FINAL SAFETY ANALYSIS REPORT

submitted to the

U.S. NUCLEAR REGULATORY COMMISSION

In Partial Fulfillment
of the Requirements for
a Class 104 License Pursuant to

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OUTLINE - UFTR SAFETY ANALYSIS REPORT

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1. INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The University of Florida Training Reactor (UFTR) is located on the campus of the University of Florida at Gainesville, in Alachua County, Florida. Gainesville is approximately in the center of Alachua County, which covers 961 square miles in the north-central part of Florida. The University of Florida campus is located approximately one mile from the center of the city of Gainesville.

The UFTR is a modified Argonaut type reactor, a light water and graphite moderated, graphite reflected, light water cooled reactor. The UFTR is currently licensed for 100 Kw (thermal) steady state power with a maximum power of 125 Kw (thermal) limited by the protection system. The UFTR originally operated from December 1959 under License Number R-56 at power levels up to the maximum of 10 Kw; in 1964, the license was amended to allow operation at power levels up to the current 100 Kw rating.(1)

The information and analyses presented in this Safety Analysis Report show that the UFTR can continue to be operated at 100 Kw (thermal) rated power without undue risk to the health and safety of the public.

1.2 General Description

The University of Florida campus is located in the Southwestern quadrant of the greater Gainesville area which has a population of about 125,000. The population within the city limits in early 1980 is about 83,000. It is approximately one mile from the center of the city (University Avenue and Main Street).(2)

The University of Florida was established by an act of the Florida Legislature in 1905, and has a current enrollment of about 30,500 students in the winter quarter of 1980 (March, 1980). Enrollment by quarters for the preceding full year has been as follows:

Winter Quarter,	1979........	29,384 students
Spring Quarter,	1979........	27,997 students
Summer Quarter,	1979........	16,131 students
Fall Quarter,	1979........	32,314 students

Expected continued but slow growth will make these figures representative for several years.

The UFTR is located on campus in the immediate vicinity of the buildings housing the College of Engineering and the College of Journalism. The Nuclear Sciences Center, which houses the Department of Nuclear Engineering, is annexed to the reactor building. Normal access to the reactor building is through the doors leading to the Nuclear Sciences Center. Authorized personnel may also enter the reactor building by other routes through normally locked doors on a keyed basis only. Ordinary access by these alternate routes is restricted to approved personnel by keeping the other doors to the reactor building locked at all times.

Most of the Gainesville area, including the site of the training reactor is underlain by a loamy fine-sand type of soil derived from residual weathering of the "Hawthorne Formation". Except where buildings and landscaping intervene, the present contour of the site rises on a 16 percent slope from west to east; consequently, the reactor is partially buried in the side of a hill. The construction of the reactor facility, access control, and standard procedures are designed to prevent or minimize injury in the event of aircraft crash, civil disturbance, attempted sabotage and other externally-derived events. (2)

The UFTR is of the general type known as the Argonaut, with some modifications to adapt it to a university training program by improving shielding and minimizing the possibility of accident. The reactor is heterogeneous in design, currently using 93 percent enriched uranium-aluminum fuel elements. (Design and safety analyses are currently underway to investigate the possibility of using $\sim$ 4.8% enriched SPERT fuel rods in the UFTR core. This analysis for the fuel change will be completed in the near future.) Water is used as the coolant and also as moderator. The remainder of the moderator consists of graphite blocks which surround the boxes containing the fuel plates and the water moderator. The fuel is contained in MTR-type plates assembled in bundles. Each bundle is composed of 11 fuel plates, each of which is a sandwich of aluminum clad over a uranium-aluminum alloy "meat". There are four control blades (3-safety and 1-regulating), of the swinging-arm type, consisting of four cadmium vanes protected by magnesium shrouds which 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 reactor facility with shielding removed is presented in Figure 1-1.

The biological shield is made of cast-in-place concrete with sections of barytes carefully located to reduce the overall shield thickness. Access to the ends and top of the reactor is provided by removal of ordinary concrete blocks cast to fit openings.

The reactor core has a two slab geometry and is presently composed of 21 fuel bundles and 3 dummy bundles arranged in six water-filled aluminum boxes, surrounded by reactor grade graphite.

All reactor operations are supported by the following systems:

1. Reactor instrumentation, protection and control
2. Primary coolant system

3. Secondary coolant system
4. Primary water make-up system
5. Purification system
6. Reactor vent system
7. Shield water tank system
8. Radiation monitoring system
9. Radioactive waste disposal system

The primary coolant (demineralized water) is pumped upward around the fuel plates and then fed by gravity through the side orifices to the heat exchanger, where the primary coolant transfers the heat from the reactor. The heat is removed by the secondary coolant system to the storm sewer. There is no mixing of water between the two systems.

The reactor protection system provides reactor trips that can be classified into two groups; nuclear instrument and process instrument-type trips. The nuclear-type trips are full reactor trips, causing the dumping of the primary water besides the standard drop of control blades, and include:

1. Fast period
2. Exceeding maximum allowable power (125%)
3. A 10% reduction of high voltage to the neutron chambers.

Process instrument-type trips, also called rod-drop trips, cause the drop of control blades without dumps of the primary water, and include the nine (9) items in the following list:

1. Loss of power to the reactor vent blower system
2. Loss of power to the reactor vent diluting system
3. Loss of power to the reactor secondary system deep well pump when at or above 1 Kw
4. Loss of power to the primary coolant pump
5. Drop of secondary flow below 60 gpm
6. Drop in shield water tank below set point
7. Reduction of primary coolant flow below 30 gpm (inlet)
8. Loss of primary coolant level (outlet)
9. High temperature of primary coolant returning from the reactor.

As usual, manual reactor trip is also available at all times.

The Radiation Control Office is responsible for implementing the radiation protection program. Aside from this task, the Radiation Control Office performs the following services for the reactor:

1. Personnel monitoring service
2. Radiation instrument calibration and maintenance
3. Radioactive material handling and safety procedures
4. Decontamination
5. Personnel records
6. Solid and Liquid Radioactive Waste Disposal.

Although the Radiation Control Office provides solid radioactive waste disposal service, labeling and bagging of waste is the responsibility of the UFTR personnel. All pertinent information must be provided to this office by the UFTR personnel. These and any other matters concerning radiation and safety procedures are covered in detail in the "Standard Operating Procedures" manual of the UFTR. (3)

The major experimental facilities in the UFTR are illustrated in the vertical view line drawing of the UFTR shown in Figure 1-2 and include:

1. Sixteen (16) vertical foil slots placed at intervals in the graphite between the fuel compartments, each are 3/4 in. x 1 in.
2. Three (3) vertical experimental holes located centrally with respect to the six (6) fuel compartments (boxes):

   i) Center Vertical Port (CVP) with 2 inch diameter
   ii) West Vertical Port with 1 1/4 inch diameter
   iii) East Vertical Port with 1 1/2 inch diameter

3. Five (5) vertical square holes filled with 4 inch x 4 inch removable graphite stringers;
4. A horizontal thermal column having six (6) 4 inch x 4 inch removable stringers flanked on each side by 2 additional thermal column positions with removable stringers which are infrequently used;
5. A shield tank placed against the west face of the reactor opposite the fuel boxes and thermal column;
6. Six (6) horizontal openings, 4 inches in diameter, located symmetrically on the center plane of the reactor and normally filled with shield plugs, only one of which (south) goes all the way to the core region;
7. A horizontal throughport consisting of a 2.05 inch ID aluminum tube with 20 ft. length running east-west across the reactor. Shield plugs or other shielding appropriate to experiments in progress are normally inserted into these ports which are clearly identified in Figure 1.2.

As quoted in Section 1.3.1, the safety rods have the following current experimentally verified reactivity worths as of March 1980:

Safety 1 with $\approx$ 1.4% $\Delta k/k$
Safety 2 with $\approx$ 1.3% $\Delta k/k$
Safety 3 with $\approx$ 2.2% $\Delta k/k$

and the regulating blade has a total worth of $\sim$1.0% $\Delta k/k$. The maximum allowable worth of any single unconstrained experiment is 0.6% reactivity. The measured shutdown margin with the most reactive blade out is

$\sim 2.9\% \Delta k/k$.

The UFTR is a reactor used for instructional and university research activities, therefore it is designed so that safety is maximized without excessive restraints on the different activities planned. As quoted in Reference 3, the inherent safety of the UFTR is based on four design features. First, the amount of excess reactivity in the reactor is limited to less than 2.3% $\Delta k/k$. Second, the reactor has negative temperature and void coefficients. In addition, the reactor is provided with sufficient interlocks and safety trips to make a hazardous incident extremently improbable.

