protection and Control"

5.4 UFTR SOP-D.2, ARadiation Work Permit@

5.5 UFTR SOP-D.4, "Removing Irradiated Samples from UFTR Experimental Ports"

5.6 UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers"

5.7 UFTR SOP-E.2, "Alterations to Reactor Shielding and Graphite Configuration"

5.8 10 CFR Part 20, "Standards for Protection Against Radiation"

5.9 Guide for Irradiations in the UFTR

5.10 Guide for Production of Short-Lived Isotopes in the UFTR

5.11 Radiation Control Manual

5.12 Radiation Control Techniques

5.13 Policy Statement for Transfer of Radioactive Materials Between the UFTR R-56 License and the University of Florida 356-1 State License

6.0 Records Required (as Applicable)

6.1 UFTR Operations Log

6.2 UFTR Form SOP-A.5A (Request for UFTR Operation)

6.3 Radiation Control Form RC-1 (University of Florida Radioactive Materials User Statement of Training and Experience)

6.4 UFTR Form SOP-D.4A (Record of Sample Irradiation and Disposition)

6.5 UFTR Form SOP-D.4B (Sample Record Index)

6.6 UFTR Form SOP-D.6A (University of Florida Training Reactor/University of Florida Radioactive Material Transfer Record)

6.7 UFTR Form SOP-D.6B (University of Florida/University of Florida Training Reactor Radioactive Material Transfer Record)

6.8 UFTR Form SOP-D.6C (University of Florida Training Reactor/University of Florida Activated Foil Transfer Record)

6.9 UFTR Form SOP-D.6D (University of Florida/University of Florida Training Reactor Neutron Radiography Film Cassette Transfer Record)

6.10 UFTR Form SOP-D.6E (University of Florida/University of Florida Training Reactor Rabbit System Sample Package Transfer Record)

7.0 Instructions

7.1 Definitions

7.1.1 *Experiment* shall mean any apparatus, device, or material installed in the reactor core or experimental facilities which is not a normal part of the reactor core or experimental facilities.

NOTE: For the purposes of UFTR SOP-A.5, a reactor experiment needing an approved run request form, UFTR Form SOP-A.5A (Request for UFTR Operation), is defined as whenever the reactor is started up, achieves criticality, or is operated at any power level with any equipment, apparatus, or material inserted into or modifying the core, thermal column, shield tank, or any of the reactor control, safety, coolant, or auxiliary systems (exclusive of standard startup sources) for the purpose of conducting a research experiment, class or training experiment, or test of reactor behavior.

Additionally, any reactor run made for a class experiment or exercise that is not exclusively of a reactor training nature, is an experiment and hence should have an approved run request form on file.

7.1.2 Experiments are classified in four (4) categories (see UFTR Technical Specifications) as follows:

Class I – Routine experiments such as gold foil irradiation and other routine established exercises.

Class II – Relatively routine experiments which need to be documented for each new group of experimenters performing them, or whenever the experiment has not been carried out for one calendar year or more by the original experimenter, and which pose no hazards to the reactor, the personnel or the public.

Class III – Experiments which pose significant questions regarding the safety of the reactor, the personnel, or the public.

Class IV – Experiments which have a significant potential for hazard to the reactor, to personnel, or to the public.

NOTE: The operation of a Class III or Class IV experiment requires RSRS approval of an analysis of associated operational characteristics, hazards and safety considerations (including emergency procedures and steps to mitigate the consequences of potential accidents as delineated in Section 7.2.2.3).

7.2 Requesting Use of the Reactor for an Experiment

7.2.1 A properly completed and approved UFTR Form SOP-A.5A (Request for UFTR Operation) shall be filed in the Run Request Book in the Reactor Control Room prior to operating the reactor for an experiment; no experiment may be performed without proper approval via UFTR Form SOP-A.5A.

NOTE: UFTR Form SOP-A.5A (Request for UFTR Operation) may also be used to document reactor usages which are not technically experiments and hence which do not require this form; such documentation may be used at the discretion of facility management.

7.2.1.1 All new experiments shall require an initial, detailed run request (UFTR Form SOP-A.5A) to be generated and approved.

7.2.1.2 A comprehensive run request for a series of well-established experiments may be generated to cover a group of experiments selected from a list of all standard Class I and Class II experiments previously performed in the reactor. The comprehensive run request (UFTR Form SOP-A.5A) must identify all approved experiments by full experiment title, date last run, and expected date the experiment will be run under the comprehensive request.

7.2.1.3 Each different class or group utilizing a comprehensive run request for a series of experiments will require a separate run request (UFTR Form SOP-A.5A) for its comprehensive set of experiments.

7.2.2 Run Request Preparation

7.2.2.1 The UFTR Form SOP-A.5A (Request for UFTR Operation) contained in Appendix I must be filled out and approved in its entirety prior to experiment insertion. The experimenter may utilize assistance from other sources as necessary to complete the run request form.

7.2.2.1.1 The experimenter must provide a brief summary of the proposed experiment detailing the names of the experiments, source of support, experiment objectives and significance of the experiment. The experimenter should also provide information on expected publications. If the format of UFTR Form SOP-A.5A allows sufficient space, this information may be included on the run request form under Section I (Introductory Administrative Information).

NOTE: This information is needed for the UFTR Annual Report.

7.2.2.1.2 The experimenter, with UFTR staff support as necessary, should also complete Section II (Operational Information) of UFTR Form SOP-A.5A giving the operational information for running the experiment.

7.2.2.1.3 The principal investigator, co-investigator or a designated alternate should sign under Section III (Experimenter Approval) of UFTR Form SOP-A.5A indicating the information given in Sections I and II of the run request form is correct to the best of the individual's knowledge. Students authorized by the experimenter to work with the experiment are to be listed in Section III as well.

7.2.2.1.4 The experimenter, with UFTR staff or other support as necessary, must complete Section IV (Irradiation Information) of UFTR Form SOP-A.5A to assure adequate control over materials inserted in the UFTR experimental ports and to assure proper controls and preparations for experiment removal after irradiation. Limiting experiment parameters are designated in Section IV via Table A.5A which is used to track any applicable limiting parameter(s) as an experiment is run one or more times.

7.2.2.1.5 The experimenter with other support as necessary must also complete Section V (Personnel Limitations and Requirements) of UFTR Form SOP-A.5A; only those individuals listed as authorized to remove radioactive material from the UFTR cell will be allowed to do so under the limitations and documentation requirements of UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers." The final decision about whether radiation control supervision will be required outside the cell and whether radioactive material will be allowed to leave the reactor cell, Nuclear Reactor Building, or Nuclear Sciences Building must be made by the Reactor Manager and Radiation Control Officer in approving the experiment with input from the experimenter providing the necessary supporting information.

7.2.2.1.6 The Reactor Manager or Facility Director shall assure that Section VI (Experiment Approval Requirements) of UFTR Form SOP-A.5A is completed to include indicating whether a new proposal is needed, assigning the Experiment Category (see Section 7.2.2.2) and indicating whether RSRS approval is required. All other blanks in Section VI are completed by signatures and dates of those approving the experiment.

NOTE: New proposals will normally only be needed for experiments of a type not run previously or for which detailed information is needed. The information supplied in Section I, Part (1) will be sufficient for most routine experiments so a full proposal will not be needed.

7.2.2.1.7 Section VII (Experiment Modification) of UFTR Form SOP-A.5A must be completed by the Reactor Manager or Facility Director and the Radiation Control Officer for all changes on the run request after original approval.

NOTE: Significant changes on a Class III or Class IV experiment run request will require RSRS approval.

7.2.2.2 Requirements for Experiment Approval

7.2.2.2.1 *Class I* experiments shall be approved by the Reactor Manager; Class I experiments should also be approved by the Radiation Control Officer.

7.2.2.2.2 *Class II* experiments shall be approved by the Reactor Manager and the Radiation Control Officer.

7.2.2.2.3 *Class III* experiments shall be approved by the Reactor Manager and the Radiation Control Officer after review and approval by the RSRS.

7.2.2.2.4 *Class IV* experiments shall be approved by the Reactor Manager and the Radiation Control Officer after review and approval by the RSRS. Specific emergency operating instructions shall be established for conducting such experiments.

7.2.2.3 Special instructions apply for approval of Class III and Class IV experiments; at least three (3) working days prior to the RSRS meeting at which the proposed experiment is to be reviewed, the experimenter, as a minimum, must submit the following information to the Reactor Manager for transmittal to RSRS members:

A. Experiment Description to include:

1. Detailed description of the experiment and its apparatus,

2. Operational and/or safety considerations including the use or production of hazardous material and associated radiation and/or chemical hazards.

   NOTE: A copy of "Hazardous Chemicals Desk Reference," N. Sax, R. Lewis, is normally available at the facility for use in evaluating hazards.

3. Estimated potential effects on reactor reactivity and UFTR operational characteristics.

B. Experimental Procedures for Three Experimental Phases:

1. Preparatory to reactor operation,

2. During reactor operation,

3. Subsequent to reactor operation.

C. Analysis of maximum credible accident(s) associated with the experiment.

D. Emergency procedures and steps proposed to minimize the probability of such accidents and to mitigate the consequences of postulated accidents.

E. A one-page summary of the experiment including the names of the experimenters and source of support as well as objectives and significance of the experiment. The information in this summary is needed for submittal to the U.S. Nuclear Regulatory Commission and to the U.S. Department of Energy as part of the UFTR Annual Report and may be included on the Request for UFTR Operation (UFTR Form SOP-A.5A).

F. Additional information as applicable to aid the RSRS in their evaluation and the operating staff in preparing for the experiment.

7.3 Running the Experiment

7.3.1 Reactor Operations Staff Duties for Experiment Insertion

7.3.1.1 Before any experiment is installed in the reactor, the UFTR operator responsible for running the reactor shall assure that a properly approved UFTR Form SOP-A.5A (Request for UFTR Operation) has been filed and that the planned operation does not exceed any of the limits placed upon the experiment via the approved run request.

7.3.1.1.1 The UFTR reactor operator shall assure understanding of the experiment and its run request including any limitations.

7.3.1.1.2 If the experiment has not been run previously, the reactor operator shall assign and enter a ARun Request Number@ and ADate of Initial Experiment Run@ on the first page of UFTR Form SOP-A.5A; the reactor operator should also record the applicable experiment information on the list of experiments run year to date.

7.3.1.1.3 The reactor operator shall assure the planned operation will not exceed any limiting parameters delineated in Table A.5A on the last page of UFTR Form SOP-A.5A, by checking the margin on the ALimiting Experiment Parameter(s)@ listed in Table A.5A.

