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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

License Renewal
UFTR Operating License R-56, Docket 50-83

Subject: **UFTR Responses to Request for Additional Information (ML113560528)**

Please find attached our responses to request for additional information.

The UFTR licensing basis reconstitution efforts continue with further planned revisions to our safety analyses, Technical Specifications, FSAR, Emergency Plan, and ALARA program.

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

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

Executed on December 21, 2012.

Brian Shea
Reactor Manager

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

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UFTR RAI Responses

Question 1:
The regulations in Title 10 of the Code of Federal Regulations (10 CFR) Section 20.1301(a)(1) state that the total effective dose equivalent to individual members of the public likely to receive the highest dose from licensed operation may not exceed 0.1 rem (1 mSv) in a year. Section 11.1.1.1 of the 2002 UFTR Safety Analysis Report (SAR) provides estimates of the argon 41 (Ar-41) release rate based on two measurements of stack Ar-41 activity levels. In Appendix E of the UFTR responses to RAIs 11-1 and 11-2, dated November 6, 2008, another estimate is provided for the Ar-41 release that provides the basis for deriving a limit on maximum reactor operating hours of 235 hr/month. In responses to additional RAIs dated February 26, 2010, UFTR stated (see response to RAI #6) that "the information provided in the SAR should be substituted by more recent data." Please provide a reference to measurements or "more recent data," which provides the basis for the Ar-41 release rate of 9.228E-05 Ci/sec (or 1.24E-05 Ci/m3).

Response 1:
The Ar-41 values (9.228E-05 Ci/s and 1.24E-05 Ci/m^3) provided in Appendix E of the UFTR RAI response, dated April 7, 2008, originated from direct measurement of the full-power equilibrium Ar-41 release on November 1, 2007. The measurement was performed, in part, to satisfy TSSR 4.2.4(2) in accordance with UFTR SOP-E.6, Argon-41 Concentration Measurement. The measured values obtained during performance of this surveillance activity on November 1, 2007, provide the basis for the Ar-41 release rate of 9.228E-05 Ci/s (1.24E-05 Ci/m^3) referenced in Appendix E of the RAI response dated April 7, 2008.

Four Ar-41 samples were taken from the core vent system sampling port at the base of the stack on November 1, 2007. The sample activities were determined and an average was taken of the three samples with the highest Ar-41 concentration. This average resulted in a full-power undiluted Ar-41 release concentration of 8.147E-04 Ci/m^3.

Accounting for the measured core vent flow of 240 cfm, and the measured stack dilution flow of 15,532 cfm, resulted in a flow diluted release concentration at the top of the stack of 1.24E-05 Ci/m^3.

Multiplying the flow diluted release concentration at the top of the stack of 1.24E-05 Ci/m^3 by the release flow rate of 15,772 cfm (7.444 m^3/s) resulted in a stack release rate of 9.228E-05 Ci/s.

The most recent performance of UFTR SOP-E.6, Argon-41 Concentration Measurement, was completed in October 2008. This data is used to calculate UFTR Ar-41 releases using a revised methodology as described in our response to RAI Question 2.

Revisions to the UFTR Technical Specifications, FSAR, and Emergency Plan are in-progress.

Question 2:
NUREG-1537, Part 1, Section 11.1.1.1, "Airborne Radiation Sources," requests discussion and calculations that show that facility design ensures doses to the facility staff and the public will not exceed 10 CFR Part 20 limits for effluents. The UFTR SAR, provides calculated results for the most exposed individual and highest air doses external to the facility from stack releases. However, Section 11.1.1, does not appear to discuss the collective doses for facility staff in the reactor cell and immediate vicinity or for members of the public in adjacent areas and rooms. Please provide discussion with calculations, during normal operations and in the event of the ventilation damper isolating the reactor cell, demonstrating that the resultant doses for the maximum concentration in the reactor room and released from the facility (i.e., by seepage) are within the limits of 10 CFR Part 20. Additionally, please provide a recent U.S. Environmental Protection Agency (EPA) COMPLY code calculation for radiation dose to a non-occupational maximally exposed individual from airborne radioactivity releases at UFTR, as referenced in UFTR SAR, Section 11.1.7.