Third, the amount of contained fission products is relatively small. And fourth, there is an extremely low probability that these fission products can escape. Nevertheless, because of the high population density of the campus, the reactor is housed in a structure with a minimum number of penetrations sealed against gas leakage. A negative pressure is maintained in the reactor building such that all gaseous effluent within the cell is withdrawn by means of the reactor vent system through a filter system which is continuously monitored for radiation activity.

Possible failures or accident situations have been analyzed and discussed in Chapter (15), including the effects of a rapid reactivity insertion, radioactive fission product release and loss of coolant flow in the case of 100 Kw (thermal) operation of the UFTR.

1.3 Comparison Tables

1.3.1 Comparison with Similar Facility Designs

The UFTR which has been operational since May, 1959, is currently licensed for operation at 100 Kw (thermal).

Similar functional, licensed reactors are located at the University of California, Los Angeles - (UCLA), at the University of Washington in Seattle, Washington, at the Virginia Polytechnic Institute at Blacksburg, Virginia and in the United Kingdom. A comparison of the nuclear characteristics of the UFTR to those of the UCLA Nuclear Reactor is shown in Table 1-1. The UCLA Nuclear Reactor was chosen because of the great similarity between the UCLA R-1 reactor and the UFTR as briefly described in the following paragraphs.

The 100 Kw UCLA Argonaut Reactor (UCLA R-1) consists of a core of six aluminum boxes arranged in two parallel rows of three boxes each, the rows being separated by and surrounded with graphite. Four fuel bundles are placed within each box, each bundle consisting of 11 uranium-aluminum alloy fuel plates clad with aluminum. The graphite on one side of the reactor is extended to provide a thermal column, and on the opposite side is placed a water shield tank as in the UFTR design. Completely surrounding the shield water tank, thermal column, and core is a concrete shield of external dimensions approximately 18 feet in all directions equipped with several beam ports and access tubes. The UFTR also has such a concrete shield.

The primary coolant of the UCLA Argonaut reactor as with the UFTR is demineralized water which is pumped upward over the fuel plates and is then fed by gravity to the system heat exchanger where it meets the secondary coolant flowing directly from the city water line. The secondary coolant flows from the heat exchanger to a hold-up tank with a retention time of approximately 15 minutes before it is dumped into a municipal storm drain. The coolant system for the UCLA R-1 Reactor is shown in Figure 1-3 (4); it is very similar to the UFTR cooling system presented in Chapter 5 of this Safety Analysis Report.

The nuclear characteristics of the UFTR are also similar to those of other water-moderated reactors using similar fuel plates such as the LITR, MTR, BSTF, Borax I, II and III, and Argonaut. (5)

1.3.2 Comparison of Final and Preliminary Information

This Safety Analysis Report is submitted for license renewal without substantive changes from the previously licensed, with approved modifications, UFTR reactor system. As such this current Safety Analysis Report stands as the FSAR for the UFTR license renewal effort.

1.4 Identification of Agents and Contractors

No modifications are necessary for relicensing the UFTR for 100 Kw operation. Therefore, no agents or contractors need to be identified at this time. Plans to increase the maximum power of the UFTR to 500 Kw have been considered but are not yet nearing finalization. A study of the releases associated with the "design basis accident" at this higher power operation has been partially completed but the actual redesign of the core is still in the analysis stage. The ~4.8% enriched SPERT fuel rods are being investigated for this purpose.

1.5 Requirements for Further Technical Information

This Safety Analysis Report is serving as both Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report for the UFTR facility because it is not a new design. The UFTR is already licensed and has been operational since May, 1959 when it was first licensed to operate at 10 Kw (License Number R-56). The current SAR is submitted to support relicensing of the existing system as currently operated at a rated power of 100 Kw under License Number R-56 Amendment Number 8 effective January 28, 1964. No changes are being proposed in this SAR. No further technical information should be required in support of the issuance of the renewed Operating License at 100 Kw (thermal).

Table 1-1

Comparison Table
Argonaut Reactor Characteristics

	UFTR*	UCLA R-1**
Type	Heterogeneous, Thermal	Heterogeneous, Thermal
Thermal Power	100 Kw	100 Kw
Flux Level (at 100 Kw)	1.8×10^{12} n/cm^{2}sec in center vertical port	1.5×10^{12} n/cm^{2}sec - Thermal 1.8×10^{12} n/cm^{2}sec - Epithermal 2.0×10^{10} n/cm^{2}sec - Fast
Excess reactivity	1.00% $\Delta k/k$	1.85% $\Delta k/k$
Clean, cold critical mass	3.07 kg U-235	3194.4 gm U-235 (Ref. 5)
Effective prompt neutron lifetime	2.8×10^{-4} sec	2×10^{-4} sec
Uniform water void coefficient	-0.2% $\Delta k/k$/% void	-0.164% $\Delta k/k$/% void
Temperature Coefficient	-0.3×10^{-4} $\Delta k/k/°F$	-0.865×10^{-4} $\Delta k/k/°C$
U-235 Mass coefficient	0.4% $\Delta k/k$/% U-235 mass	0.3% $\Delta k/k$/% U-235 mass
Startup Source	$\leq$ 25 curies Sb-Be 1 curie Pu-Be	6.6 millicurie Ra-Be
Reflector	Graphite (1.6 gm/cc)	Graphite (1.6 gm/cc)
Moderator	H_2O and graphite	H_2O and graphite

*Values for the UFTR system are taken primarily from Reference 4 except for those based on more current records and determinations.

**Values for the UCLA R-1 reactor system are taken from UCLA R-1 reactor characteristics chart dated April, 1978 (6) plus Howard's Thesis on redesign of the UCLA R-1 system (7) where the information was not available in the characteristics chart of April, 1978.

	UFTR	UCLA R-1
Fuel		
Fuel Assembly	24 bundles with 11 plates/bundle	24 bundles with 11 plates/bundle
Fuel Material	U-Al alloy	U-Al alloy
Fuel Enrichment	93% enriched	19 bundles: 93.18% U-235 5 bundles: 93.123% U-235
Fuel Loading	3354.61 gm U-235	3356.86 gm U-235, excluding burnup
Plate Thickness	0.070 in.	0.070 in.
Thickness U-Al	0.040 in.	0.040 in.
Thickness Clad	0.015 in.	0.015 in.
Plate Width	2.845 in.	2.845 in.
Plate Length	25.625 in.	25.625 in.
Water Channel Spacing	0.137 in.	0.137 in.
Al to H_2O Volume Ratio	0.49	0.51
"Meat" composition	14.5 wt. % U-Al alloy	13.4 wt. % U-Al alloy
Coolant		
Type	Demineralized H_2O	Demineralized H_2O
Minimum Resistivity	5×10^5 ohm-cm	5×10^5 ohm-cm
Normal Resistivity	$\sim 1 \times 10^6$ ohm-cm	$\sim 1 \times 10^6$ ohm-cm
Primary Flow (at 100 Kw)	40 gpm (scram at 30 gpm)	16.0 gpm
Secondary Flow	200 gpm (nominal)	22.5 gpm

	UFTR	UCLA R-1
Coolant (continued)		
Primary Equilibrium Temperature Inlet (100 Kw)	86°F ± 2°F	100°F ± 5°F
Primary Equilibrium Temperature Outlet (100 Kw)	103°F ± 2°F	142°F ± 5°F
Secondary Well Water Equilibrium Inlet and Outlet Temperatures (100 Kw)	∿73°F/∿77°F	-
Control Blades		
Type	Cd, swinging vane, gravity fall	Cd, swinging vane, gravity fall
Number	3 safety, 1 regulating	3 safety, 1 regulating
Insertion Time	1.0 sec (maximum)	1.0 sec (maximum)
Removal Time	∿100 sec (minimum)	∿100 sec
Blade Worth, Safeties	Safety #1=1.4% Δk/k Safety #2=1.3% Δk/k Safety #3=2.2% Δk/k	Safety #1=1.56% Δk/k Safety #2=1.68% Δk/k Safety #3=1.60% Δk/k
Blade Worth, Regulating	∿1.0% Δk/k	1.01% Δk/k
Minimum Shutdown Margin (actual)	∿2.9%	2.31%
Reactivity addition rate, maximum allowed	0.06% Δk/k/sec	.077% Δk/k/sec
Shield (concrete)		
Sides, center	6 ft., cast, barytes	6 ft., cast, magnetite
Sides, ends	6 ft. 9 in., cast barytes	6 ft. 8 in. cast, magnetite