NOTE: If there are no special limiting parameters, then Table A.5A is not required.

7.3.1.2 Requirements for running Class III and Class IV Experiments

7.3.1.2.1 Class III and Class IV experiments will be inserted, disassembled or removed from the reactor only under the direct supervision of the Reactor Manager or a duly authorized representative.

7.3.1.2.2 All personnel present in the reactor cell during the running of Class III and Class IV experiments must be familiar with the Radiological Emergency procedures (UFTR SOP-B.1 and UFTR SOP-B.2) and shall be certified as such using UFTR Form SOP-B.1. The only allowed exceptions are visitors to the control room and up to two individuals in the cell accompanied by a member of the UFTR technical staff who is certified as second person qualified.

7.3.1.2.3 All gases which may cause a hazard through neutron activation shall be exhausted from experiments or experimental facilities installed in or near the core or surrounding graphite to the environment via the Reactor Vent System.

7.3.1.3 Adjustments and alterations to permanent and temporary shielding as well as alterations to graphite configurations shall be controlled via UFTR SOP-E.2, "Alterations to Reactor Shielding and Graphite Configuration."

7.3.1.4 Radiation surveys following shielding adjustments as well as alterations to graphite configurations shall be controlled via UFTR SOP-E.2, "Alterations to Reactor Shielding and Graphite Configuration."

7.3.1.4.1 Normally when radiation surveys are deemed necessary, it will be adequate to check experiment radiation levels first at one kW and to extrapolate one decade. As power is raised one decade, the extrapolation should be confirmed and the process repeated until the desired power level is achieved.

7.3.1.4.2 If any permanent reactor shielding has been removed or displaced in preparation for insertion of an experiment:

A. A representative of the Principal Investigator(s) and the Radiation Control Office must be present during initial insertion of the experiment.

B. Radiation Control Personnel will monitor and document radiation levels following initial startup and during any initial subsequent increases in power level.

7.3.2 Reactivity Considerations for Experiments

7.3.2.1 An experiment shall not be inserted or removed from the core unless one of two conditions is met:

7.3.2.1.1 All control blades are fully inserted, or

7.3.2.1.2 The experiment's absolute reactivity worth is known to be less than that which could cause a 20 second positive stable period.

7.3.2.2 During startup, the total reactivity worth of an experiment will be determined and logged in the UFTR Operations Log by the following method:

A. Determine the 1 watt stable critical position without the experiment inserted either from the current run (preferred) or from the best recent clean startup.

B. Determine the 1 watt stable critical position with the experiment inserted.

C. Using blade worth curves, determine the total reactivity worth associated with the difference in blade positions at 1 watt with the experiment removed and with the experiment inserted.

NOTE: If xenon buildup or other reactivity effects prevent a determination of experiment worth based on the best recent clean startup, then the experiment worth should be determined by determining the present 1 watt critical position without the experiment followed by shutdown, experiment insertion and restart to 1 watt.

D. Record the reactivity worth of the experiment in the UFTR Operations Log; the reactivity worth may also be recorded on the run request (UFTR Form SOP-A.5A) for future reference. This record should be made if the reactivity worth is significantly different from that expected.

7.3.3 Generation of short-lived sample activity for purposes such as neutron activation analysis may be facilitated by the use of the Rabbit Sample Transfer System.

NOTE A: All samples inserted via the rabbit system are known to have reactivity worth less than that which could cause a 20 second positive stable period provided the material is not fuel material.

NOTE B: Reactivity worths of samples inserted via the rabbit system are determined by estimating the regulating blade position before and after insertion, determining the reactivity difference using the regulating blade reactivity worth curve and recording the resultant reactivity worth in the UFTR Operations Log.

7.3.4 The reactor operator should update Table A.5A (UFTR Record of Irradiation and Experimental Parameter Limitations) as applicable prior to removing the experiment from the reactor. The reactor operator shall update Table A.5A as applicable prior to the next insertion of the experiment.

7.4 Removing Radioactive Material from UFTR Experimental Facilities

7.4.1 Any radioactive material producing radiation levels greater than 200 mR/hr on contact shall be placed in a shielded container or otherwise shielded to reduce potential exposure.

7.4.2 Any radioactive material producing radiation levels greater than 200 mR/hr on contact with its shielded container shall not be removed from the UFTR without prior consent of Radiation Control.

7.4.3 The controlling UFTR SOPD.4 , "Removing Irradiated Samples from UFTR Experimental Ports" shall be used to control removal of samples and other activated and/or contaminated materials from UFTR experimental ports to include samples removed via the Rabbit System.

7.4.4 Two qualified individuals shall monitor experiment removal from the UFTR.

7.4.4.1 A licensed UFTR staff member must monitor removal of all experiments.

NOTE: Remote monitoring at the console is considered to meet this requirement for samples removed via the Rabbit System.

7.4.4.2 Radiation Control will monitor removal of all experiments and samples from the reactor. A Radiation Control representative must be present (radiation control qualified UFTR personnel are considered acceptable) for sample removal from all ports in the reactor cell. For sample removal via the Rabbit System, a certified rabbit system operator is acceptable.

NOTE: Items that are parts of experiments but only subject to beams extracted from the reactor may be removed by a single person and without reactor power reduction provided that proper monitoring is performed.

7.5 Releasing Radioactive Material from the UFTR R-56 License

7.5.1 Implementation of UFTR SOP-D.6 shall be in conformance with the "Policy Statement for Transfer of Radioactive Materials Between the UFTR R-56 License and the University of Florida 356-1 State License."

7.5.2 The controlling UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers" shall be used to document the release and transfer of radioactive material from the UFTR R-56 License to the University of Florida 356-1 State License.

7.5.3 If a radioactive sample on the UFTR R-56 License is to be removed from the UFTR site then UFTR Form SOP-D.6A (University of Florida Training Reactor/University of Florida Radioactive Material Transfer Record) or in certain cases shortened forms represented by UFTR Form SOP-D.6C (University of Florida Training Reactor/University of Florida Activated Foil Transfer Record) for foils, UFTR Form SOP-D.6D (University of Florida/University of Florida Training Reactor Neutron Radiography Film Cassette Transfer Record) for the neutron radiography film cassette, or UFTR Form SOP-D.6E (University of Florida/University of Florida Training Reactor Rabbit System Sample Package Transfer Record) for samples irradiated via the Rabbit System must be completed to control and document the transfer.

7.5.4 If radioactive samples or other material on the University of Florida 356-1 State License is to be transferred to the UFTR R-56 License for irradiation, then UFTR Form SOP-D.6B (University of Florida/University of Florida Training Reactor Radioactive Material Transfer Record) or in certain cases shortened forms represented by UFTR Form SOP-D.6C (University of Florida Training Reactor/University of Florida Activated Foil Transfer Record) for foils, UFTR Form SOP-D.6D (University of Florida/University of Florida Training Reactor Neutron Radiography Film Cassette Transfer Record) for the neutron radiography film cassette, or UFTR Form SOP-D.6E (University of Florida/University of Florida Training Reactor Rabbit System Sample Package Transfer Record) for samples irradiated via the Rabbit System must be completed to control and document the transfer.

Appendix D SOP-A.8, "Pneumatic Rapid Sample Transfer (Rabbit) System" (in response to Q 10-3)

UFTR OPERATING PROCEDURE A.8

1.0 Pneumatic Rapid Sample Transfer (Rabbit) System

2.0 Approval

Reactor Safety Review Subcommittee.................. ______________________ __________
Date

Director, Nuclear Facilities .... ______________________ __________
Date

3.0 Purpose and Discussion

3.1 This procedure provides instructions for operation of the Pneumatic Rapid Sample Transfer (Rabbit) System to assure proper reactor and radiological controls and documentation are implemented when the Rabbit System is used for irradiations.

3.2 The Pneumatic Rapid Sample Transfer (Rabbit) System is used to support experimental irradiations in the UFTR:

3.2.1 In accordance with UFTR SOP-A.5, "Experiments," and provided properly completed UFTR Form SOP-A.5A (Request for UFTR Operation) authorizing the use of the rabbit system has been approved for the experiment;

3.2.2 In accordance with UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers"; and

3.2.3 In accordance with UFTR SOP-D.4, "Removing Irradiated Samples from UFTR Experimental Ports."

4.0 Precautions and Limits

4.1 Persons shall be certified to use the rabbit control system to make sample insertions into the UFTR only if:

4.1.1 They have read and understood the contents of this procedure as demonstrated by trial runs on using the rabbit system; and

4.1.2 Training on how to operate the rabbit system has been completed and the individual has been certified as qualified to operate the rabbit system as evidenced by a completed and approved UFTR Form SOP-A.8A (Rabbit System Operator Certification) on file for the prospective rabbit system operator.

NOTE: This FORM SOP-A.8A (Rabbit System Operator Certification) requires the name and signature of the prospective rabbit system operator.

4.2 Communications with the Console

4.2.1 The installed communications systems shall be used to obtain approval of the reactor operator prior to accomplishment of any of the steps of normal operation as well as to inform the reactor operator of any equipment, personnel or radiological abnormality encountered during operation of the pneumatic sample delivery system.

4.2.2 Sample insertions shall not be made without appropriate informative communications with and approval of the reactor operator in the control room.

4.2.3 The communications system should be used to keep the reactor operator at the console fully informed of any abnormality as well as the progression of normal activities.

4.3 Nitrogen Propellant

4.3.1 Unregulated nitrogen supply pressure should be at least 200 psi.

4.3.2 Nitrogen pressure downstream of the regulator should be controlled at about 25 psi (not to exceed 50 psi) and should not be less than 20 psi.

4.3.3 Nitrogen reservoir bottle should be physically secured at all times except while removing or replacing the bottle.

4.4 An empty sample holder should be inserted cap side down into the receiving station (after informing the control room operator) for insertion into the reactor for a short time and returned as a check on system operation prior to sample insertion.

4.5 Capsule Return Backlit Push-button Functions

4.5.1 "Capsule Return" lamp indicates capsule is in receiving station at end of manual or automatic cycle of operation.

4.5.2 "Capsule Return" push button allows manual control of capsule return portion of manual or automatic operation of rabbit system sample irradiation cycle.

4.5.3 If capsule has not returned normally to the capsule receiving station, "Capsule Return" push button permits manual actuation of return gas pressure.