Response 2:
Exposure from Ar-41 During Off-Normal or Accident Conditions
As described in Chapter 9 of the UFTR FSAR, during normal reactor operations a loss of electrical power to the reactor vent damper will result in closing of the damper and an automatic reactor trip. Shortly following the reactor trip, Ar-41 production ceases. With the core vent damper closed, the potential exists for the remaining Ar-41 to diffuse into the reactor cell and then be released by cell leakage. Though credible, this event scenario is considered an off-normal or accident condition and therefore it is not analyzed in detail in Chapter 11 of the UFTR FSAR.

Consideration was given to including this event scenario in Chapter 13 of the UFTR FSAR. The unplanned loss of electrical power to the core vent damper however has no potential to cause fuel cladding failure and Ar-41 production ceases when the reactor trips in this scenario. Based on this, and the properties of the Ar-41 isotope, the potential release of fission products hypothesized in the MHA scenario remains the limiting event and therefore there is no need for a detailed quantitative analysis of the unplanned core vent damper closure scenario.

Occupational Exposure from Ar-41 During Normal Reactor Operations
As described in Chapter 9 of the UFTR FSAR, the design of the reactor cell HVAC and core vent system ensure that leakage and accumulation of Ar-41 into the reactor cell is prevented by drawing air from the cell, through the reactor and out the exhaust stack.

The only routine occupational exposure from Ar-41 occurs during performance of stack effluent surveillance measurements involving manual grab samples of stack effluent. This surveillance has a semiannual frequency and surveillance related exposures are kept ALARA.

Estimated Annual Dose in the Unrestricted Area from Ar-41 Released During Routine Reactor Operations
Regulation 10 CFR 20.1101(d) imposes an ALARA constraint on airborne emissions of radioactive material to the environment such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent (TEDE) in excess of 10 mrem per year from these emissions. This constraint ensures that dose from

airborne emissions make up no more than 10% of the 100 mrem per year limit of 10 CFR 20.1301(a)(1) and therefore this analysis will focus on ensuring compliance with the ALARA constraint.

While, in principle, the dose resulting from the release of radionuclides to the atmosphere can be determined by environmental monitoring, at the low levels consistent with the limit of the constraint, it is not reasonable to distinguish the portion attributable to UFTR Ar-41 emissions from that which is due to background radioactivity. Therefore, an expected dose must be determined analytically.

To ensure compliance with the annual TEDE constraint of 10 CFR 20.1101(d), the UFTR limits Ar-41 produced by administratively limiting effective full-power hours of operation (EFPHs). Periodic surveillance measurements of the stack effluent are performed to determine instantaneous Ar-41 concentration. Based on this instantaneous concentration and stack release point parameters, a monthly EFPH limit is calculated to ensure compliance with the annual TEDE constraint of 10 CFR 20.1101(d). Prior to reactor operation, the cumulative EFPHs for the month are compared to this monthly limit to prevent exceeding the monthly limit.

The air concentration at any point in the environment is an extremely complex function of the quantity of the radioactive material released, the configuration of the facility from which the material is released, the distance from the point of the release to the locations of interest, the meteorological conditions, and various depletion processes which remove the radioactive material from the effluent plume as it moves from the point of release to the location of the receptor. To avoid excessive conservatism which result in further constraints on UFTR energy generation, this complexity necessitates the use of a computer code. Additionally, consistent with the low level specified by the ALARA constraint, the UFTR has determined that the effort and expense of implementing a detailed site specific environmental model are not practical or reasonable.

Diffusion and atmospheric turbulence are the primary processes acting to reduce the Ar-41 concentrations in the plume. The degree of dilution resulting from atmospheric turbulence and diffusion depends upon the stability of the atmosphere, the joint frequency distribution of wind speed and direction, and the distance from the point of release to the location of the receptors. Additional factors that influence dilution include the height at which the release occurs, the rise of the effluent plume due to the momentum and/or thermal buoyancy of the gases in the effluent, and the relationship between the height of the release and the heights of the building from which the release occurs and surrounding structures.

When determining average concentrations over a long time period such as the annual average air concentrations of interest, assuming a neutral atmospheric stability is appropriate (Ref 11.4). For the case where atmospheric stability is neutral, the distance from the source to the point of maximum concentration can be calculated (Ref 11.1).

Based on the discussion above, the distance to the most exposed member of the public will be calculated and compliance with the constraint limit will be demonstrated using the NRC endorsed computer code COMPLY (Ref 11.3).

The computer code COMPLY assesses dose from airborne releases using varying amounts of site-specific information in four screening levels. In Level 1, the simplest level, only the quantity of radioactive material possessed during the monitoring period is entered. At Level 4, the COMPLY code produces a more representative dose estimate and provides for a more complete treatment of air dispersion by requiring the greatest amount of site-specific information (Ref 11.3).