	UFTR	UCLA R-1
Shield (concrete) (continued)		
Middle	Barytes concrete blocks	Cast concrete blocks
Top	5 ft. 10 in.	5 ft. 10 in. magnetite blocks
End	3 ft. 4 in.	3 ft. 4 in. magnetite blocks
Experimental Facilities		
Thermal column, horizontal	60 in x 60 in x 56 in high	60 in x 52 in x 43 in long (removable)
Thermal column, vertical	2 ft. diameter x 6 ft.; H_2O or D_2O	Provision for installation
Shield test tank	5 ft. x 5 ft. x 14 ft. high	5 ft. x 5 ft. x 14 ft. 6 in. deep
Experimental holes	6 horizontal, 4 in diameter 5 vertical, 4 in x 4 in 3 vertical, 2,1 1/2, 1 1/4 in diameter	2 horizontal, 6 in diameter 4 horizontal, 4 in diameter 3 vertical, 1 7/8 in diameter
Foil Slots	16 vertical, 3/8 in x 1 in	16 vertical, 3/8 in x 1 in
Horizontal Throughport	2.05 in. ID x 20 ft length	-
Removable thermal column dry room	-	56 in x 56 in x 40 in long (east-west)
Shield Ventilation	250 cfm, room air	-

1.6 Material Incorporated by Reference

The following documents which have been referenced throughout this report can be found under Docket Number 50-83:

1. University of Florida Training Reactor Security Plan
2. University of Florida Training Reactor Standard Operating Procedures

1.7 Electrical, Instrumentation, and Control Drawings

Electrical instrumentation and control (EI&C) drawings for the UFTR reactor system are taken from Reference presented in Figures 1-4 through 1-9 in this section. For uniformity of nomenclature, abbreviations used in the drawings for the UFTR are defined in Table 1-2.

TABLE 1-2

ABBREVIATIONS USED IN UFTR

ELECTRICAL, INSTRUMENTATION AND CONTROL DRAWINGS

AMM	AMMETER
AMP	AMPLIFIER
AUTO	AUTOMATIC
B/S	BISTABLE
CAL	CALIBRATE
CIC	COMPENSATED ION CHAMBER
COMPA	COMPARATOR
COMPUT	COMPUTER
CPS	COUNTS PER SECOND
DN	DOWN
HV	HIGH VOLTAGE
INT'LK	INTERLOCK
LIN	LINEAR
LOG	LOGARITHMIC
MAG	MAGNETIC CLUTCH
MAN	MANUAL
NI	NUCLEAR INSTRUMENTATION
P/S	POWER SUPPLY
PA	POWER AMPLIFIER
PC	PRIMARY COOLANT
PWR	POWER
REG	REGULATING ROD
RPI	CONTROL BLADE (ROD) POSITION INDICATION
UIC	UNCOMPENSATED ION CHAMBER
W/D	WITHDRAWAL
W/R	WIDE RANGE DRAWER (CHANNEL)

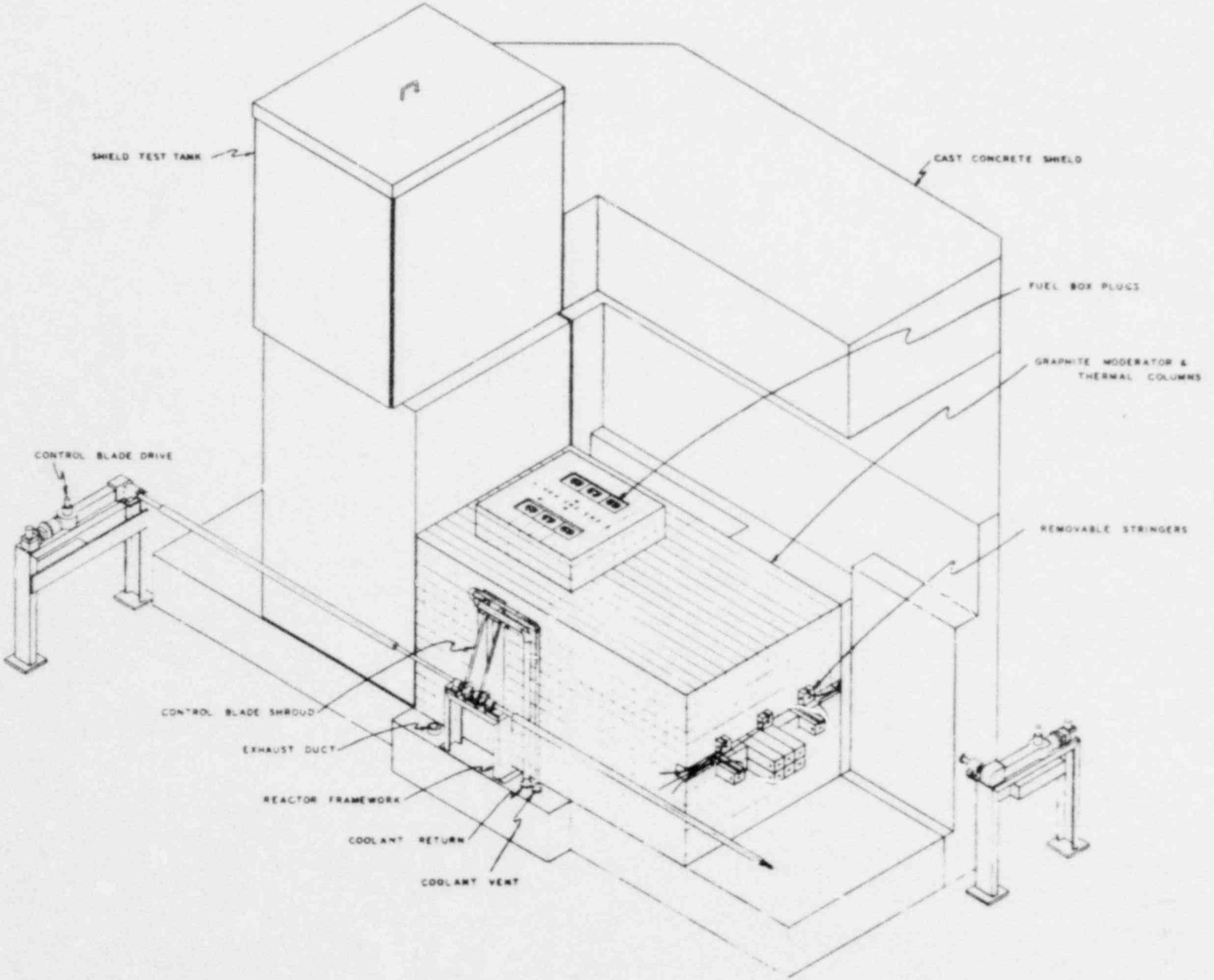


Figure 1-1. Isometric Sketch of the UFTR with Shielding Removed.

