EMERGENCY PLAN

for the

UNIVERSITY OF FLORIDA TRAINING REACTOR

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UFTR EMERGENCY PLAN

LIST OF FIGURES

Page

LIST OF TABLES

Page

1.0 INTRODUCTION

1.1 Scope of the UFTR Emergency Plan

The University of Florida Training Reactor (UFTR) Emergency Plan is designed to cope with emergencies which arise as a result of, or in connection with, reactor operations. Where possible, it adopts the standard campus procedures that are widely practiced and understood by campus emergency response teams. However, the plan deals primarily with emergency responses that are required by the unique nature of the research reactor facility and the credible accidents that might arise within the facility.

1.2 Basis for the UFTR Emergency Plan

The emergency planning requirements for research reactors are specified in 10 CFR, Part 50, Appendix E.[1] Applicable guidance in emergency planning is set forth in Revision 1 to Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors" (March 1983)[2] and in ANSI/ANS-15.16-1982, "Emergency Planning for Research Reactors."[3] These documents were used as the basis for development of the UFTR Emergency Plan.

The UFTR Emergency Plan and associated Standard Operating Procedures meets or exceeds the requirement and guidelines delineated in these documents.

1.3 Characteristics of the UFTR Facility

1.3.1 Reactor Characteristics

The UFTR is of an Argonaut-UTR type, with some modifications to adapt it to the university training program. The reactor core is heterogeneous in design currently using about 20 kg (uranium) of 19.75% enriched uranium silicide-aluminum (U_3Si_2-Al) fuel contained in aluminum cladding. Water is used as the coolant and also as the moderator. The remainder of the moderator consists of graphite blocks which surround the boxes containing the fuel plates and the water moderator. The graphite also serves as a reflector. The fuel is contained in MTR-type plates assembled in bundles and contained in 6 water-filled boxes surrounded by reactor grade graphite. Each bundle is composed of 14 fuel plates, each of which is a sandwich of aluminum cladding around a uranium silicide-aluminum "meat."

There are four swinging-arm-type control blades (three safety and one regulating) consisting of four cadmium vanes protected by magnesium shrouds that operate by moving in a vertical arc within the spaces between the fuel boxes. These blades are moved in or out by mechanical drives or they may be disconnected by means of electromagnetic clutches and allowed to fall into the reactor. The drives, located outside the reactor shield for accessibility, are connected to the blades by means of long shafts. An isometric sketch of the UFTR facility with the shielding removed is presented in Figure 1.1.

The biological shield is made of cast-in-place concrete 3 to 6 ft. thick. Access to the ends and top of the reactor is provided by removal of concrete blocks cast to fit openings and to prevent radiation streaming.

1.4 UFTR Facility Location

The UFTR building is located on the campus of the University of Florida at Gainesville, in Alachua County. Figure 1.4 shows the geographic location of Alachua County with Gainesville at its center in the north central portion of the Florida peninsula. Figure 1.5 shows the location of the University of Florida campus within the City of Gainesville. As shown in Figure 1.5, the University of Florida campus is in the southwestern quadrant of the greater Gainesville area approximately one mile from the center of the city (University Avenue and Main Street).

The Nuclear Sciences Center (Building 634) is annexed to the reactor building which is labeled "UFTR (Building 557)" in Figure 1.6. Distances to key campus structures are shown via concentric circles drawn with the UFTR as the center, the first circle having a 250 ft. radius and the rest being at 500 ft. increments from the reactor building located at the center point. A detailed UF campus map showing all major arteries along with building locations, landmarks and boundaries is shown in Figure 1.7. Emergency vehicular approach to the reactor building is via one of three service drives delineated in Figure 1.8: the reactor service drive leading from Gale Lemerand Drive to an area west of the reactor building, the Journalism lot service drive leading from Stadium Road to an area east of the reactor building (limited during construction to enlarge the Journalism building), and the Nuclear Sciences Center service drive also leading from Gale Lemerand Drive to an area south of the reactor building.