The UFTR discharges Ar-41 through an exhaust stack approximately 9.1 meters above ground level. Based on the most recent surveillance measurements in October 2008, the emission rate of Ar-41 in the stack effluent is 1.351E-04 Ci/s (Ref 11.2). A summary of the October 2008 surveillance measurements is provided in Table 11-1.

Table 11-1 Summary of the UFTR Release Point Data Taken During the October 2008 Semiannual Ar-41 Surveillance Measurements

Core Vent Flow	0.10384 m^3/s
Stack Dilution Flow	6.3281 m^3/s
Ar-41 Concentration	2.100E-05 Ci/m^3
Total Stack Velocity	10.896 m/s

The maximum ground level concentration occurs on the plume center line at the downwind distance as follows (Ref 11.1):

$$\sigma_z = \frac{h_e}{\sqrt{2}}$$

where:

σ_z = vertical deviation of plume contaminant (m);

The effective stack height (h_e) can be calculated from the following equation (Ref. 11.1):

$$h_e = h + d(\frac{v_s}{\mu})^{1.4}$$

where:

h = physical stack height (9.1 m);
d = stack diameter (0.876 m);
v_s = stack effluent velocity; and
μ = mean wind speed (m/s).

The distance (x) at which the maximum concentration occurs (d_{max}) can then be determined by solving for 'x' given the vertical diffusion parameter determined previously from the effective stack height using (Ref 11.4):

$$\sigma_z = (0.06x)\ \frac{1}{\sqrt{(1 + 0.0015x}}$$

where:

x = d_{max} = distance from point of release to receptor (m);

A 30-year wind rose is used to describe the average wind speed and wind direction. This wind summary data is provided in Table 11-2.

Table 11-2 Wind Summary for January 1, 1980 to December 31, 2009 for the Gainesville Regional Airport as Reported by NOAA Online Climate Data (Ref 11.5)

Direction - From	Frequency	Speed (m/s)
N	5.90%	3.35
NNE	4.50%	3.50
NE	5.20%	3.65
ENE	5.20%	3.71
E	7.50%	3.60
ESE	4.10%	3.50
SE	3.70%	3.55
SSE	3.10%	3.50
S	4.50%	3.60
SSW	3.30%	3.76
SW	3.50%	3.96
WSW	4.60%	4.32
W	7.50%	4.07
WNW	4.90%	3.60
NW	4.60%	3.40
NNW	3.80%	3.29
Calm	22.60%	0.00
Variable	1.60%	2.11
	Mean Wind Speed =	2.81

Using the COMPLY computer code, the maximum expected TEDE, signified as $TEDE_{max}$, received by the most exposed member of the general public located at d_{max} may now be estimated. The result of calculating the annual TEDE to the general public from routine releases of Ar-41 into the unrestricted area is given in Table 11-3.

Table 11-3 Maximum Expected Annual Dose in the Unrestricted Area from Ar-41 Released During Routine Reactor Operations

μ (m/s)	h_e (m)	σ_z (m)	d_{max} (m)	$TEDE_{max}$ (mrem)
2.81	14.9	10.6	202	19.5

It should be noted that in order to receive the dose shown in Table 11-3, an individual would be required to continuously occupy the specified location (202 meters from the release point) for a full year while the reactor operated continuously for a year.

The calculated dose shows that the maximum expected Ar-41 concentration at the location of the most exposed member of the public results in greater than the ALARA constraint of 10 mrem/year but remains well within the 100 mrem/year limit of 10 CFR 20.1301(a)(1).

As discussed previously, the UFTR calculates a monthly EFPH limit based on surveillance measurements to ensure compliance with the annual TEDE constraint of 10 CFR 20.1101(d). Based on the measurements taken during the October 2008 performance of this surveillance and the associated TEDE result in Table 11-3, the UFTR is limited to 375 EFPHs per month.

This choice of ALARA constraint as the analysis limit, in combination with associated Technical Specifications, conservative occupancy assumption, and analysis above, provide reasonable assurance that dose resulting from UFTR Ar-41 emissions will meet the ALARA constraint of 10CFR20.1101(d) and be well within the limit of 10CFR20.1301(a)(1).