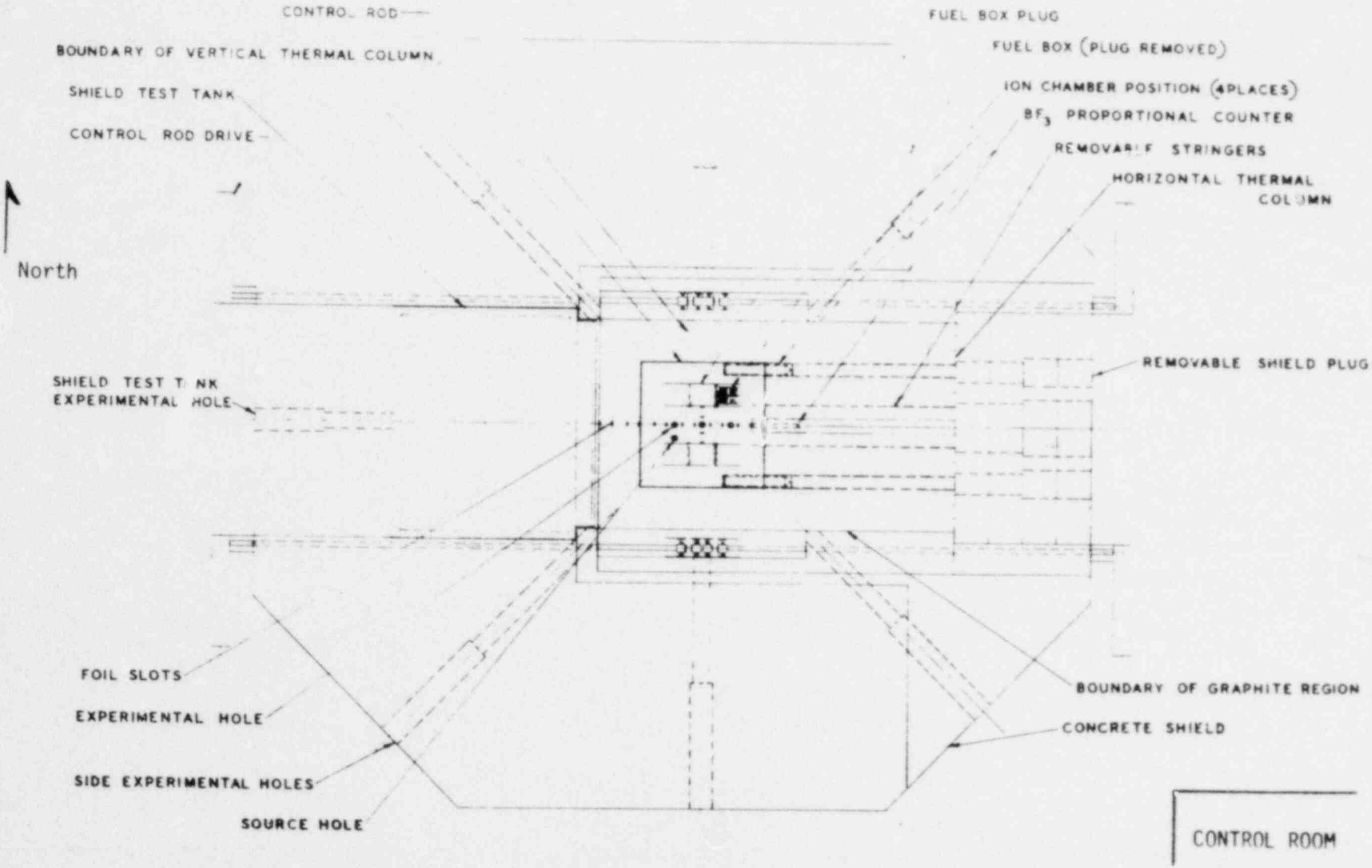


Figure 1-2. Vertical View Line Schematic of Major Experimental Facilities in the UFTR.

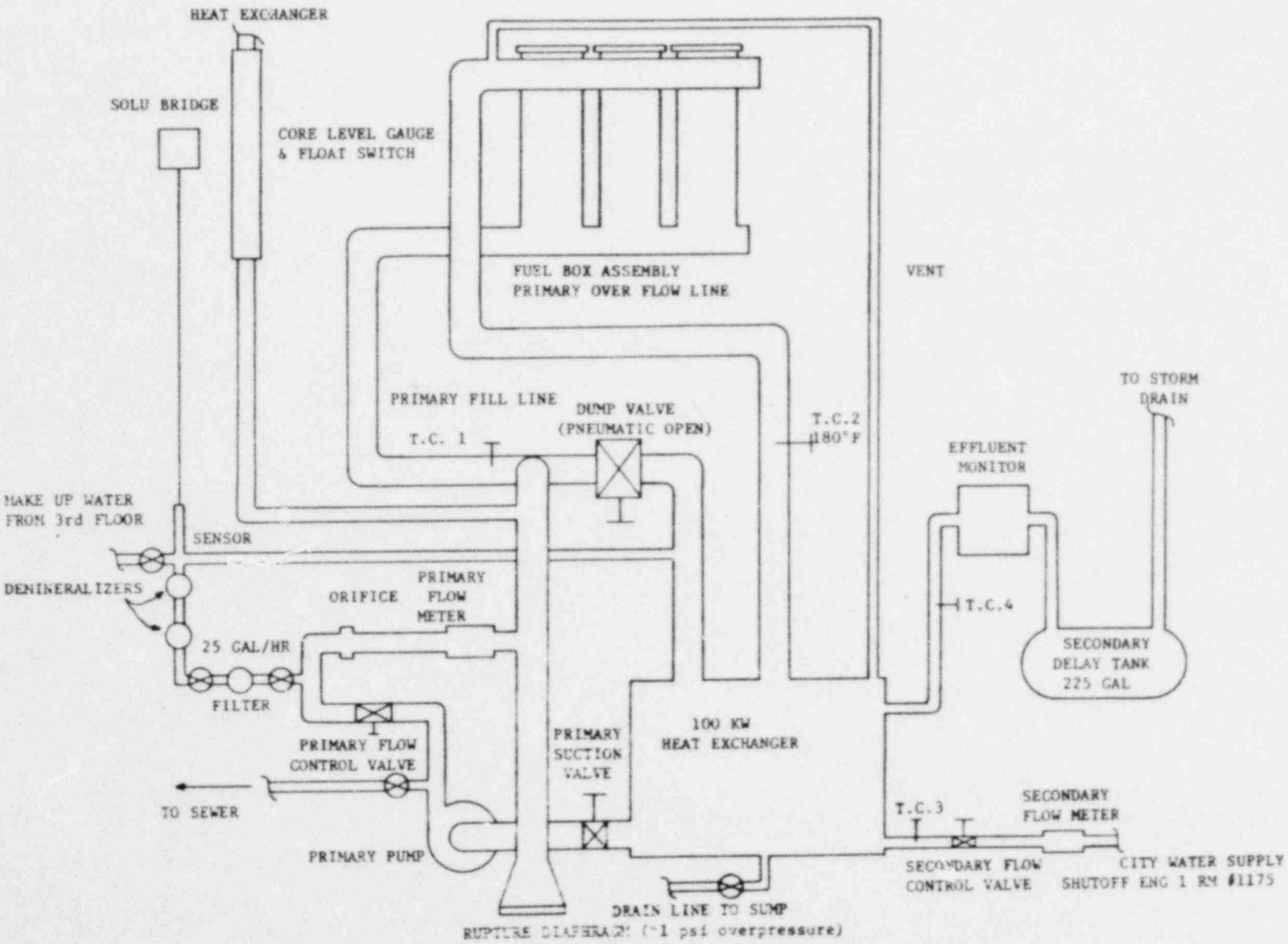


Figure 1-3. UCLA R-1 Reactor Coc ing Systems.

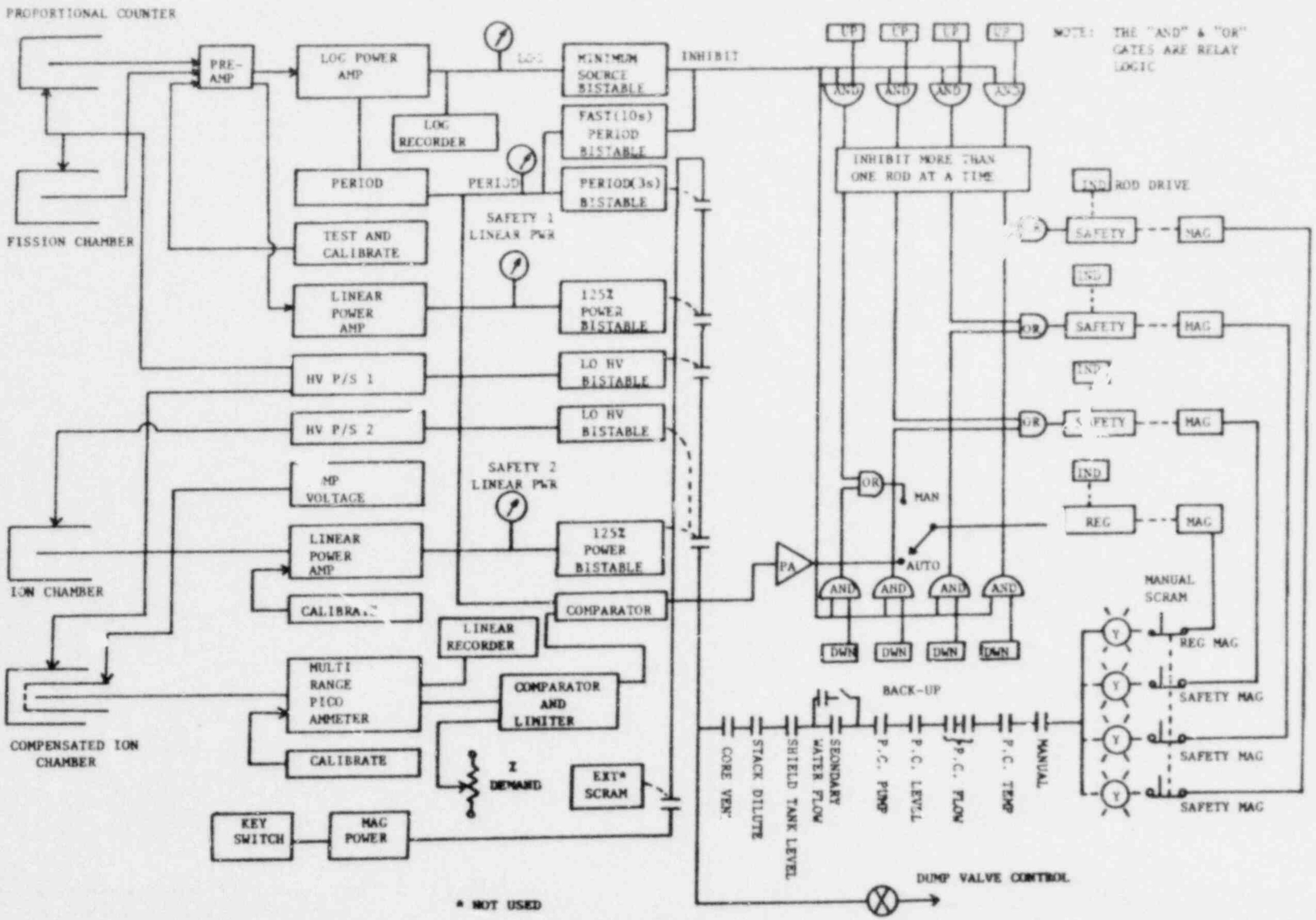


Figure 1-4. UFTR Instrumentation and Scram Logic Diagram.