4.5.4 "Capsule Return" push button permits release of gas pressure from the system with gas bottle supply valve shut, regulator diaphragm under tension (attempting to maintain downstream pressure) and gas supply control solenoid valve energized.

4.6 Personnel Radiological Controls for Rabbit System Operations

4.6.1 Prior to operations with the rabbit system:

4.6.1.1 The applicable, completed and approved UFTR Form SOP-A.5A (Request for UFTR Operation) should be checked by the reactor operator to assure any special instructions for the rabbit system operator are followed;

4.6.1.2 Certification of the rabbit system operator should be verified by the reactor operator to include having a completed UFTR Form SOP-A.8A (Rabbit System Operator Certification) on file;

NOTE: UFTR Form SOP-A.8A requires the signature of the rabbit system operator candidate as well as the rabbit system/reactor operator trainer and the Reactor Manager or Facility Director before certification is complete.

4.6.1.3 Portable survey instrumentation should be stationed near the rabbit system glove box to check activity/dose rates from irradiated samples;

4.6.1.4 The two detectors used to monitor and survey samples in the receiving station or glove box should be recorded in the Daily Operations Log by the reactor operator at the console; and

4.6.1.5 Dosimeter and/or ring badge should be attached to the rabbit system operator's extremities used in the glove box if significant dose is expected.

4.6.2 Gloves should be worn at all times by personnel handling rabbit capsules.

4.6.3 Excessive contamination of rabbit capsules and/or irradiated samples should be avoided. The following controls apply:

4.6.3.1 Any rabbit capsule in the control station glove box should be visually checked for integrity prior to each use; capsule should be cleaned and/or replaced, as necessary.

4.6.3.2 The number of capsules in use with the rabbit system should be minimized to facilitate radiological controls.

4.6.4 Rabbit capsules should not routinely be removed from the glove box except as required for sample loading/unloading, checks for integrity or disposal.

4.6.5 Samples reading $\geq$200 mR/hr on contact with the receiving station shall not be removed from the receiving station without permission of the Reactor Manager and Radiation Control personnel so that additional container shielding can be used for transfer.

CAUTION

Samples or other materials reading $\geq$200 mR/hr on contact (with shielded container) shall not be transferred between the UFTR R-56 License and the State 356-1 License.

4.6.6 Samples reading $\geq$1.5 R/hr at 1 foot unshielded shall not be removed from the glove box without permission of the Reactor Manager and Radiation Control Officer.

CAUTION

Specific instructions delineated on an approved Run Request (UFTR Form SOP-A.5A) may supercede this limit and may dictate much more restrictive levels which shall be the effective control on removing samples.

4.6.6.1 The detector (preferably a GM counter) attached to the rabbit system glove box is normally calibrated to indicate about 1000 cpm per 50 mR/hr; so samples in the rabbit receiving station should indicate approximately 1000 cpm on the monitor at 50 mR/hr contact. However, it is important to remember that this correlation is very approximate as it depends strongly upon the nature of the radioactivity involved.

4.6.6.2 For informational purposes, Table 1 in Appendix II contains data correlating contact dose rates measured with GM Detector Eberline E-530, Serial Number 1879, compared to readings indicated on the rabbit system glove box monitoring GM Detector Eberline RM-20, Serial Number 361.

4.6.6.3 The monitoring function performed using the glove box detector and the handheld survey instrument as read by the rabbit system operator and recorded by the control room reactor operator are considered to meet the monitoring requirements of UFTR SOP-A.5, "Experiments," and UFTR SOP-D.4, "Removing Irradiated Samples from UFTR Experimental Ports," for removal of experiments from the reactor.

4.6.7 In the event of <u>any</u> radiological abnormality including but not limited to breach of sample containment, suspected airborne or smearable contamination in excess of allowable limits, excessive irradiated sample activity or excessive radiation levels in the glove box, or failure of the exhaust fan, the rabbit control system operator shall:

4.6.7.1 Cease all activity;

4.6.7.2 Stand away from the glove box limiting movement of all personnel at the rabbit control station to prevent the potential spread of contamination; and

4.6.7.3 Contact the reactor operator for further instructions via intercom or telephone.

4.7 Contamination Control

4.7.1 The glove box exhaust fan shall be in operation during use of the rabbit system.

4.7.2 Gloves, waste disposal bags and labeled trash receptacles should be available near the rabbit control station glove box.

4.7.3 On a weekly basis when the rabbit system has been used, whenever the rabbit system has been used for more than 20 capsule insertions in any one day and also as appropriate for experiment conditions, a smearable contamination survey should be performed to include at least 3 swipes or area checks as follows:

4.7.3.1 Around the glove box access;

4.7.3.2 Around the contamination survey instrument;

4.7.3.3 In the path of travel between the glove box and the NAA counting laboratory room.

4.7.4 Any signs of smearable surface contamination outside the glove box in excess of UFTR SOP-D.1, "UFTR Radiation Protection and Control," limits shall be immediately reported to the reactor operator and Radiation Control; the area shall be decontaminated and proven to have contamination within acceptable limits before rabbit operations may resume.

4.8 Before securing the rabbit system from daily operations, all steps of Section 7.3 shall be sequentially performed. The pneumatic rapid sample delivery system should be partially secured on standby by performing asterisked steps in Section 7.3 for two conditions:

4.8.1 Rabbit system is to be secured with the reactor operating, or

4.8.2 Rabbit system will not be used for more than one (1) hour with or without the reactor operating.

4.9 Irradiated samples (byproduct material) returning to the glove box after removal from the reactor are considered to be removed from the UFTR R-56 License to the State 356-1 License per the applicable statement of "Policy for Transfer of Radioactive Materials Between the UFTR R-56 License and the University of Florida 356-1 State License." This transfer must be documented per the requirements of UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers." This transfer occurs upon completion of the applicable form—either UFTR Form SOP-D.6A or UFTR Form SOP-D.6E. Provided they are not removed from the UFTR site (NAA and Radiochemistry Laboratory as well as reactor cell), no further documentation of the transfer of samples is necessary.

NOTE: If the rabbit samples are not to be removed from the UFTR R-56 License, then removal from the reactor and subsequent storage are to be documented and tracked using UFTR SOP-D.4, "Removing Irradiated Samples from UFTR Experimental Ports," and its two forms.

CAUTION

If samples are to be removed from the NAA/Radiochemistry Laboratory complex, then additional documentation of the transfer while on the State 356-1 License may be required per UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers."

5.0 References

5.1 UFTR R-56 License

5.2 UFTR Safety Analysis Report

5.3 UFTR Technical Specifications

5.4 UFTR Standard Operating Procedures (A.1, A.5, D.1, D.4 and D.6)

5.4.1 UFTR SOP-A.1, "Pre-operational Checks"

5.4.2 UFTR SOP-A.5, "Experiments"

5.4.3 UFTR SOP-D.1, "UFTR Radiation Protection and Control"

5.4.4 UFTR SOP-D.4, "Removing Irradiated Samples from UFTR Experimental Ports"

5.4.5 UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers"

5.5 Statement of "Policy for Transfer of Radioactive Materials Between the UFTR R-56 License and the University of Florida 356-1 State License"

6.0 Records Required

6.1 UFTR Daily Operations Log Entries

6.2 Radiation and Swipe Survey Results

6.3 UFTR Form SOP-A.5A (Request for UFTR Operation)

6.4 UFTR Form SOP-A.8A (Rabbit System Operator Certification)

6.5 UFTR Form SOP-D.4A (Record of Sample Irradiation and Disposition)

6.6 UFTR Form SOP-D.4B (Sample Record Index)

6.7 UFTR Form SOP-D.6A (University of Florida Training Reactor/University of Florida Radioactive Material Transfer Record)

6.8 UFTR Form SOP-D.6B (University of Florida/University of Florida Training Reactor Radioactive Matèrial Transfer Record)

6.9 UFTR Form SOP-D.6E (University of Florida Training Reactor/University of Florida Rabbit System Sample Package Transfer Record)

7.0 Instructions

7.1 Rabbit System Preparation and Checkout

7.1.1 Check communications by informing reactor operator, "Setting up rabbit for operation."

7.1.2 Ensure regulator set screw tension completely released (regulator operator "backed out").

7.1.3 Inspect rabbit facilities visually to establish or verify:

7.1.3.1 Proper hose connections:

7.1.3.1.1 To in-core rabbit assembly; make up hose connection if disconnected;

7.1.3.1.2 Exhaust venting from the glove box (vent line from glove box to reactor core vent system);

7.1.3.1.3 Hose connecting regulated nitrogen supply to rabbit control station;
7.1.3.1.4 Poly hose connections from the reactor cell to rabbit control system (connecting gas and capsule transit line);

7.1.3.2 Radiation detectors available and operating:

7.1.3.2.1 Portable radiation/survey meter, and

7.1.3.2.2 Glove box area monitor;

7.1.3.3 Glove box interior properly prepared to include:

7.1.3.3.1 Glove box light,

7.1.3.3.2 Absorbent paper,

7.1.3.3.3 Clean transfer container (as required),

7.1.3.3.4 Rabbit capsule of verified integrity,

7.1.3.3.5 Operability of receiving station (leave station open).

7.1.4 Prepare receiving station:

7.1.4.1 Turn on exhaust blower (control switch on glove box);

7.1.4.2 Turn on rabbit power supply (control switch on rabbit control station);

7.1.4.3 Assure nitrogen pressure regulator set screw is backed out;

7.1.4.4 Energize supply solenoid control valve (switch on reactor control panel);

CAUTION

This step shall be performed only by a licensed reactor operator with approval of the Reactor Manager or designated alternate. This step requires a **line entry** in the UFTR Daily Operations Log.

7.1.4.5 Open or verify open rabbit receiving station (latch open).

7.1.5 Prepare rabbit system valve lineup:

7.1.5.1 Core vent sample line valve (rabbit station) shut;

7.1.5.2 Rabbit control vent line valve (rabbit station) open;

7.1.5.3 Sample transit line valve (reactor cell) open; and

7.1.5.4 Gas supply/return line valve (reactor cell) open.

7.1.6 Prepare nitrogen supply:

7.1.6.1 Crack open, then fully open nitrogen tank supply valve;

7.1.6.2 Screw in regulator set screw until downstream pressure gauge indicates in the range of 25 psi (22-25 psi preferred).

NOTE: If the pressure on the downstream side exceeds 25 psi, back out 1 full turn on the set screw until less than about 25 psi, press the "RETURN" button on the rabbit control panel and wait for the rabbit system to cycle. After the cycle is completed, attempt to set downstream pressure again. Because receiving station is open, these operations do not result in anything being inserted into the core.