References

11.1 Slade, D.H. Meteorology and Atomic Energy – 1968, TID-24190

11.2 UFTR S-4 Argon Measurement Surveillance completed on October 14, 2008.

11.3 RG 4.20

11.4 EPA 520/1-89-001

11.5 NOAA Online Climate Data Center

3.7.2 Argon-41 Discharge

APPLICABILITY: Modes 1 and 2.

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

SPECIFICATIONS:

(1) The individual member of the public likely to receive the highest dose shall not be expected to receive a total effective dose equivalent in excess of 10 mrem per year from Ar-41 emissions resulting from licensed UFTR operation.

(2) Energy generation (kW- hrs) of the UFTR shall be limited to ensure the total effective dose equivalent specified in 3.7.2(1) is not exceeded.

BASIS: Reference 10 CFR 20, Regulatory Guide 4.20, and FSAR Chapter 11.

SURVEILLANCE REQUIREMENT

	SURVEILLANCE	FREQUENCY
SR 3.7.2.1	Verify UFTR energy generation is within the limit	Daily
SR 3.7.2.2	Verify the expected total effective dose equivalent to the individual member of the public likely to receive the highest dose from UFTR Ar-41 emission is within the limit.	Semiannual
SR 3.7.2.3	Determine the UFTR energy generation limit based on measurement of the stack effluent discharge.	Semiannual

Question 3:
NUREG-1537, Part 1, Section 11.1.1.1, "Airborne Radiation Sources," requests discussion and calculations that show that facility design ensures doses to the facility staff and the public will not exceed 10 CFR Part 20 limits for effluents. The UFTR SAR (Section 11.2.2.1 and Table 11-4) utilizes a dilution factor of 200:1 in calculations for stack diluted emissions and maximum release concentrations for Argon-41. Please provide justification, including the source and derivation, for this dilution factor.

Response 3:
A 200:1 dilution factor has been applied to the plume rising vertically from the stack since approved as part of Technical Specification Amendment 8 in 1964. After an extensive search, which included interviews with current and prior staff, we are unable to locate any original source documents that detail the derivation of the 200:1 atmospheric dilution factor. Though the original derivation of this value cannot be located, the 200:1 atmospheric dilution factor was shown to be very conservative in the UFTRs RAI response dated April 7, 2008.

Having an approved atmospheric dilution factor in the Technical Specifications is outside the scope of Technical Specification content standards. The revised methodology used to calculate UFTR Ar-41 releases is described in our response to RAI Question 2.

Revisions to the UFTR Technical Specifications, FSAR, Emergency Plan, and ALARA program are in-progress.

Question 4:
NUREG-1537, Chapter 13, "Accident Analyses" recommends maximum hypothetical accident (MHA) dose analysis to the public. The MHA presented in the 2002 UFTR SAR, as supplemented, does not appear to discuss the non-occupational dose to on-site occupants of the building such as students, faculty, visitors, etc. Please provide a dose assessment for the maximum exposed individual member of the public that shows compliance with 10 CFR 20.1301 for the unrestricted areas adjacent to the UFTR, such as the Nuclear Science Center, Reed lab, and other adjoining buildings for the MHA analyses. Please describe the assumptions used and any systems, plans, procedures or stay times for which credit is taken in the analysis such as: ventilation system status, leakage rate into the building, radiation exposure from the radioactive cloud shine, evacuation procedure and timing, etc.

Response 4:
Maximum Hypothetical Accident and Fuel Handling Accident
For the UFTR, the limiting accident scenarios involving release of fission products into the reactor cell both involve severe mechanical damage to a plate type fuel element. Since the analysis methodology for both scenarios is essentially the same, they are both discussed here in the same section.

The Maximum Hypothetical Accident (MHA) for the UFTR is a core-crushing accident in which the core is assumed to be severely crushed in either the horizontal or vertical direction by postulating that a 4500 lb concrete shield block is inadvertently dropped directly onto the core. Because the core is still structurally well shielded even when the last shield block is removed, it is difficult to conceive of how the core could actually be crushed. Nevertheless, even though the possibility of this hypothetical accident is extremely remote, the assumption is made that dropping the concrete shield block could result in severe mechanical damage to the fuel and significant release of fission products.

For the Fuel Handling Accident (FHA), the scenario assumes that one irradiated fuel element is damaged during a core offload or reload, fuel inspection, or other irradiated fuel handling operation. Fuel handling operations allow moving only one bundle at a time and are designed to ensure the fuel handlers are constantly shielded from the irradiated fuel assembly. The FHA is considered the most limiting credible accident for the UFTR and therefore it is used as the accident basis for Emergency Planning purposes.