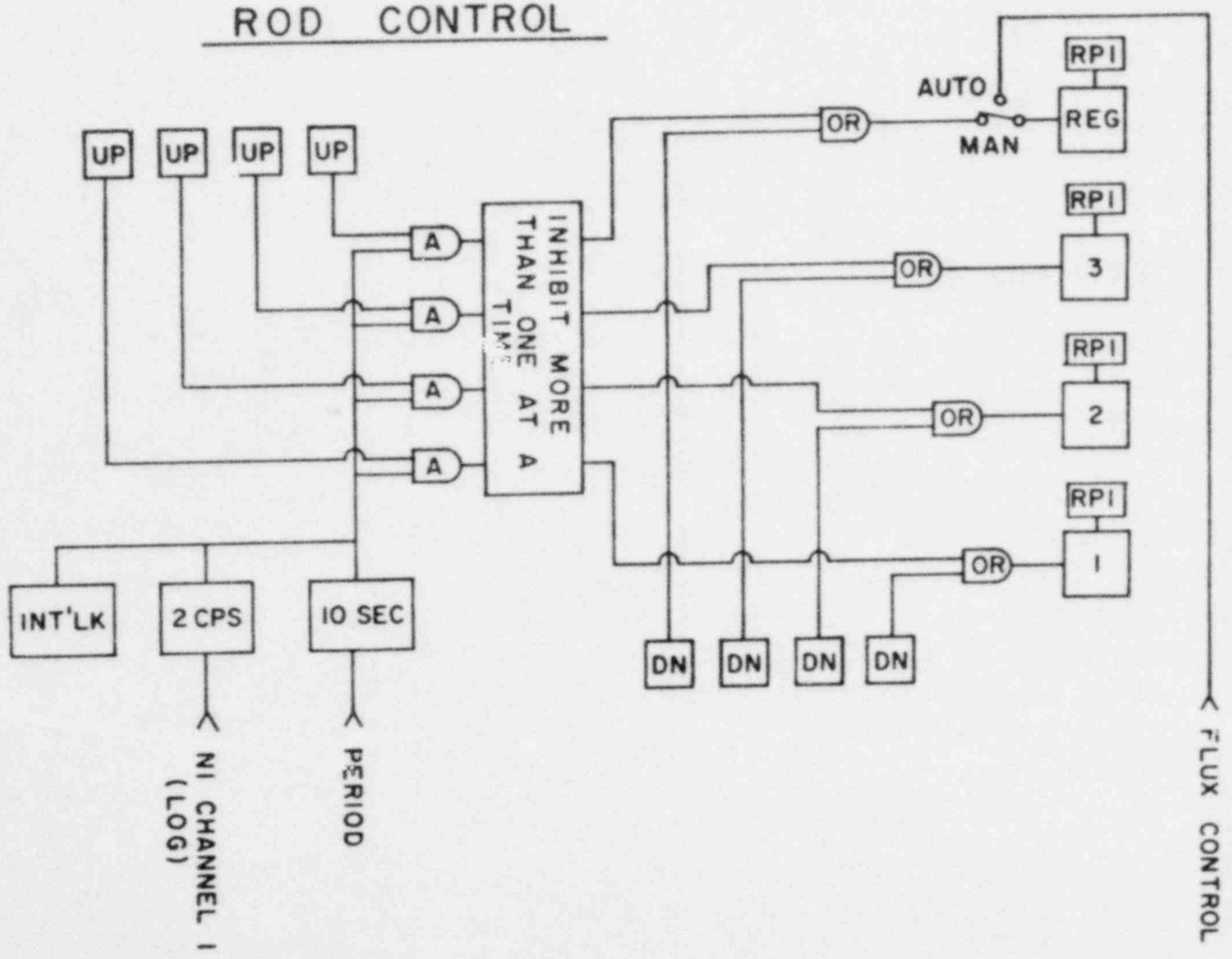


Figure 1-5. ROD CONTROL: UFTR Control Blade (Rod) Withdrawal Inhibit System.

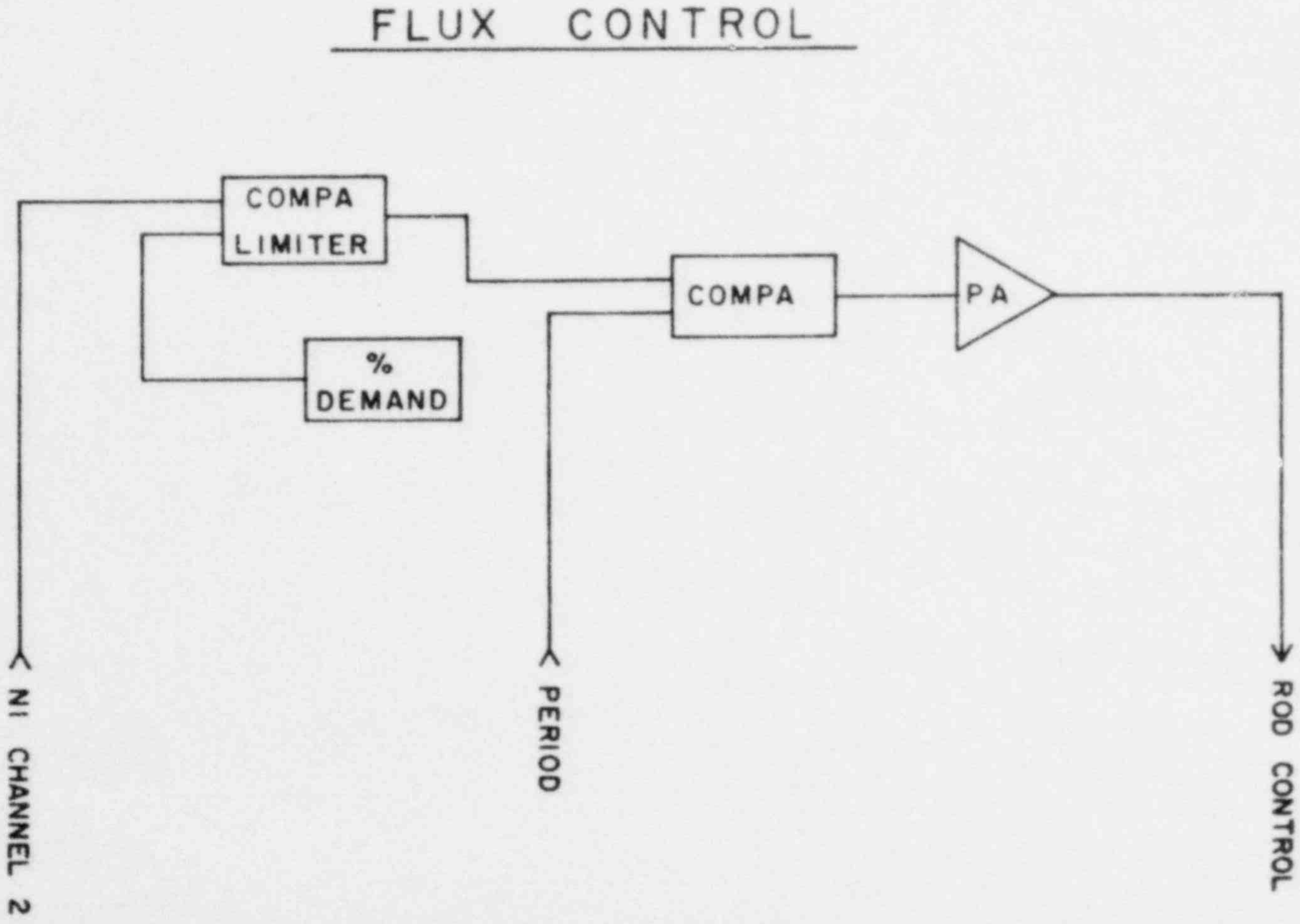


Figure 1-6. FLUX CONTROL: Schematic for UFTR Neutron Flux Control System.