CAUTION

If the rabbit system has been disconnected prior to the current experiment run, an initial test insertion with an observer stationed at the hose connections in the cell to check for gas leakage and travel of the sample container during the initial test insertion should be performed prior to reactor startup to assure proper sample delivery.

7.1.7 Verify that rabbit system operator is qualified with completed UFTR Form SOP-A.8A (Rabbit System Operator Certification) on file to certify qualification.

7.1.8 Make Daily Operations Log entry: "rabbit system energized and ready for operation; certified operator [*name*] on duty"; or similar appropriate notation.

7.2 Rabbit System Operation (Rabbit System Operator Actions Except as Noted)

7.2.1 Put gloves on;

7.2.2 Insert one empty capsule prior to initial sample insertion to test the sample insertion system for proper operation and verify proper return—manual mode should be used;

CAUTION

Assure control room reactor operator is informed of all insertion/removal steps as for all other insertions.

7.2.3 If automatic (sample insertion and removal) mode is to be used, set timer for required length of irradiation;

7.2.4 Place capsule on prepared surface inside glove box or place sample on prepared work surface on lab bench;

> **CAUTION**
>
> If the sample contains radioactive material, as from a previous irradiation, then the insertion into the UFTR experimental port requires a transfer of that material from the University of Florida 356-1 License to the UFTR R-56 License. In such cases the transfer must be controlled using SOP-D.6, "Control of UFTR Radioactive Material Transfers," and the transfer must be documented on UFTR Form SOP-D.6B (University of Florida/University of Florida Training Reactor Radioactive Material Transfer Record) or UFTR Form SOP-D.6E (University of Florida Training Reactor/University of Florida Rabbit System Sample Package Transfer Record) as desired. The transfer to the R-56 license must be documented prior to insertion into the UFTR via the rabbit system which is the point at which the transfer is considered to occur to assure proper control of licensed material.

7.2.5 Place sample into rabbit capsule;

7.2.6 Place prepared rabbit capsule near receiving station;

7.2.7 Place rabbit capsule cap side down into receiving station inside glove box;

7.2.8 Seal receiving station;

7.2.9 In conformance with the governing run request, inform reactor operator:

"Inserting sample *{number and/or type}* for {specify time} in {*specify mode*}" (or other appropriate information); "

7.2.10 Reactor operator — make entry in Daily Operations Log indicating sample identification, insertion, time of insertion, mode of insertion and expected length of insertion;

7.2.11 After verbal acknowledgment is received from the reactor operator, press "AUTOMATIC Operation" for automatic operation mode **<u>or</u>** "Manual INSERT" for manual operation mode:

7.2.11.1 Automatic sequence of operations (following depressing "Auto" push button):

1. "Auto" light on for sample insertion portion of cycle, capsule inserts into reactor;

2. "Auto" light on, "Timing" light on for sample irradiation portion of cycle, timer counting down;

3. "Auto" light on, "Timing" light off for sample removal portion of cycle, sample removed from reactor;

4. "Auto" light off, "Capsule Return" light on for completion of cycle, capsule returned to receiving station.

7.2.11.2 Manual sequence of operations (following depressing "Manual" switch):

1. "Manual" light on, capsule inserting;

2. "Timing" light on, "Manual" light on, capsule at rest in reactor;

3. *When "Capsule Return" push button is depressed:*
   "Manual" light on, "Timing" light off, capsule ejecting from reactor;

4. "Manual" light off, "Capsule Return" light on, capsule in receiving station.

7.2.12 Reactor operator — note and record (in Daily Operations Log) the change in reactivity due to sample insertion, as determined by regulating blade position changes; if little or no change is noted, make Daily Operations Log entry noting sample has "negligible reactivity effect," or other appropriate notation.

CAUTION

If the sample contains radioactive material, this entry indicates transfer of the sample from the University of Florida 356-1 Radioactive Materials License to the UFTR R-56 Reactor License.

7.2.13 In manual mode of operation, at end of desired irradiation period, inform reactor operator: "Retrieving sample" *(or other appropriate information)* and depress "Capsule Return."

NOTE: The rabbit system operator should similarly alert the reactor operator a short time before automatic return of the sample capsule.

7.2.14 When visible ("Capsule Return" lamp) and audible indication of rabbit sample return is noted:

7.2.14.1 Inform reactor operator of capsule return;

7.2.14.2 Observe glove box radiation monitor:

7.2.14.2.1 Report reading to reactor operator,

7.2.14.2.2 Reactor operator should record reported reading on glove box radiation monitor at this point;

7.2.14.3 Reactor operator — note capsule return in Daily Operations Log;

CAUTION

In the above sequence, if a sample does not return as expected, or any specific problem affecting the rabbit system or reactor is noted, the reactor operator shall be informed immediately; the rabbit system operator should remain at the rabbit control station to assist with problem diagnosis, corrective action, and contamination control as necessary unless radiation, radioactive contamination or potential airborne contamination indicate otherwise.

7.2.14.4 Transfer or store sample as follows:

NOTE: Glove box radiation monitor is calibrated to read 1000 cpm per 50 mR/hr though there is considerable variation depending on the matrix of material irradiated.

7.2.14.4.1 Observe glove box radiation monitor and report reading to reactor operator; if reading excessively high contact reactor operator for instructions, otherwise

7.2.14.4.2 Unlatch receiving station to allow rabbit capsule to fall to glove box floor;

7.2.14.4.3 Monitor glove box radiation monitor reading in cpm as well as sample radiation levels on contact and at 1 foot using portable survey meter;

7.2.14.4.4 Report all three (3) radiation level related readings to reactor operator;

7.2.14.4.5 Reactor operator — record three (3) radiation level related readings in the Daily Operations Log; this entry indicates transfer of the sample to the University of Florida 356-1 Radioactive Materials License;

NOTE: The monitoring function performed using the glove box detector and the handheld survey instrument as read by the rabbit system operator and recorded by the control room reactor operator are considered to meet the monitoring requirements of SOP-A.5, "Experiments," and SOP-D.4, "Removing Irradiated Samples from UFTR Experimental Ports," for removal of experiments from the reactor.

7.2.14.4.6 Reactor operator — log the transfer of the irradiated sample from UFTR R-56 License to University of Florida 356-1 License per UFTR SOP-D.6, "Control of UFTR Radioactive Material Transfers" using either UFTR Form SOP-D.6A (University of Florida Training Reactor/University of Florida Radioactive Material Transfer Record) or UFTR Form SOP-D.6E (University of Florida Training Reactor/University of Florida Rabbit System Sample Package Transfer Record) as desired;

7.2.14.4.7 Inspect physical integrity and guide ring wear of rabbit capsule;

7.2.14.4.8 Unscrew rabbit capsule cap;

7.2.14.4.9 Remove sample from rabbit capsule (use care to maintain sample containment integrity when handling tools such as tweezers are utilized);

7.2.14.4.10 Place sample in transfer container or shielded container as necessary.

7.2.15 In removing the sample from the glove box:

7.2.15.1 Check sample activity with portable radiation survey instrument;

7.2.15.2 Verify radiation levels are acceptable for handling;

7.2.15.3 Transfer sample to control, storage or analysis location within the NAA/Radiation Chemistry Laboratory complex.

CAUTION

Samples shall not be removed from the Reactor Building without further documentation of transfer to the University of Florida 356-1 License per UFTR SOP-D.6.

7.3 Securing the Rabbit System

NOTE: Sections 7.3.2, 7.3.5, 7.3.6 and 7.3.7 marked with asterisks (*) may be performed to shut down the rabbit system temporarily when no samples are to be inserted for a period of time; this is not considered to be securing the rabbit system but only putting it on standby to assure proper control of rabbit system operations per Section 4.8.

7.3.1 With reactor secured, rabbit capsule removed from receiving station and rabbit glove box vent fan operating, depress "Capsule Return" backlit push button (to purge activated gas from rabbit system);

7.3.2 *Close nitrogen supply tank valve to remove gas supply to the regulator;

7.3.3 Open rabbit system receiving station door to assure no insertion occurs while depressurizing rabbit system;

7.3.4 Manually cycle rabbit control station by depressing "Capsule Return" backlit push button to release upstream nitrogen pressure;

7.3.5 *Back off the regulator set screw to release tension;

7.3.6 *Turn off power to rabbit control panel via control panel backlit "Power" push button;

7.3.7 *Secure power to (deenergize) gas supply solenoid — reactor operator function (make Daily Operations Log entry indicating that rabbit system is deenergized);

7.3.8 Shut rabbit system manual valves (control station vent and reactor cell valves/sample transit line and gas supply/return line);

7.3.9 Secure lights unless needed for further operations;

CAUTION

Exhaust blower should not be secured until after the reactor is shut down and system lines have been purged.

7.3.10 Reactor operator — make Daily Operations Log entry, "rabbit system is secured" or other appropriate notation.

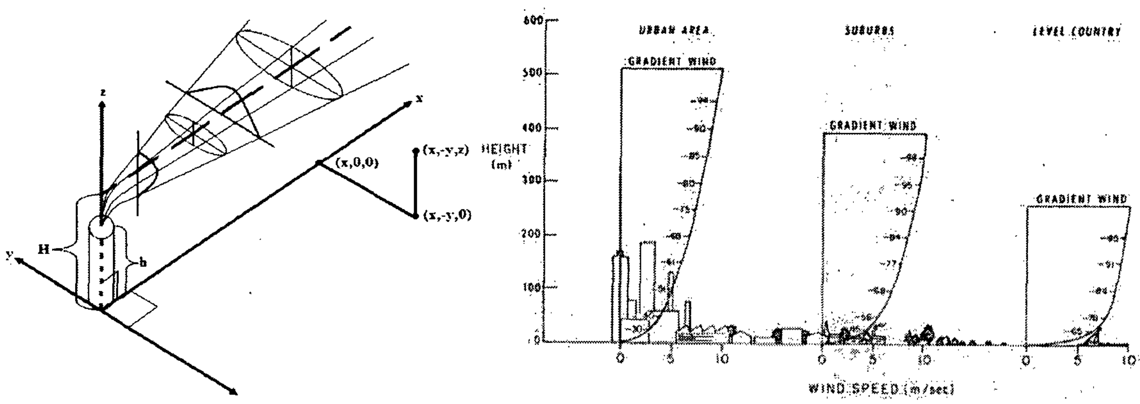
NOTE: If required or desired, the switch supplying the blower motor, illumination, and rabbit system control panel may be opened behind and to the right of the glove box.