Typically, the UFTR is shutdown from power operation for more than seven days prior to commencing irradiated fuel handling operations. In all cases, the reactor would be shutdown from power operation for at least three days to allow substantial decay of fission product inventory. Since the primary water may be drained from the core immediately after shutdown, any fission product release is assumed to be directly to the air of the reactor cell as a conservative measure. Technical Specifications require at least three days to pass after UFTR power operations before not only fuel handling but also before moving the last two layers of concrete blocks to access the fuel. This limits the potential consequences of a fuel handling accident and precludes damaging fuel with a dropped shield block before three days have elapsed.

The following data and assumptions are used to evaluate the source terms associated with these accidents:

1. The reactor is operated continuously at 100 kW steady-state power for 30 days with an equilibrium concentration of fission products.
2. The fuel element with highest power is selected for evaluation. These calculations are described in FSAR Chapter 4.
3. Radioisotope inventories are calculated three days after shutdown from power operation.
4. The radioisotopes of greatest significance for release are radioiodine and the noble gases, krypton and xenon.
5. For the MHA, it is postulated that the core would undergo severe mechanical damage due to the core crushing accident and that this damage would be sufficient to expose fuel surface areas equivalent to stripping all the cladding from one face of one fuel plate (Ref. 11). It is further assumed that 100% of the gaseous activity produced within the recoil range of the fission fragments (1.37E-03 cm), or 1.919E-01% of the total gaseous activity instantaneously escapes from the exposed fuel surfaces into the reactor cell (Ref. 1). This is a very conservative estimate given the low fuel temperature and the fact that not all fission gases would move out of the fuel and occupy the full volume of the recoil range within the aluminum clad.
6. For the FHA, it is postulated that the irradiated fuel element would undergo severe mechanical damage due to damage during fuel handling operations and that this damage would be sufficient to split the fuel bundle into two pieces exposing a fuel surface area equivalent to a guillotine type break of all 14 fuel plates. It is further assumed that 100% of the gaseous activity produced within the recoil range of the fission fragments (1.37E-03 cm) or 4.567E-03% of the total gaseous activity instantaneously escapes from the exposed fuel surfaces into the reactor cell (Ref. 1). This is a very conservative estimate given the low fuel temperature and the fact that not all fission gases would move out of the fuel and occupy the full volume of the recoil range within the aluminum clad.

Radionuclide Inventories

Radionuclide inventories for the highest power fuel element were calculated using the ORIGEN-S code (Ref. 2) under the assumptions in the previous paragraph.

Activities of the krypton, iodine, and xenon isotopes for the highest power element are given in Table 13-6 along with the inventory that is assumed to escape from the damaged fuel into the air of the reactor cell.

Table 13-6 Calculated Radionuclide Inventories (Ci) Three Days After Shutdown

Isotope	Highest Power Fuel Element	MHA 1.919E-01 % of Highest Power Fuel Element	FHA 4.567E-03 % of Highest Power Fuel Element
Kr-85	8.77E-02	1.68E-04	4.00E-06
Kr-85m	8.95E-04	1.72E-06	4.09E-08
Kr-88	3.73E-06	7.16E-09	1.70E-10
I-129	1.09E-07	2.10E-10	5.00E-12
I-130	1.56E-04	3.00E-07	7.14E-09
I-131	1.02E+02	1.95E-01	4.64E-03
I-132	1.11E+02	2.12E-01	5.05E-03
I-133	2.96E+01	5.67E-02	1.35E-03
I-135	1.49E-01	2.85E-04	6.79E-06
Xe-133	2.49E+02	4.77E-01	1.14E-02
Xe-133m	1.98E+00	3.80E-03	9.06E-05
Xe-135	4.15E+00	7.97E-03	1.90E-04
Xe-135m	1.53E-02	2.93E-05	6.96E-07

Dose Calculations

Doses were calculated for the most exposed member of the public and for facility staff. Occupational exposure limits are discussed in 10 CFR 20.1201 and public exposure limits are discussed in 10 CFR 20.1301.