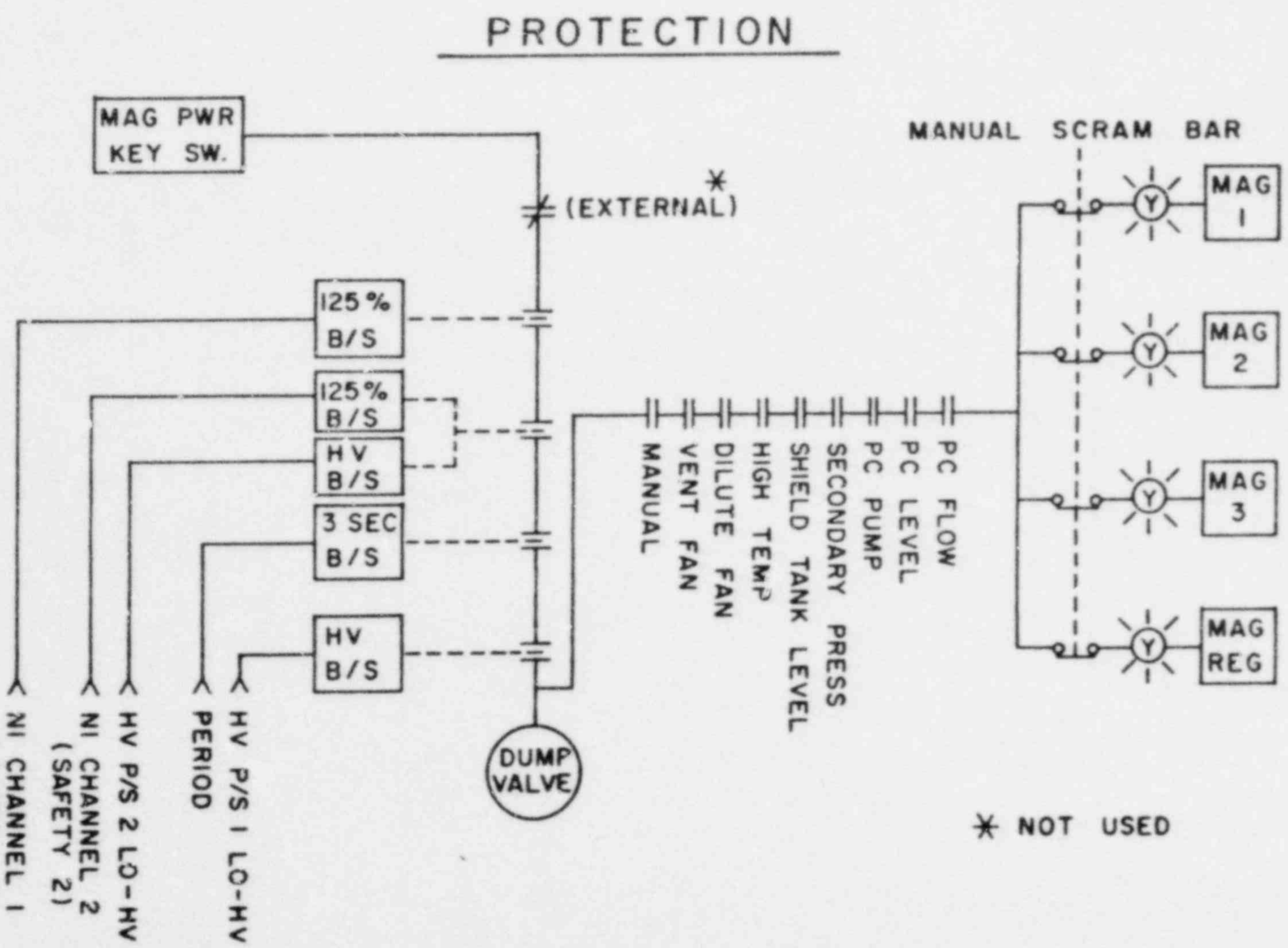


Figure 1-7. PROTECTION: Schematic Diagram of the UFTR Reactor Protection System.

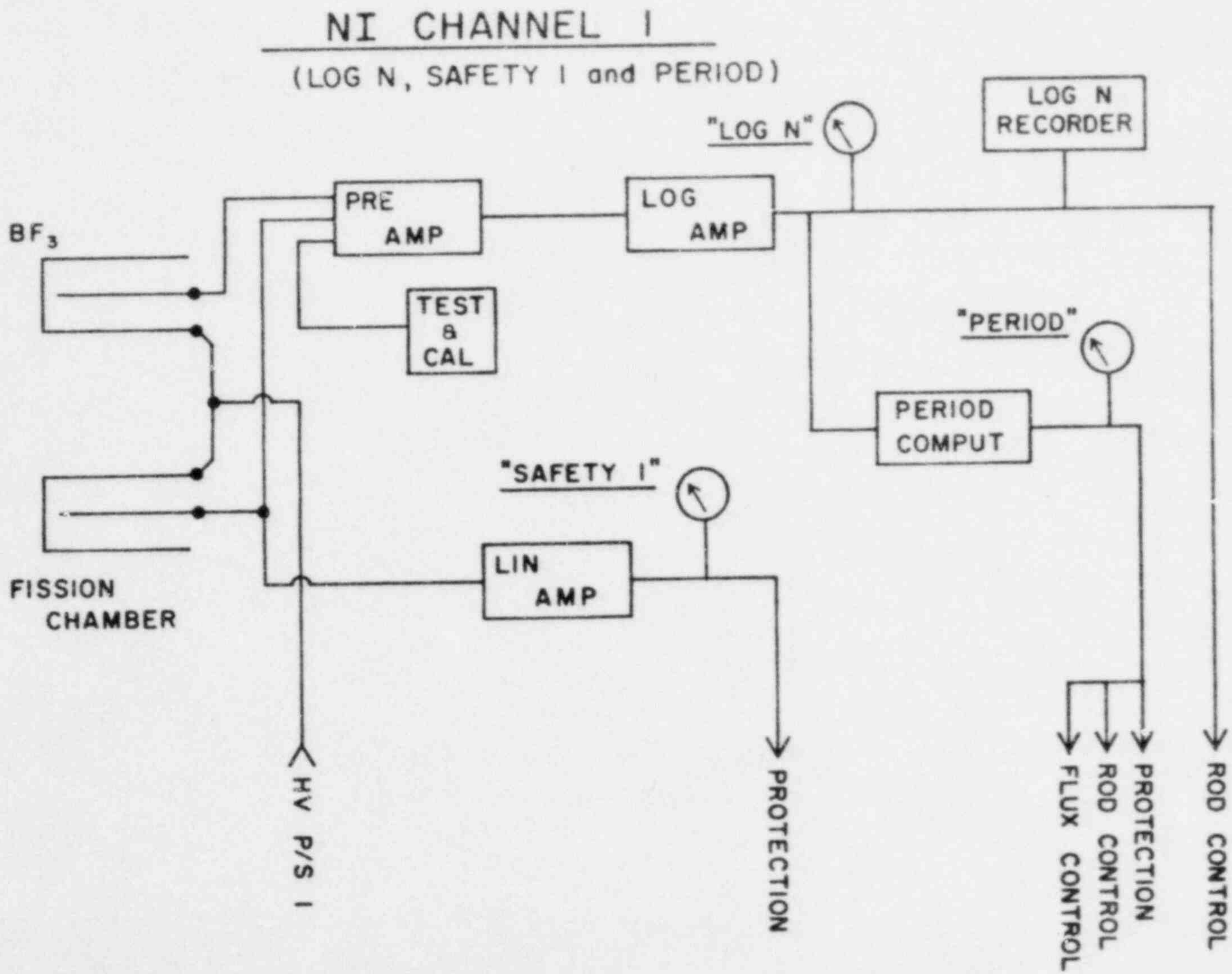


Figure 1-8. NI CHANNEL 1: UFTR Nuclear Instrumentation Channel 1 Diagram (Log N, Safety #1 and Period Channels).