1.5 Credible Accidents and Consequences

Credible accidents for the UFTR were discussed in the University of Florida Safety Evaluation Report (SER).[5] That discussion was based largely upon a generic study of various postulated types of accidents leading to cladding failure. Generic accidents analyzed by Battelle Pacific Northwest Laboratory include:

(1) Insertion of Excess Reactivity
(2) Explosive Chemical Reaction
(3) Graphite Fire
(4) Fuel-Handling Accident
(5) Core-Crushing Accident

For case 1, the conversion SAR concludes that no credible nuclear excursion could lead to fission product release since there would be no fuel or clad melting. The second scenario was considered impossible because rapid metal-water reactions will not occur in the UFTR. Similarly, a serious graphite fire resulting in damaged fuel is dismissed because the set of required conditions is essentially not possible. In addition, the core-crushing accident was analyzed as the UFTR Maximum Hypothetical Accident (MHA) for the UFTR. For the so-called MHA, the UFTR was run at full power for 30 days to build up equilibrium fission products followed by assumed instantaneous release of 100% of the noble gases and volatiles produced within recoil range of the clad surface (or 2.7% of total volatile activity) in one fuel bundle. Such a scenario is not credible, however, and was intended only to demonstrate how dose rates at various distances are affected by various building leakage rates up to and including total failure. Although no fuel was melted, mechanical damage was assumed to cause effective cladding removal and resultant gaseous activity release as discussed above. Since the event is the unlikely dropping of a shield block,

this event was considered extremely unlikely; again, it was used only as the maximum hypothetical accident, not a credible accident.

Therefore, in agreement with the Battelle study, it was concluded that the most credible accident was the loss of cladding on one fuel plate due to a fuel handling accident. The cladding loss accident lacks a detailed causal explanation, but intuition suggests that the outer plates of a fuel element are the most likely to suffer mechanical damage. The Battelle postulated cladding loss is equivalent to two sides of a single fuel plate. In the LEU core, for the Fuel Handling Accident (FHA), the reactor is assumed to operate at 100 kW steady state power for 4 hours per day for 30 days. Then the fuel element with highest power was selected for evaluation with the accident applied to the highest power fuel bundle with a 3 day delay since at least 3 days are required to pass after the last reactor operation at power before not only fuel handling but also before moving the last two layers of protective concrete blocks to access the fuel to limit possible potential consequences of fuel handling accidents and to preclude damaging a fuel bundle with a dropped shield block before 3 days have elapsed. For the FHA, the assumption continues that the cladding would be stripped from the selected LEU fuel bundle for the fuel handling accident.

As indicated in Table 1.1, the radiological exposure from the FHA calculated for a member of the public at closest approach would be much less than 1.0 mRem whole body dose from the noble gases and less than 6 mRem to the thyroid from the iodine gases. Correspondingly, occupational radiological exposure would be much less than 1.0 mRem whole body dose and less than 3 mRem to the thyroid. For these accidents, radiation doses to the public in unrestricted areas as well as workers would be far below the limits stipulated in 10 CFR 20.

Even so, the assumptions used in these calculations are believed to be very conservative for three reasons:

(1) First, it is highly unlikely that dropping a fuel element would be severe enough to cause fuel damage equivalent to stripping the cladding from an entire fuel plate.

(2) Second, fuel transfer operations cannot begin immediately after shutdown. The shielding blocks first must be removed from the structure to reveal the fuel elements in the core. In addition, the UFTR does not shut down and immediately begin to manipulate fuel. Typically, the UFTR will shut down from power operations for more than 7 days prior to commencing fuel-handling operations. In all cases, the reactor would be shutdown from power operations at least 3 days to allow substantial decay of fission product inventory. In addition, the last two layers of shield blocks over the core area will not be removed for at least 3 days after the last operation at power.

(3) The UFTR would not usually operate 4 hours/day for a 30-day period. The reactor has a license limit of 23.5 MW-hours per month but the UFTR averaged less than 25.0 MW-hours per year for a typical ten-year period (9/81–8/91).

Table 1.1

Summary of Occupational and Public Dose Results for the Fuel Handling Accident (FHA) for the LEU Fueled Core

Occupational Radiological Exposure Rate from LEU Core				
	Thyroid Dose Rate		Whole Body Dose	
Distance	Rate (rem/hr)	5-Minute Exposure (rem)	Rate (rem/hr)	5-Minute Exposure (rem)
Inside Reactor Building	0.0285	0.0024	5.63×10^{-5}	4.69×10^{-6}