For occupational dose limits, the Section 20.1201 limits are as follows:
An annual limit, which is more limiting of the total effective dose equivalent (TEDE) being equal to 5 rem, or the sum of the deep-dose equivalent (DDE) and the committed dose equivalent (CDE) to any individual organ or tissue other than the lens of the eye being equal to 50 rem.
In addition, Section 20.1201 places limits on the exposure to the lens of the eye and the skin of the whole body and the extremities. However, of the isotopes present in the inventory at the initiation of the accident, the only contribution to the skin dose is from Kr-85 according to Federal Guidance Report No. 11. Thus, it is reasonable to assume that the most limiting case for the occupational dose assessment is either the 5 rem limit for TEDE or the 50 rem limit for the sum of the DDE and CDE for an individual organ.

For the public dose limits, the Section 20.1301 limit of concern is as follows:
The total effective dose equivalent to individual members of the public from licensed operation is limited to 0.1 rem in a year.

Occupational Exposure

The location of the accident is inside the reactor cell which represents the immediate surroundings of the reactor.

The following assumptions were used in this analysis:

1. The fission product release is uniformly dispersed within the volume of the reactor cell.
2. The free air volume of the reactor cell is conservatively calculated to be 4.99E+04 ft^3.

3. Breathing rate: 3.33E-04 m^3/s (Ref. 4)
4. Dose coefficients are taken from Federal Guidance Reports No. 11 and 12. (Refs. 4, 5)

Dose results are given as a dose rate, which can be used to assess the evacuation and reentry of facility staff in the event of an accident.

Dose conversion factors used to calculate thyroid and TEDE doses for the occupational exposures are shown in Table 13-7. The values are obtained from Ref. 4 and Ref. 5.

Table 13-7 Dose Conversion Factors

Isotope	DDE_{eff} Dose Coefficient ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Coefficient (Sv/Bq)	DDE_{thy} Dose Coefficient ($Sv\text{-}m^3/Bq\text{-}s$)	CDE_{thy} Dose Coefficient (Sv/Bq)
Kr-85	1.19E-16		1.18E-16	
Kr-85m	7.48E-15		7.33E-15	
Kr-88	1.02E-13		1.03E-13	
I-129	3.80E-16	4.69E-08	3.86E-16	1.56E-06
I-130	1.04E-13	7.14E-10	1.04E-13	1.99E-08
I-131	1.82E-14	8.89E-09	1.81E-14	2.92E-07
I-132	1.12E-13	1.03E-10	1.12E-13	1.74E-09
I-133	2.94E-14	1.58E-09	2.93E-14	4.86E-08
I-135	7.98E-14	3.32E-10	8.01E-14	8.46E-09
Xe-133	1.56E-15		1.51E-15	
Xe-133m	1.37E-15		1.36E-15	
Xe-135	1.19E-14		1.18E-14	
Xe-135m	2.04E-14		2.04E-14	

Calculated TEDE and thyroid doses for the occupational exposures for the MHA and FHA are shown in Tables 13-8 and 13-9. Exposure is given in terms of dose rate in rem per hour and the exposure received over a 5-minute period. A period of 5 minutes is considered a reasonable time for a worker in the reactor cell to evacuate the cell in event of an accident.

Table 13-8 Summary of Occupational Radiological Exposure for the MHA

Location	Thyroid Dose		TEDE Dose	
	Rate (rem/hr)	5 Minute Exposure (rem)	Rate (rem/hr)	5 Minute Exposure (rem)
Inside Reactor Cell	47.25	3.938	1.525	0.127

Table 13-9 Summary of Occupational Radiological Exposure for the FHA

Location	Thyroid Dose		TEDE Dose	
	Rate (rem/hr)	5 Minute Exposure (rem)	Rate (rem/hr)	5 Minute Exposure (rem)
Inside Reactor Cell	1.125	9.37E-02	0.036	3.03E-03

Exposures in both accident scenarios are less than the annual occupational dose limits. In the case of the MHA scenario, a worker could remain in the reactor cell for approximately one hour immediately following the accident before coming near an occupational dose limit. For the case of the FHA, many hours of exposure may occur before the doses would reach the occupational dose limits.

Public Exposure

As described in Chapter 9 of the UFTR FSAR, under normal operation the design of the reactor cell HVAC and core vent system ensures that leakage from the reactor cell and accumulation of radionuclides in the reactor cell is prevented by drawing air from the cell, through the core vent system and out the exhaust stack where it is monitored and diluted.

In previous UFTR MHA and FHA analysis, the UFTR assumed that core vent and HVAC systems are shutdown following a FHA or MHA and that gaseous radionuclides released during the postulated accident escape the reactor cell by leakage.