Appendix E Ar-41 Effluent Dispersion Analysis and Dose Evaluation (in response to Q 11-1 and 11-2)

Introduction

Atmospheric plume dispersion modeling, integrating atmospheric statistical dynamics, diffusion, and meteorological data may be applied to achieve an estimate of the downwind concentration of Ar-41 effluent released during steady state operation of the University of Florida Training Reactor (UFTR). The atmospheric modeling approach utilized to determine effluent levels is based on the methods constructed by Pasquill and further expounded upon by Briggs and Turner [1 – 4], with related methodologies applied in US Atomic Energy Commission studies [5]. We note that these methods have been adopted and used as a basis for methodologies adopted by the Environmental Protection Agency, Federal Coordinator of Meteorology, and the American Society for Mechanical Engineers [1, 4, 6, 7].

Wind direction and atmospheric conditions such as temperature, solar radiation, and wind speed distinctly affect the path of effluents dispersed from an exhaust stack [1 – 4, 8]. The specific time of day versus night conditions are important, due to environmental changes in the lapse rate from the combined effects of heating and cloud cover. These varying conditions, along with the accepted mathematical models, allow the concentration of Ar-41 to be conservatively estimated with a simple one-wind, Gaussian computer code employing proper model physics: STAC2 (Version 2.1) Build 1.5b (hereafter referred to as 'STAC2.1') [7]. Note that while wind speed and temperature specifically affect effluent concentration, wind direction simply determines the vector location along which the effluent flows. The basis of STAC2.1 is a Gaussian plume model. The Gaussian model, illustrated in Fig. 1 (a), describes, in three-dimensions, the theoretical path of a plume emerging from the stack: straight downwind, horizontally, and vertically [4]. These directions correspond, respectively to a coordinate system along the x-axis (parallel to the wind vector), y-axis, and z-axis. Figure 1(a) illustrates the basic plume and plume centerline (bold, dashed line parallel to the x-axis). The "H" in the figure represents the effective stack height relative to the plume centerline, and "h" is the physical height of the stack. The profile of the plume is detailed with the elliptical and parabolic sketches to demonstrate three dimensional depths.



(a) (b)

Figure 1: (a) Coordinate System of Gaussian distributions straight downwind, horizontal, and vertical [4]; (b) Effect of Terrain Roughness on the General Wind Speed Profile [1]

In addition, frictional (drag) effects on wind speed can be approximated using a terrain category typical of the region where the atmospheric transport is occurring. For the University of Florida campus, the terrain is assumed to be urban with a flat landscape. The comparison between urban, suburban, and rural, to capture specific effects of different terrain on wind speed profiles, is shown in Fig. 1(b) [1, 4]. As surface roughness decreases, the depth of the affected atmospheric layer becomes more shallow, and the wind speed profile becomes steeper. The numbers reflected in the curves refer to normalized percentages of the wind gradient at various heights.

The UFTR, an Argonaut design, produces Ar-41 by neutron activation in the course of operations. This effluent is discharged from the air handling equipment from the exhaust stack adjacent to the reactor building. The limiting parameter for the operating duty cycle of the UFTR is the concentration of Ar-41; monthly concentration averages in uncontrolled spaces for Ar-41 must not exceed 1.00E-8 Ci/m^3 (note: 1 Ci/m^3 = 1μCi/mL), at 100% reactor power, per state and federal guidelines (10CFR20) [9, 10]. The UFTR is in close proximity to many building structures on the Florida campus, including the Ben Hill Griffin Football Stadium, other engineering departments, parking garages, and students' residence halls. The closest student residence hall, East Hall, is located approximately 190m west-southwest of the UFTR.

Calculation Theory Implemented in STAC2.1: Gauss, Pasquill, and Briggs

The Ar-41 concentrations, emitted from the UFTR stack, are calculated based on standard ASME effluent diffusion equations and Pasquill stability classes determined from atmospheric conditions, which are cast as input parameters for STAC2.1 [1, 2, 4, 7]. The principal governing equation for the determination of down-wind ground concentration is given in Eq. (1), with variables cast as: concentration of effluent (Ar-41) released (χ) in Ci/m^3, release rate (Q) in Ci/s, effective stack height (h) in m, average wind speed (u_s) in m/s, horizontal standard dispersion coefficient ($\sigma_y = \sigma_y(x)$) as a function of (x) distance from the stack in meters, vertical dispersion coefficient ($\sigma_z = \sigma_z(x)$) as a function of distance from the stack in meters, and horizontal shift from the centerline (y) in m. As can be seen by inspection of Eq. (1), the maximum predicted ground (z=0) concentrations occur immediately downwind from the stack, where there is no horizontal shift (y = 0).

$$\chi(x,y) = \frac{Q}{\pi\sigma_y(x)\sigma_z(x)u_s}\exp\left\{-\left[\frac{h^2}{2\sigma_z(x)^2}+\frac{y^2}{2\sigma_y(x)^2}\right]\right\} \qquad (1)$$

An "effective" stack height (h), in meters, is calculated, using a conservative buoyant plume estimate, and is the height of the plume centerline above the source accounting for the rise of the physical effluent discharged at the stack. The height of the plume centerline is computed by STAC2.1, while the height of the physical stack is an input parameter. The crosswind dispersion coefficients, σ_y and σ_z are determined by the atmospheric stability classes ("A" through "F") and were originally created by Pasquill, where "A" is the most *unstable* condition, and "F" is the most *stable*.

Relative "stability" is determined by the amount of solar radiation, wind speed, outside temperature, relative lapse rate (0.65 °C/100m for the case of the UFTR), and the effluent release time of day (day or night) [1, 2]. Characteristically, "unstable" is considered warm and sunny (daytime), while "stable" is cool and overcast (nighttime). Table 1 describes, in general, the characteristics attributed to each class.

Table 1: Pasquill Weather Condition Categories [2]

Category	Time of day	Typical Conditions	Weather Descriptions	Wind m/s	Wind Direction – Stand. Dev.
A	Day	Extremely Unstable	Very Sunny Summer	1	+- 25 deg
B		Moderately Unstable	Sunny and Warm	2	+- 20 deg
C		Slightly Unstable	Average Daytime	5	+- 15 deg
D	Night	Neutral Stability	Overcast Day/Night	5	+- 10 deg
E		Slightly Stable	Average Nighttime	3	+- 5 deg
F		Moderately Stable	Clear Nighttime	2	+- 3 deg

In addition, with regard to the effluent (Ar-41), STAC 2.1 takes into account the half-life, density ratio to air, specific heat of the bulk effluent, and the molecular weight (for ppt-v determinations, if required). In addition, STAC2.1 accounts for general terrain altitude as a tunable parameter for density corrections.

Validation of STAC2.1 Results both "By-Hand" and using CALPUFF

The release rate, specific to the UFTR, was calculated to be 9.228 E-5 Ci/s (The details of this release source term are depicted in Eq. (2) – (4) [1, 2, 4, 11-13]. Additional parameters in these equations, relative to the UFTR reactor, are: the undiluted release rate of Ar-41 from the reactor at 100kW (full power) (8.147 E-4 Ci/m^3), the total stack flow rate for Ar-41 from the core vent and dilution fan ($\dot{f}$) (15772 ft^3/min or 7.444 m^3/s), the dilution factor (Λ) from the dilution fan and core vent (dimensionless) (0.0152168), and the flow diluted release concentration at the top of the stack (ψ = 1.24E-5 Ci/m^3) [12, 13]. The fan flow rate value was determined as a result of the most recent service to the dilution fan. This dilution factor (Λ) takes into account that Ar-41 comes from the core (reactor) via the core vent, which is then dispersed by both the core vent and the dilution fan [12, 13].

$$\Lambda = \frac{\text{Core Vent Flow Rate } \frac{ft^3}{min}}{f \frac{ft^3}{min}} \tag{2}$$

$$\psi \frac{Ci}{m^3} = 8.147\,E-4 \frac{Ci}{m^3} * \Lambda \tag{3}$$

$$\dot{R} \frac{Ci}{s} = \psi \frac{Ci}{m^3} * f \frac{m^3}{s} \tag{4}$$

In STAC2.1, the release rate was initially modeled assuming a unit source to calculate general maximum concentrations straight downwind from the stack. Final concentrations of Ar-41, for the UFTR, were calculated by multiplying these general concentrations by the specific release rate, 9.228 x 10^{-5} Ci/s.

All calculations were verified, independently, by hand, as shown in Table 1. Tabulated values for σ_y and σ_z, atmospheric conditions for Gainesville, Florida, and the stack height and release rate for the UFTR were applied to Eq. (1) for the hand calculation. Concentrations were compared for various distances from the UFTR versus those computed using STAC2.1 for the year between July 2004 and July 2005 assuming extremely unstable conditions.

In addition, we note that the temperature of the effluent was assumed to be the same as the average ambient temperature; 23.05°C. The average *daytime* wind azimuth direction for the year was from 167.11°, and the average ground wind speed was 2.42 m/s. As shown in the last row of Table 2, the differences in the concentrations determined via tabular "by-hand" values or STAC2.1 code runs were less than 3.61% within 500m, and less than 0.77% within 100m downwind of the stack. To explain the differences, the "by-hand" computations do not account for all of the physics (buoyant plume rise with temperature, decay at time of arrival, etc), and are less robust than used in the STAC2.1 calculations [14].

Table 2: Urban Pasquill Class "A" Ground Level Concentration of Ar-41 Hand Calculation vs. STAC2.1 Results at Various Distances from the UFTR (July 2004 – July 2005)

Distance from building (m)	50	100	500	Assumed
UFTR release rate (Ci/s)	9.228E-05	9.228E-05	9.228E-05	Calculated
Effective height of effluent release (m)	12.3	12.3	12.3	[12]
Pasquill Category (Daytime)	A	A	A	Assumed
Wind speed at the stack (m/s)	3.99	3.99	3.99	[12]
Sigma y (m)	10.97	21.89	107.35	[1, 4]
Sigma z (m)	10.00	20.00	100.00	
By Hand Concentration: (Ci/m³) (Eq. 1)	3.15E-08	1.39E-08	6.81E-10	[1, 4]
STAC2.1 Multiplier: Release Rate is Unity	3.39E-04	1.50E-04	7.11E-06	Calculation
STAC2.1 Concentration: Multiplier * UFTR Release Rate (9.228E-5 Ci/m³)	3.13E-08	1.38E-08	6.56E-10	Calculation
% Difference: STAC2.1 vs. By Hand	-0.70%	-0.77%	-3.61%	Calculation

'CALPUFF' is an EPA approved California puff and slug atmospheric dispersion modeling program for accurate concentration and effluent spread prediction over complicated terrain [15]. Puffs are circular, Gaussian mappings of effluent concentrations, while slugs are elongations of these puffs using Lagrangian and Gaussian methods. Four CALPUFF models were created using summer weather conditions, details for the UFTR stack, Ar-41 characteristics, a flat, uniform terrain associated with Gainesville, FL, no "over water" effects, and using an urban wind model. The four studies included combinations of puff and slug models with two different wind extrapolation methods; power law and similarity methods. A STAC2.1model was created to match the average weather conditions, flat terrain, and urban model, as well as the UFTR and Ar-41 parameters used in CALPUFF, and then compared to each of the four cases. The results of this comparison are given in Tables 3 and 4.