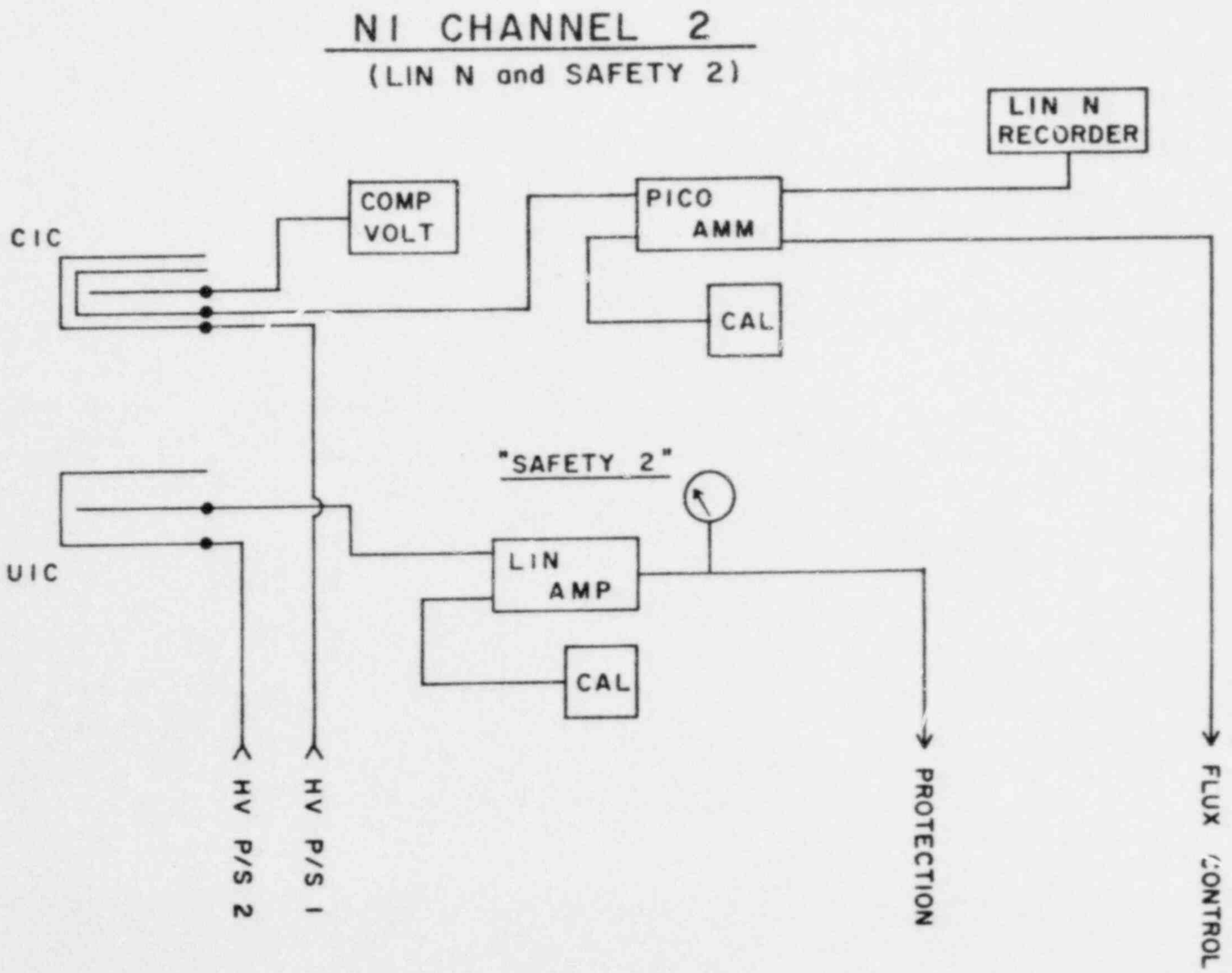


Figure 1-9. NI CHANNEL 2: UFTR Nuclear Instrumentation Channel 2 (Linear N and Safety #2 Channels).

2. SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification of Location. The UFTR is located on the campus of the University of Florida, Alachua County. Figure 2-1 shows the geographic location of Alachua County with Gainesville at its center in the North Central portion of the Florida peninsula. Figure 2-2 shows the location of the University of Florida campus within the city of Gainesville. The city of Gainesville is approximately in the center of Alachua County, which covers 961 square miles in the north-central part of Florida, approximately midway between the Atlantic Ocean and the Gulf of Mexico. Gainesville is in the Central Highlands of the Florida peninsula. The nearest approach of the Gulf of Mexico is about 50 miles to the southwest, and the Atlantic Ocean is about 65 miles to the east. As shown in Figure 2-2, the University of Florida campus is in the southwestern quadrant of the greater Gainesville area which has a population of about 125,000. The city proper has a population of about 83,000. The UF campus is approximately one mile from the center of the city (University Avenue and Main Street).

The Nuclear Sciences Center is annexed to the reactor building which is labeled Building No. 557 in Figure 2-3. Concentric circles are shown with the UFTR as the center, the first circle having a 250 ft. radius and the rest being at 500 ft. increments from the central reactor building point. The site is 50 ft. south of Reed Laboratory (No. 131); the closest residence hall is East Hall which is approximately 750 ft. due west of the reactor building. The reactor is located about 600 ft. north of the J.W. Reitz Student Union, about 100 ft.west of the Journalism Building and 250 ft. due east of the Materials Building and about 95 ft. due east of the Westside Chiller Unit (Air Conditioner Cooling Tower). The J. Hillis Miller Health Center complex is about 3,000 ft. southeast of the UFTR. Similarly, most of the residence halls, fraternity houses, and Lake Alice, a small lake within the University of Florida boundaries, are found within the same range.

2.1.1.2 Site Area Map. The site map indicated in Figure 2-2 shows the property boundaries of the University of Florida campus. The site boundary lines are the same as the property lines. The locations of the principal existing structures on the University of Florida campus including the reactor building are shown in Figure 2-4.

The exclusion area for this reactor facility (as defined in 10 CFR Part 100) is the reactor building itself since this is a low power training and research reactor.

2.1.1.3 Boundaries for Establishing Effluent Release Limits. Under the regulations of 10 CFR 100, a restricted area is defined for the purpose of establishing access control to protect individuals from exposure to radiation

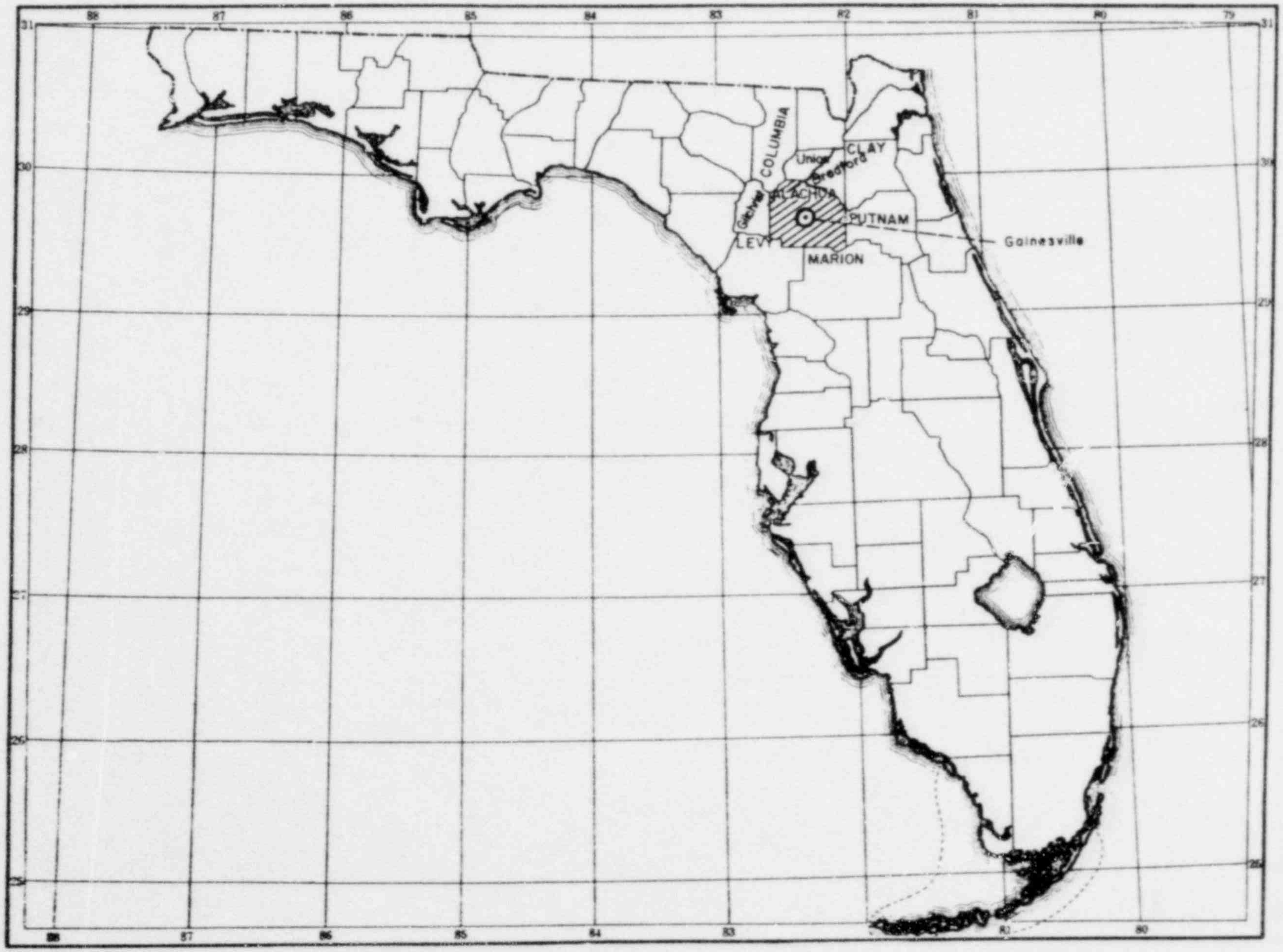


Figure 2-1. Relative Geographic Location of Alachua County and Gainesville in the State of Florida.