In this revised FHA and MHA analysis, the UFTR is proposing a change to the UFTR Technical Specifications to eliminate the Area Radiation Monitor / Evacuation alarm interlock that shuts down the core vent, stack dilute, and HVAC systems. This facility modification will ensure that any potential accumulation of radionuclides in the reactor cell resulting from a postulated accident are directed out the exhaust stack where they will be diluted and better dispersed.

Following completion of this facility modification, postulated doses to the most exposed member of the public can be estimated, assuming the reactor stack as release point, for the FHA and MHA scenarios using the NRC endorsed computer code COMPLY at Level 3 (Ref. 7). The source terms for these postulated accidents are discussed earlier and tabulated in Table 13-6.

The following assumptions were used in this analysis:

1. Locations outside the reactor building are exposed to a fractional release of the total iodines due to plateout of iodines inside the reactor cell and reactor ventilation components. The portion available for release outside of the reactor cell is conservatively assumed to be 25% (Ref. 6).
2. The most exposed member of the public is conservatively assumed to continuously occupy a location 10 meters away from the UFTR stack outside of a building or shelter of any type. This is the approximate ground level distance from the base of the stack to the closest walking path which is located in the unrestricted area just east of the reactor building.

3. The most exposed member of the public is conservatively assumed to get all their meat, milk, and vegetables from on-campus gardens and farms.
4. Following the accident, no credit is taken for radiological decay during holdup in the reactor cell prior to release through the stack.

Using the COMPLY computer code, the maximum postulated TEDE received by the most exposed member of the general public may now be estimated. The result of calculating this annual TEDE for the MHA is given in Table 13-10. The result of calculating this annual TEDE for the FHA is given general in Table 13-11.

Table 13-10 Summary of Maximum Postulated Public Radiological Exposure for the MHA

Most Exposed Location	TEDE (mrem/year)
10 meters from reactor stack	6.1

Table 13-11 Summary of Maximum Postulated Public Radiological Exposure for the FHA

Most Exposed Location	TEDE (mrem/year)
10 meters from reactor stack	0.1

Exposures in both accident scenarios are significantly less than the annual public dose limit of 0.1 rem in one year.

References:

1. NUREG/CR-2079, "Analysis of Credible Accidents for Argonaut Reactors", prepared by Battelle Pacific Northwest Labs, Richland, WA for the Nuclear Regulatory Commission, Washington, DC, April 1981.
2. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," ORNL/TM-2005/39, Version 6.11.
3. USNRC, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria", NUREG 1537, Part 2, February 1996.
4. Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL, 1988.
5. Federal; Guidance Report 12, "External Exposure to Radio nuclides in Air, Water, and Soil," ORNL, 1993
6. Regulatory Guide 1.4.
7. COMPLY Computer Code

Question 5:
The MHA presented in the 2002 UFTR SAR, as supplemented, states "the current UFTR Environmental Impact Appraisal limits the UFTR to 235 full power hours per month." Additionally, the UFTR Emergency plan (EP), Revision 15, dated February 2007 contains assumptions and initial condition for reactor operation of 100 kW steady-state power for 4 hours per day for 30 days. The EP assumption is stated to be conservative based on the premise that the UFTR has a license limit of 23.5 MW-hours per month. Please provide justification for the limit on hours of operation including the source and derivation of this assumption for the stated limit and reference to the license condition or technical specification that provides this license limit.

Response 5:
After an extensive search, which included interviews with current and prior staff, the source and derivation of the 23.5 MW-hour monthly limit cannot be located. We believe the limit was intended to restrict Ar-41 releases to the environment. This limit is no longer needed as the Ar-41 effluent is monitored and operating hours are limited by UFTR Technical Specifications. The revised methodology used to calculate UFTR Ar-41 releases and monthly run-time limit is described in our response to RAI Question 2.

The UFTR Emergency Plan was designed to handle accident consequences determined to be credible. The operating hour assumptions of the Fuel Handling Accident (FHA) analysis performed as part of the HEU-LEU conversion analysis provide a reasonable power history for a credible FHA scenario. This scenario was not intended to be the limiting accident analysis case.

To eliminate confusion, however, the UFTR has revised the operating hour assumption for the FHA to be the same as the MHA. This revised analysis is detailed in our response to RAI question 4.

Revisions to the UFTR Technical Specifications, FSAR, and Emergency Plan are in-progress.