Table 3: STAC 2.1 and CALPUFF/CALGROUP Comparison with a Puff Model

Models	Similarity Theory			Power Law		
	Maximum Conc. (Ci/m³)	% Diff. in Conc.	Distance from Stack (m)	Maximum Conc. (Ci/m³)	% Diff. in Conc.	Distance from Stack (m)
STAC2.1 (Maximum)	1.83E-08	30.71	103	1.83E-08	19.61	103
STAC2.1 (Same Distance as CALPUFF)	1.49E-08	6.43	79	1.49E-08	-2.61	79
CALPUFF/CALGROUP	1.40E-08	N/A	79	1.53E-08	N/A	79

Table 4: STAC 2.1 and CALPUFF/CALPGROUP Comparison with a Slug Model

Models	Similarity Theory			Power Law		
	Maximum Conc. (Ci/m^3)	% Diff. in Conc.	Distance from Stack (m)	Maximum Conc. (Ci/m^3)	% Diff. in Conc.	Distance from Stack (m)
STAC2.1 (Maximum)	1.83E-08	23.65	103	1.83E-08	18.83	103
STAC2.1 (Same Distance as CALPUFF)	1.49E-08	0.68	79	1.49E-08	-3.25	79
CALPUFF/CALGROUP	1.48E-08	N/A	79	1.54E-08	N/A	79

Maximum concentrations computed using STAC2.1 and CALPUFF software models were compared for each of the cases. It was found that the *relative distance* where the *maximum concentration occurred* was as much as 31% different between the two models. This distance of the maximum concentration was identical in all four CALPUFF models. The maximum concentration values differed from between ~19% and 31%, depending on whether a puff or slug model, or wind extrapolation power law or similarity theory was employed. STAC2.1 results most closely matched the slug, power law model. Comparisons between concentrations for the same downwind distances differed between the codes by only ~1% to 6 %. The best model relative to a comparison with STAC2.1 is the 'CALGROUP slug and wind extrapolation power law model,' which resulted in a percent difference of +/- ~19%.

Overall, the amalgam of all of these results demonstrate that STAC2.1 yields a conservative and reasonable estimate for the effluent concentration of Ar-41 downwind from the stack, and can therefore be used in establishing Ar-41 concentrations for UFTR operations.

STAC2.1 Concentration and Dose Results for the UFTR

STAC2.1 was used to calculate conservative concentrations. Remember that the highest daytime concentrations, closest to the stack, occur for Pasquill class "A," the most unstable condition. In addition, for class "C", while the concentrations are lower overall, the concentrations remain above the prescribed limit further from the stack. To ascertain the Ar-41 concentrations for the UFTR, while accounting for atmospheric influences, local weather condition measurements were acquired from the local conditions recorded daily by the Department of Physics Weather Station [2, 4]. The information located in Tables 5 and 6 are the average temperatures, wind directions, wind speeds, and Pasquill Classes attributed for the yearly period between July 2004 and July 2005 surrounding the UF campus. Table 5 contains daytime, 7am – 7pm, results, while Table 6 has the nighttime, 8pm – 6am, information. The tables also include mean values for quarterly periods and the total year. Again, we note that the monthly average computed for Ar-41 based on operation of the reactor must not exceed the maximum limit of 1.00E-8 Ci/m^3 [9].

Table 5: *Daytime* Monthly, Quarterly, & Yearly Atmospheric Averages (2004-July 2005)

Monthly Quarters, & Year	Temp		Wind Direction	Ground Wind Speed		Pasquill Classes
	F	C	Degrees	mph	m/s	
Jul '04-Sept '04	83.38	28.54	160.77	5.09	2.28	A
Oct '04-Dec '04	69.21	20.67	143.81	6.63	2.96	B
Jan '05-Mar '05	63.73	17.63	182.61	5.31	2.37	C
Apr '05-Jul '05	77.63	25.35	181.25	4.66	2.08	A
Jul '04-Jul '05	73.49	23.05	167.11	5.42	2.42	B

Table 6: *Nighttime* Monthly, Quarterly, & Yearly Atmospheric Averages (July 2004-July 2005)

Monthly Quarters, & Year	Temperature		Wind Direction	Wind Speed		Pasquill Classes
	F	C	Degrees	mph	m/s	
Jul '04-Sept '04	77.89	25.50	158.09	3.10	1.39	F
Oct '04-Dec '04	62.94	17.19	134.13	2.47	1.10	F
Jan '05-Mar '05	57.34	14.08	183.31	3.31	1.48	F
Apr '05-Jul '05	70.90	21.61	166.16	2.66	1.19	F
Jul '04-Jul '05	67.27	19.59	160.42	2.89	1.29	F

The peak Ar-41 concentrations released, for each set of individual data, using possible different Population and Pasquill Class combinations, as well as the distance from the building where these peaks occur, are illustrated in Table 7. Note that highlighted concentrations reflect the average stability classes for each time period.

Table 7: STAC2.1 Urban Ground Peak Ar-41 Concentrations (Ci/m^3) and Distance (m) from UFTR
(Highlighted concentrations reflect the average stability classes for each time period)

Time	Average Stability Classes	Jul04-Sep04		Oct04-Dec04		Jan05-Mar05		April05-Jul05		Jul04-Jul05	
		Ci/m^3	m	Ci/m^3	m	Ci/m^3	m	Ci/m^3	m	Ci/m^3	m
Day	A	2.89E-08	50	2.62E-08	44	2.86E-08	47	2.99E-08	50	2.83E-08	45
	B	2.39E-08	79	2.16E-08	75	2.36E-08	78	2.46E-08	82	2.34E-08	80
	C	2.32E-08	119	2.09E-08	111	2.28E-08	120	2.39E-08	123	2.27E-08	115
Night	F	1.09E-08	775	1.08E-08	865	1.08E-08	750	1.09E-08	835	1.09E-08	800

The total effective dose equivalent limit determined for Ar-41 is 50 mrem per year at a maximum concentration of 1.00E-8 Ci/m^3, inhaled or ingested continuously over a year [16]. Dose is linearly related to concentration as shown in Eq. (5). Results for the quarterly averages are shown in Table 8. Table 9 shows possible limiting case scenario concentrations and doses for several buildings near the UFTR based on a continuous operation concentration with dedicated winds using the April 2005 – July 2005 data. For this exercise, the wind directions were assumed to vector toward each building.

$$\text{Dose}\,\frac{\text{mrem}}{\text{yr}} = \chi\,\frac{\text{Ci}}{\text{m}^3} * \frac{50\,\text{mrem}}{1.00\,\text{E}-08\,\frac{\text{Ci}}{\text{m}^3}} \tag{5}$$

Table 8: Total Effective Dose Rate and Maximum STAC2.1 Concentration Values for the Monthly and Yearly Averages for 2004-2005, Assuming Full Power Continuous Operation

Monthly Quarters, & Year	Day Pasquill Classes	Max Day Conc. & Dist. from UFTR		Total Effective Dose Rate
		Ci/m^3	m	mrem/year
Jul '04-Sept '04	A	2.89E-08	50	145
Oct '04-Dec '04	B	2.16E-08	75	108
Jan '05-Mar '05	C	2.28E-08	120	114
Apr '05-Jul '05	A	2.99E-08	50	150
Jul '04-Jul '05	B	2.34E-08	80	117

Table 9: STAC2.1 Total Effective Dose Rate Assuming Peak Concentration Values for Buildings near the UFTR Assuming dedicated 100% Wind Vectors from the UFTR Stack to the Building

Buildings on Campus	~Distance from UFTR (m)	~Wind Direction (deg)	Max. Conc. (Ci/m^3)	Dose (mrem/yr)
Reed Lab. (RLA)	20	180	7.14E-10	4
Weimer Hall (WEIM)	40	265	2.65E-08	133
Weil Hall (WEIL) Main Eng.	63	170	2.89E-08	145
Rhines Hall (RHN) Mat. Sci.	91	80	1.96E-08	98
Reitz Student Union (REI)	133	0	1.09E-08	55
Mech.& Aerospace Eng. C (MAEC)	137	80	1.03E-08	52
Mat. Eng. (MAE)	160	40	7.87E-09	39
East Hall (EAS) (Closest Housing)	190	80	5.75E-09	29
Gator Corner Dining (FSF)	183	95	6.16E-09	31
Mech. & Aerospace Eng. B (MAEB)	200	40	5.22E-09	26
North Hall (NOR) Housing	229	93	4.04E-09	20
Ben Hill Griffin Stadium (STA) Football	250	170	3.42E-09	17
Weaver Hall (WEA) Housing	251	80	3.39E-09	17
Riker Hall (RIK) Housing	274	85	2.86E-09	14
Van Fleet Hall (VAN) ROTC	298	110	2.43E-09	12
Tolbert Hall (TOL) Housing	309	93	2.27E-09	11
Graham Hall Housing (GRA)	320	50	2.12E-09	11
O'Connell Center (SOC) Swim & Sports	331	125	1.98E-09	10
Carse Swim/ Dive (SWIM) Athletics	343	115	1.85E-09	9
Trusler Hall (TRU) Housing	411	50	1.29E-09	6
Simpson Hall (SIM) Housing	417	55	1.26E-09	6
Parking Garage VII (OCONNEL)	463	135	1.02E-09	5

Peak concentrations show that when the UFTR is assumed to operate at 100% power for 24 hours per day, then the allowable maximum concentrations and doses of Ar-41 for dedicated wind directions exceed 1.00E-8 Ci/m^3 and 50mrem/yr. This implies a "reactor duty cycle" is needed to bring the monthly average concentration of Ar-41 below the maximum allowable concentrations.