Limit: Thyroid = 30 rem, Whole Body = 5 rem

Radiological Exposure for the Public from LEU Core							
Distance (m)	Time of Exposure (hr)	Thyroid Dose (rem)			Whole Body Dose (rem)		
		Leak Rate (% Vol/hr)			Leak Rate (% Vol/hr)		
		10%	20%	100%	10%	20%	100%
16.5	2	0.00134	0.002	0.006	1.0×10^{-6}	1.8×10^{-6}	4.9×10^{-6}
190	24	0	0	0	1.6×10^{-7}	1.8×10^{-7}	1.9×10^{-7}

Limit: Thyroid = 3 rem, Whole Body = 0.5 rem

Because of the conservative basis of the inventory and release calculations, the UFTR staff feels that it is extremely unlikely that members of the general public will receive radiation exposures greater than those permitted by 10 CFR 20 [6] when the reactor building is secured following such an accident. This position is in agreement with the UCLA staff position as described in their proposed Emergency Plan. Nevertheless, in keeping with UFTR Tech Specs and the ALARA criterion, the appropriate accident control strategy is to evacuate and secure the entire reactor building, including the reactor cell. There will be no pressure increases from a dropped element accident so maintaining the integrity of the reactor cell can greatly mitigate the radiation doses to the public. Of more direct concern is protecting personnel within the UFTR facility. Securing the facility limits releases and allows time to analyze a situation and to take advantage of decay of activity released to the cell atmosphere.

The UFTR Technical Specifications require an interlock to shut down the reactor cell air conditioning and the ventilation system when the evacuation siren is tripped whether initiation is automatic or manual.

Because the cell will be secured, any releases can be controlled and, if necessary, evacuation from nearby buildings can be effected before exceeding the limits of 10 CFR 20. For the so-called credible accident, described above, and other less serious accidents, the need for evacuation of large areas is totally unnecessary.

5.0 EMERGENCY ACTION LEVELS (EALs)

There are no credible accident scenarios that lead to exposures exceeding the guideline limits of 0.5 rem to the whole body or 3 rem to the thyroid for any individual beyond the operations boundary,[5] as shown by the values quoted in Table 1.1. Protective action guides for the general public and onsite personnel beyond the operations boundary are inappropriate. Somewhat similar concepts are employed internally for assessment of emergency status, as shown in Table 5.1. Emergency Action Levels specified in Table 5.1 and described in subsections 7.2.1, 7.3.1 and 7.4.1 of Section 7.0 are considered to be EALs for activating the emergency organization and the initiation of protective actions at the level appropriate for addressing the emergency event in question.

6.0 EMERGENCY PLANNING ZONE (EPZ)

Emergency planning zones for the UFTR are unnecessary as there are no credible accidents which lead to exposures exceeding the guideline limits of 0.5 rem to the whole body or 3 rem to the thyroid in any regions beyond the operations boundary. [3] However, simply for planning purposes, the operations boundary is established as an EPZ to conform with Table 2 of ANSI/ANS-15.16 [3] which represents an alternate method for determining the size of the EPZ. As indicated in the standard, this EPZ is selected based upon the postulated releases from credible accidents. In addition it should be noted that the UFTR authorized power level of 100 kW is well below the 2 MW limit in Table 2 for which the acceptable EPZ size is the operation boundary. The size of the area within the operations boundary is large enough to provide a response base that would support actions outside this area should this ever be needed. The predetermined protective actions for the EPZ are described in Sections 7.2.4 and 7.3.4.

Table 10.2

UFTR Safety System Operability Tests

Component or Scram Function	Frequency
Log-N period channel Power level safety channels	Before each reactor startup following a shutdown in excess of 6 hr, and after repair or deenergization caused by a power outage
10% reduction of safety channels high voltage	4/year (4-month maximum interval)
Loss of electrical power to console	4/year (4-month maximum interval)
Loss of primary coolant pump power	4/year (4-month maximum interval)
Loss of primary coolant level	4/year (4-month maximum interval)
Loss of primary coolant flow	4/year (4-month maximum interval)
High average primary coolant inlet temperature	With daily checkout
High average primary coolant outlet temperature	With daily checkout
Loss of secondary coolant flow (at power levels above 1 kW)	With daily checkout
Loss of secondary coolant well pump power	4/year (4-month maximum interval)
Loss of shield tank water level	4/year (4-month maximum interval)
Loss of power to vent system and dilution fan	4/year (4-month maximum interval)
Manual scram bar	With daily checkout