Question 6:
The dose calculation results for the MHA in Section 13.4.3, Table 13-15 and fuel handling accident (FHA) in Section 13.3.3, Table 13-10 presented in the University of Florida fuel conversion SAR and the summary of occupational and public dose results for a FHA for the low-enriched uranium (LEU) fueled core in Table 1.1 of the UFTR Emergency plan, Revision 15, dated February 2007 are expressed as whole body and thyroid doses. Please provide updated occupational and public exposure results as a Total Effective Dose Equivalent (TEDE) in accordance with 10 CFR Part 20.

Response 6:
Updated occupational and public exposure results for the revised MHA and FHA are provided in response to RAI Question 4.

Revisions to the UFTR Technical Specifications, FSAR, and Emergency Plan are in-progress.

Question 7:
Table 11-3 of the UFTR SAR summarizes liquid waste of high and low energy emitting mixed nuclides released from the UFTR as a list of maximum activity in any release (µci/ml). Additionally, the UFTR "As Low As Reasonably Achievable" (ALARA) program included as Appendix 11-B of the UFTR SAR establishes Investigational levels for UFTR Facility liquid effluents. These investigational levels are stated as a percentage of the values in 10 CFR 20, Appendix B. Please explain the relationship between these release concentrations and investigational levels to the regulatory limits in 10 CFR Part 20, Appendix B for each radionuclide in the mixture applicable to the assessment and control of doses to the public and environment.

Response 7:
The UFTR has reviewed these sections of the FSAR submitted in 2002, our current Technical Specifications, and our current ALARA program. There are administrative inconsistencies between these documents regarding liquid releases that require correction. As discussed earlier in this transmittal, revisions to the UFTR Technical Specifications, FSAR, and ALARA program are in-progress.

The relationship between the 10 CFR 20 Appendix B limits, UFTR Technical Specifications, FSAR, and ALARA program will be clarified for liquid releases as part of these revisions.

40 CFR Part 61
National Emission Standards
for Hazardous Air Pollutants

REPORT ON COMPLIANCE WITH THE CLEAN AIR ACT LIMITS FOR RADIONUCLIDE EMISSIONS FROM THE COMPLY CODE - V1.6.

Prepared by:

UFTR

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Dan Cronin

352-294-2103

Prepared for:

U.S. Environmental Protection Agency

Office of Radiation and Indoor Air

Washington, DC 20460

Maximum Estimated Argon-41 Dose to Public

SCREENING LEVEL 4

DATA ENTERED:

Nuclide	Release Rate (curies/SECOND)
AR-41	1.351E-04

Release height 14 meters.

Building height 8 meters.

The source and receptor are not on the same building.

Building width 19 meters.

Building length 39 meters.

STACK DISTANCES, FILE: recept1.dat

DIR	Distance (meters)
N	202.0
NNE	202.0
NE	202.0
ENE	202.0
E	202.0
ESE	202.0
SE	202.0
SSE	202.0
S	202.0
SSW	202.0

SW	202.0
WSW	202.0
W	202.0
WNW	202.0
NW	202.0
NNW	202.0

COMPLY: V1.6. 12/21/2012 9:33

WINDROSE DATA, FILE: gnv30.fl

Source of wind rose data: NCDC GNV
Dates of coverage: 1/1/1980 through 12/31/2009
Wind rose location: GNV
Distance to facility: 6 miles

Percent calm: 0.24

Wind FROM	Frequency	Speed (knots)
N	0.059	6.50
NNE	0.045	6.80
NE	0.052	7.10
ENE	0.052	7.20
E	0.075	7.00
ESE	0.041	6.80
SE	0.037	6.90
SSE	0.031	6.80
S	0.045	7.00
SSW	0.033	7.30
SW	0.035	7.70
WSW	0.046	8.40
W	0.075	7.90
WNW	0.049	7.00

NW	0.046	6.60
NNW	0.038	6.40

He produces his own VEGETABLES at home.

He produces his own MILK at home.

He produces his own MEAT at home.

NOTES:

The receptor exposed to the highest concentration is located
202. meters from the source in the W sector.

He produces his own VEGETABLES at his home.

He produces his own MEAT at his home.

He produces his own MILK at his home.

Input parameters outside the "normal" range:

None.

♀

COMPLY: V1.6. 12/21/2012 9:33

RESULTS:

Effective dose equivalent: 19.5 mrem/yr.

*** Failed at level 4.

This facility is NOT in COMPLIANCE.
Please send this report to your regional EPA office.
You may contact your regional EPA office to determine further action.

********** END OF COMPLIANCE REPORT **********

♀