Operation Hours for the UFTR

Using the calculated peak concentrations of Ar-41, the UFTR Effective Full Power Hours (EFPH), are shown in Table 10 for daytime conditions, since daytime is when the reactor is most likely to be run. In considering the peak concentrations, this will decrease all limit exceeding concentrations to below 1.00E-8 Ci/m^3 [9, 16]. EFPH are calculated using Eq. (6) [12, 13].

$$\mathrm{EFPH}\,\frac{\mathrm{hrs}}{\mathrm{mo}} = \frac{1.00\,\mathrm{E}-08\,\frac{\mathrm{Ci}}{\mathrm{m}^3}}{\chi\,\frac{\mathrm{Ci}}{\mathrm{m}^3}} * 720\,\frac{\mathrm{hrs}}{\mathrm{mo}} \tag{6}$$

Ar-41 concentrations (χ) are in Ci/m^3. For units of kW-hours month or kW-hours/week, one can multiply by 100kW. The 720 hours/month is a standard, assuming 24 hours/day, 7 days/ week, and ~4.286 weeks/month [13]. Note that the EFPH limit based on license requirements is 235.00 hours/month or 55.56 hours/week [13].

Table 10: UFTR Hours of Operation Based on Peak Ar-41 Concentrations (Ci/m^3) for Daytime Atmospheric Conditions

Monthly Quarters, & Year	Day Pasquill Classes	Daytime Max. Conc. & Dist. from UFTR		EFPH			
		Ci/m^3	m	hrs/mo	kW-hrs/mo	hrs/wk	kW-hrs/wk
Jul '04-Sept '04	A	2.89E-08	50	249.13	24913.49	58.90	5889.72
Oct '04-Dec '04	B	2.16E-08	75	333.33	33333.33	78.80	7880.22
Jan '05-Mar '05	C	2.28E-08	120	315.79	31578.95	74.65	7465.47
Apr '05-Jul '05	A	2.99E-08	50	240.80	24080.27	56.93	5692.73
Jul '04-Jul '05	B	2.34E-08	80	307.69	30769.23	72.74	7274.05

Therefore, on average, to remain below the annual limit of 1.00E-8 Ci/m^3, the UFTR could be run up ~307 hours/month at full power for the year, with a restriction of running up to ~240 hours/month during the late spring and summer months. However, since the additional licensing restriction is 235.00 hours/month, the UFTR may be run up 235.00 hours/month (or 55.56 hours/week) all year long. Moreover, since nighttime concentrations are lower than for daytime concentrations, the UFTR can be operated at any time of day, day or night, up to a total of 55.56 hours per week. This is a significant increase from the current EFPH for the UFTR of ~116 hours/month [13].

Dilution Factor for the UFTR

The flow diluted release concentration of Ar-41 (ψ) at the top of the stack, before being affected by the environment, is approximately 1.24E-5Ci/m^3 from Eq. (5). Dilution factors are calculated by dividing concentrations in question by 1.24E-5Ci/m^3. Table 11 shows the dilution factors for the site boundary, the distance where maximum concentration occurs, and the distance where the closest residence housing is located (East Hall at a range of 190m). The concentrations were calculated using the limiting case conditions for April 2005 – July 2005, with a wind direction towards East Hall (80^o).

Table 11: Dilution Ratios based on Concentrations and Relevant Campus Locations

Campus Relevance	Distance from UFTR	Concentration	Dilution Ratio (Value:1)
	m	Ci/m^3	
UFTR Site Boundary	30	1.48E-08	838
Maximum Concentration	50	2.99E-08	415
East Hall (Closest Dorm)	190	5.75E-09	2157

Consider that the dilution ratio for the maximum concentration (415:1) is also the maximum case instantaneous release concentration from the UFTR stack. The dilution ratio, currently used by the UFTR, is 200:1 [13]. Note that 200:1 is extremely conservative compared to the computed value of 415:1 based on results from STAC2.1, which has been shown to be conservative. Table 12 illustrates the difference between the two ratios using the concentration calculated from the UFTR SOP (6.20E-8 Ci/m^3) [12, 13], and the maximum concentration as determined by STAC2.1. It is shown that the 200:1 ratio is approximately 2.07 times more conservative than the 415:1 ratio.

Table 12: Dilution Ratio Comparison

Location	Concentration (Ci/m^3)	Dilution Ratio (Top of stack: Other)	Dilution Ratio (STAC2.1:SOP)
Top of Stack	1.24E-05	N/A	N/A
UFTR SOP (Using 200:1)	6.20E-08	200	2.07
Maximum Concentration	2.99E-08	415	

Summary and Conclusions

In summary, UF researchers performed a detailed assessment of the Ar-41 dose generated by operation of the University of Florida Training Reactor (UFTR). In particular, yearly maximum predicted concentrations, dose rates, operational limits, and dilution factors were calculated for the UFTR with impact assessments assuming dedicated wind directions to nearby campus buildings at 100% full power (100kW). Note that the total effective dose equivalent limit for Ar-41 is 50 mrem per year at a maximum concentration of 1.00E-8 Ci/m^3, inhaled or ingested continuously over a year. A Gaussian plume model based code, STAC2.1, developed and benchmarked by UF researchers, was employed to calculate the maximum concentrations and the distances where they occurred. Average daytime atmospheric conditions for the University of Florida in Gainesville, FL from 2004-2005, UFTR discharge stack parameters, and Ar-41 characteristics were established as input parameters for the code. "By Hand" Pasquill plume calculations, and detailed CALPUFF (a detailed physics model) computations were used to successfully validate STAC2.1 results; the percent differences from the "By Hand" method ranged from 0.70% to 3.61% (Table 2), and the percent differences from CALPUFF models aliased using STAC2.1 were within +/- 19% (Tables 3 – 4).

Based on the available data, the *average* maximum Ar-41 concentration determined using STAC2.1 for the reactor at full power for the year was 2.34E-8 Ci/m^3 down-wind 80m from the UFTR (Table 7). The period from April 2005 – July 2005, the warmest months with the slowest wind conditions, resulted in the highest maximum concentration of 2.99E-8 Ci/m^3 at a down-wind location 50m from the UFTR. This time period and highest maximum concentration was used as the limiting value for the dilution factors, dose rates, and concentrations for the other buildings on campus, as well as the limiting value for full power hours of operation. Concerning the buildings on campus, only buildings within ~150m of the UFTR could experience concentrations and dose rates greater than the limits (Table 9) if the reactor were continuously operated at full power; this included Weimer Hall (2.65E-8 Ci/m^3), Weil Hall (2.89E-8 Ci/m^3), Rhines Hall (1.96E-8 Ci/m^3), Reitz Student Union (1.09E-8 Ci/m^3), and the Mechanical and Aerospace Engineering C building (1.03E-8 Ci/m^3). The student residence hall closest to the UFTR, East Hall, located 190m away, had both the concentration and dose rate below the annual full operation limit: 5. 75E-9 Ci/m^3. In order to reduce the maximum concentrations (and corresponding doses) to acceptable limits, the number of allowable full power hours of operation per month were calculated (Table 10). The allowable number of hours, averaged for the year, was ~307 hours/month, with a further restriction during the summer of ~240 full power hours/month. Therefore, based on the current license restriction of 235.00 hours/month, for Ar-41 emissions, the UFTR may be run up 235.00 hours/month (55.56 hours/week) all year long. This is a significant increase from the current EFPH for the UFTR of ~116 hours/month [13]. In addition, since nighttime concentrations and resultant doses are lower than for daytime, the reactor may be run 55 hours/week continuously without exceeding limit requirements.

Finally, the current dilution factor used in the UFTR SOP is 200:1 to account for atmospheric effects. Based on an analysis of the STAC2.1 results, the limiting dilution ratio is ~415:1 (Table 11). As a result, the 200:1 ratio using in the first half century of licensing was more than twice as conservative given the actual ratio of 415:1 (Table 12).

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Appendix F UFTR Organization Chart (in response to Q 12-1)

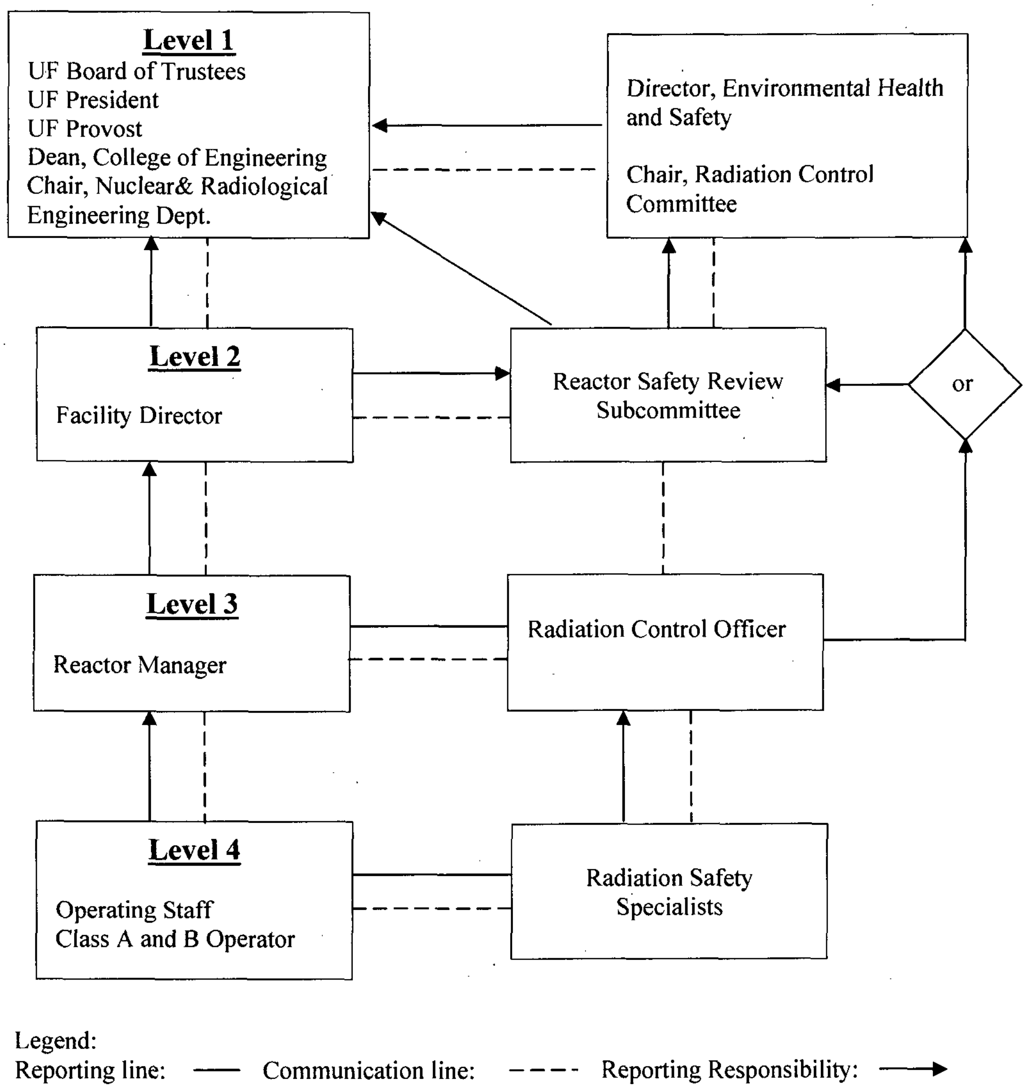

Figure 12.2 UFTR Organization Chart

Appendix G UFTR Quarterly (Q-1) Surveillance Data Sheets (in response to Q 14-6)

UFTR Quarterly #1 (Q-1 Surveillance)
CHECK OF SCRAM FUNCTIONS

Date: ____________________ Date of Last Checks: ________________________

WARNING
When any of the following checks opens the Dump Valve or results in shutting off the Primary Coolant Pump, the Dump Valve will be opened (UFTR SOP A.4, Section 7.10.2), Primary Coolant Pump will be shut off, and the system permitted to completely drain (3 minutes) before proceeding further with these checks. This delay is to preclude breakage of the rupture disc.

A. Procedures and Results:

1. **CORE VENT FAN** power loss: Raise any blade about 40 units. Shut off Vent Fan for scram. Restart core vent fan.

   (scram) ____________
   Initials

2. **DILUTION FAN** power loss: Insert scram check test adapter under Relay K-11. Use switch on adapter to bypass core vent fan scram by shunting contacts 6 and 7. Raise any blade about 40 units. Shut off core vent fan and verify no scram has occurred. Shut off Dilution Fan for scram. Restore Relay K-11 to normal. Restart Dilution Fan and core vent fan.

   (scram) ____________
   Initials

NOTICE: For the primary coolant scram checks, jumper connections are made at the small terminal box accessible at top left of the console rear center panel after rear door has been removed.

CAUTION
Make and unmake connections in the order listed to minimize probability of electrical shorts or shocks to personnel. Use the special short jumper leads. Replace terminal box cover upon completion of checks.

3. **PRIMARY COOLANT PUMP** power loss: Jumper TB 2-4 to TB 1-4 to bypass PC flow scram. Jumper TB 2-3 to TB 1-3 to bypass PC low level scram. Raise any blade about 40 units. Shut off PC Pump for scram. Cycle console power-on switch to open dump valve to permit system to drain. Remove jumper connecting TB 1-3 to TB 2-3. Leave jumper connecting TB 2-4 to TB 1-4 in place.

   (scram) ____________
   Initials

NOTE: Coolant Pump Scram indication will light during the performance of Steps 4 and 5 even though coolant scram function is bypassed.

4. **PRIMARY COOLANT LEVEL** loss: Insert test adapter under relay K-8. Shunt contacts 6 and 7 to bypass PC pump scram. Raise any blade about 40 units. Shut off PC pump to initiate scram. Cycle console power-on switch to open dump valve and permit system to drain. Remove jumper TB 1-4 to TB 2-4. Leave test adapter in place. Remove jumper from TJ1 and TJ2. Connect test lamp to TJ1 and TJ2. Shut dump valve and start PC pump. Mark PC level when test lamp indicates large change (glowing). Remove test device. Reconnect jumper to TJ1 and TJ2.

   (scram)____________ (scram level) (required ≥43.0") ____________
   Initials

5. a. **PRIMARY COOLANT FLOW** loss (inlet line sensor): Jumper TB 12-2 to TB 1-4 to bypass return line flow scram. Jumper TB 2-3 to TB 1-3 to bypass primary coolant low level scram. Raise any blade about 40 units. Raise red primary coolant flow scram setpoint on console PC Flow Meter to flow point for scram. Restore flow scram setpoint to correct setting (41 gpm). Remove jumper TB 1-4 to TB 12-2. Leave jumper TB 2-3 to TB 1-3 in place. Leave test adapter in place.

   (scram) ____________
   Initials

   b. **PRIMARY COOLANT FLOW** loss (return line sensor): Jumper TB 12-2 to TB 12-1 to bypass fill line flow scram. Raise any blade about 40 units. Shut off PC pump for scram which occurs in about 40 seconds, when return line has drained. Open the dump valve by cycling console power-on switch. Remove all jumpers and restore relay K-8 to normal.

(scram)____________ (time in seconds) ____________
Initials

Value from previous surveillance (seconds): ____________Evaluation: __________

6. **NEUTRON CHAMBER HIGH VOLTAGE REDUCTION:**

   a. **10% Drop in Neutron Chamber High Voltage (W/R Drawer):**

   Raise any 2 blades about 40 units. Pull W/R Drawer forward about 12 inches and depress W/R Drawer High Voltage Test Switch for scram and water drop. Or, alternatively, connect a voltage meter to the high voltage output on the W/R Drawer and dial the HV supply down until the trip occurs. Verify the trip occurs at less than 10% difference from the specified calibration voltage. Reset the high voltage to the original NI calibration voltage.

   NI Calibration Voltage ____________

   Trip Voltage/Percent Drop____________

   Depress PC Pump switch. Reinsert W/R Drawer.
   (water dump and scram) ____________
   Initials

   b. **10% Drop in Neutron Chamber High Voltage (Safety Channel #2):**

Open right rear console door. Raise any 2 blades about 40 units. Reach over rear swinging panel and depress Safety Channel #2 High Voltage Switch for scram and water drop. Or, alternatively, connect a voltage meter to the high voltage connection inside the swinging door and dial down the high voltage supply until the trip occurs. Verify the trip occurs at less than 10% difference from the specified calibration voltage. Reset the high voltage to the original NI calibration voltage.

NI Calibration Voltage ____________

Trip Voltage/Percent Drop ____________

Restore rear panel. Depress PC Pump switch.

(water dump and scram) ____________
Initials

7. **SHIELD TANK LOW WATER LEVEL:**

a. Remove hooks from crane sling. Attach sling to lifting lugs on shield tank shield block by using the shackles. Remove shield block and place on southeast corner of concrete reactor structure (should not rest on the steel bridge). Remove shield tank aluminum cover.

b. Raise any control blade about 40 units. Mark water level on switch body as a reference. Loosen clamp (7/16" wrench is required) and slowly raise assembly out of the water. Check that water level on switch body at scram corresponds to level on detector.

(scram) ____________
Initials

c. Restore switch to normal.

<u>**NOTE:**</u> Check water level at this time and make up demineralized water if needed. Enter start time of water makeup and total amount added into operating log and under <u>Comments</u> in Section D.

CAUTION
Do not overfill tank. One inch of water equals 14.7 gallons of water, and at 1 gpm takes 14.7 minutes. Enter stop time of water makeup into operating log when water makeup is completed.

8. **SECONDARY COOLANT PUMP** power loss:

a. Shift secondary water cooling to well water mode and verify secondary scram logic is in well water scram logic. Energize well pump to turn on secondary cooling.

b. Raise any blade to about 40 units. Simulate reactor power above 1 kW.

c. Inside the center rear console panel, switch the well water low flow bypass switch to bypass.

d. Secure the well water pump. Verify "SEC PRESS" scram illuminates and scram occurs about 10 seconds after the annunciator illuminates.

e. Return the well water low flow bypass switch to normal. Return all controls to normal.

(scram) ____________
Initials

9. **SECONDARY COOLANT FLOW** loss (city water):

a. Shift secondary water cooling to city water mode. Shift secondary coolant water scram logic to city water mode.

b. Raise any blade to about 40 units. Simulate reactor power above 1 kW.

c. Throttle city water flow and verify scram occurs at 8 gpm (higher is allowed) with <u>no</u> 10-second time delay. Return all controls to normal.

(scram) ____________
Initials

NOTE: Actual scram is set conservatively at 10 gpm.

10. **SECONDARY COOLANT FLOW** loss (well water):

a. Shift secondary water cooling to well water mode. Shift secondary coolant water scram logic to well water mode.

b. Raise any blade to about 40 units. Simulate reactor power above 1 kW.

c. Throttle well water flow to 60 gpm. Verify "SEC PRESS" scram illuminates and scram occurs about 10 seconds after the annunciator illuminates. Return all controls to normal.

(scram) ____________
Initials

REV 3, 2/03
TCN: 9/06, 11/06, 9/07

11. **CONTROL CONSOLE ELECTRICAL POWER** loss:

a. Raise all control blades about 40 units. Turn off console power by depressing the console power (power-on) green lighted switch. Restore power and verify that reactor is in scram condition (all scram lights illuminated) and that all control blades are at bottom limits.

b. Restore Power (all rods on the bottom).

(water dump and scram) ____________ Initials

B. Completion of Checkout and Restoring Reactor to Operable Condition:

1. Replace aluminum cover on small terminal box ____________ Initials

2. Replace all control console rear doors ____________ Initials

3. Replace shield tank cover and shield block ____________ Initials

4. Record quantity of water added to shield tank in Section D (Comments) ____________ Initials

5. Temperature Monitor and Recorder Checks:

a. Verify offset and gain values for each temperature channel are unchanged from the values entered on current UFTR Form SOP-E.4A (Temperature Monitor/Recorder Calibration Check) data sheet ____________ Initials

b. Ensure Temperature Recorder mechanical print head moves freely and that temperature values track with those displayed on Temperature Monitor

____________ Init ials

6. Ensure secondary coolant water is supplied from appropriate supply ____________ Init ials

7. Perform a Daily Preoperational Checkout ____________ Init ials

C. Non-Reactor Trip Checks:

1. Check to assure the source interlock initiates at ≥ 2 cps

(2 cps expected)
(cps) ____________

____
Initials

2. Check level at which extended range light goes out and assure it goes out at less than 500 cps (400 cps expected, 600 required) .... (cps) ____________

____
Initials

D. Comments (reference applicable section for all comments):

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____________________ ______________ __________________________

Performed By Date Rx Manager/Facility Director Date