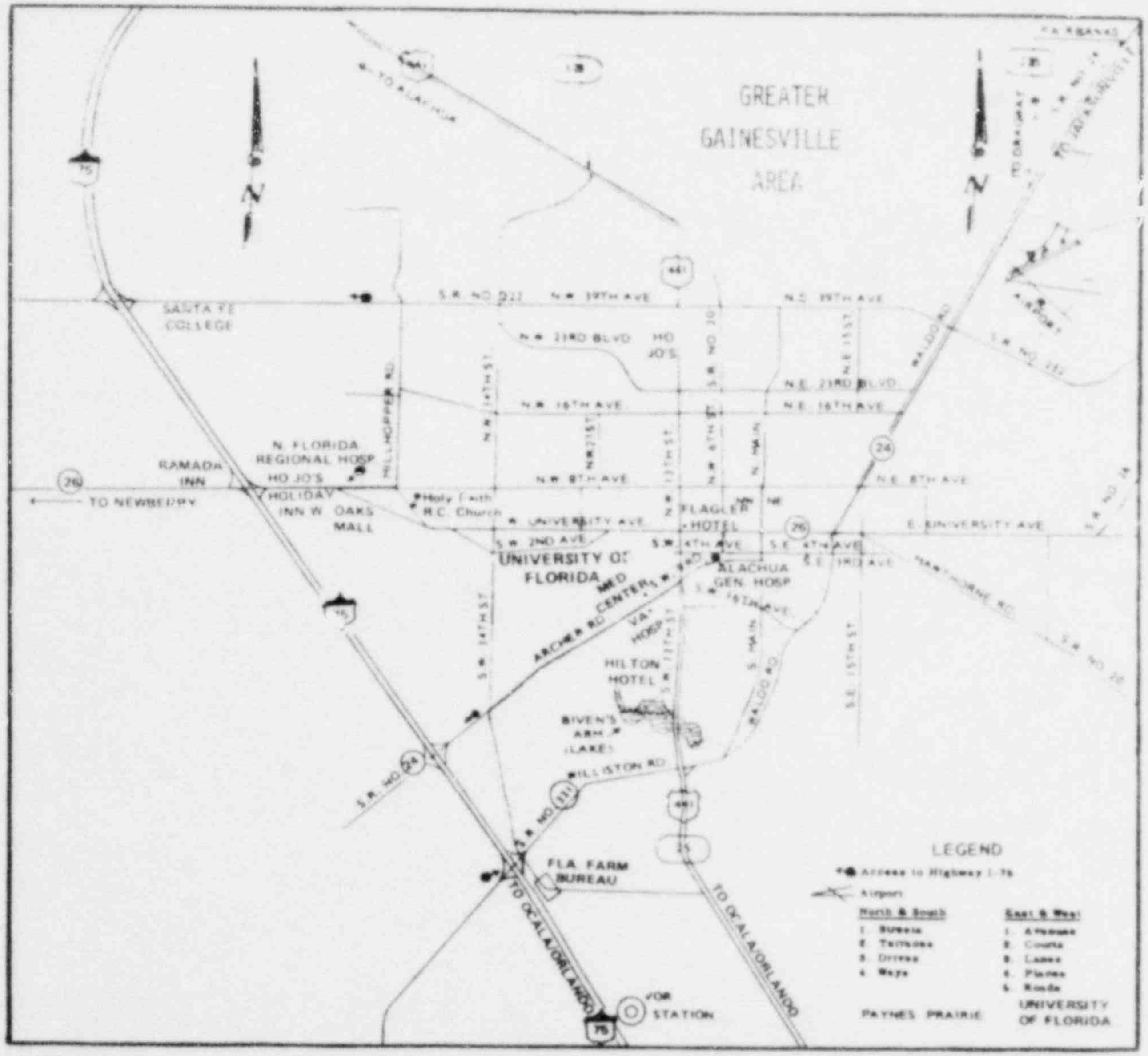


Figure 2-2. Map of Greater Gainesville Area Showing Placement of University of Florida Campus and Major Landmarks.

POOR ORIGINAL

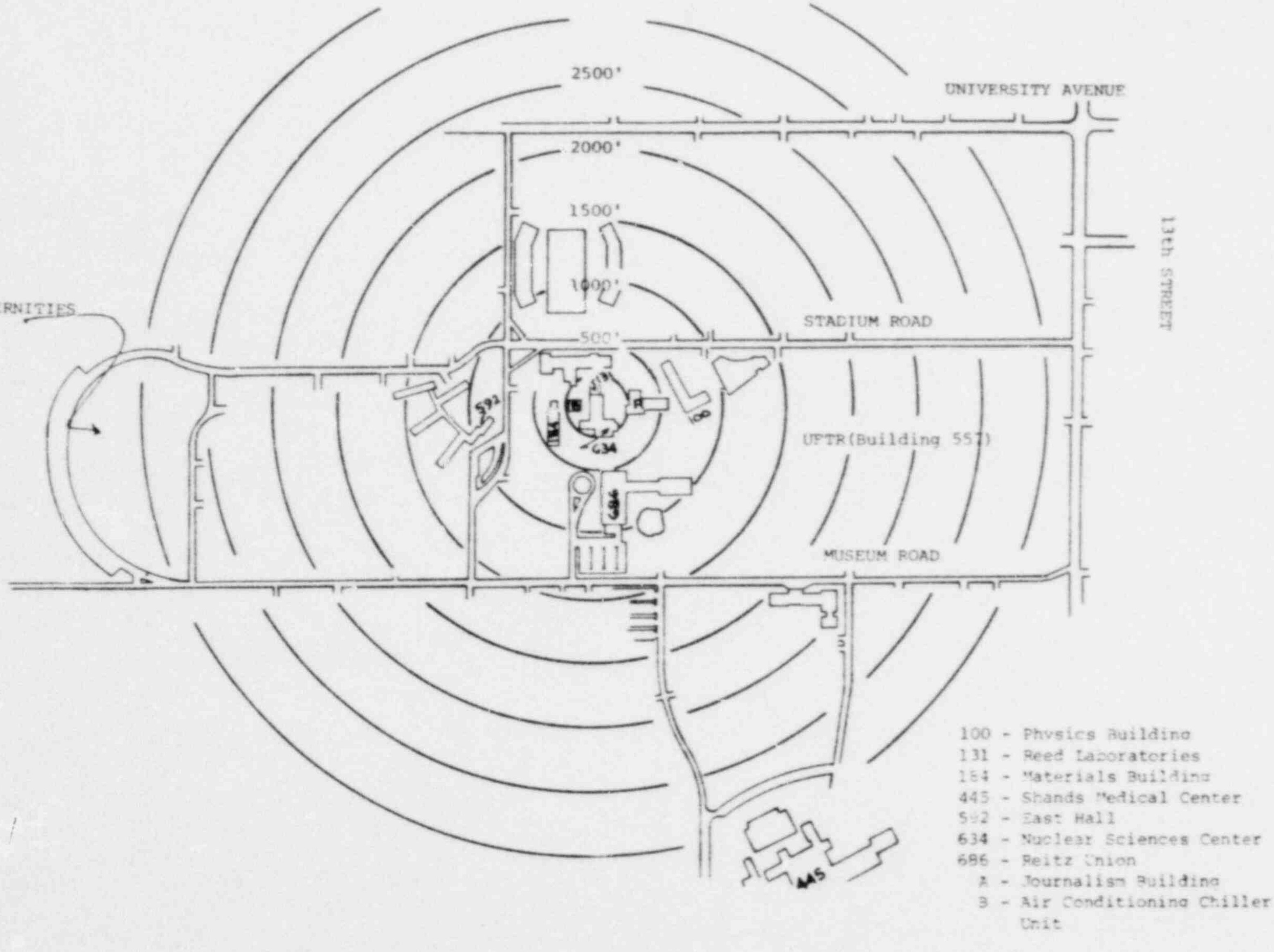


Figure 2-3. UFTR Building Placement on University of Florida Campus with Respect to Major Campus Arteries and Buildings.

Figure 2-4. University of Florida Campus Map With Building Locations, Primary Landmarks and Boundaries.

and radioactive materials. For the UFTR, the reactor cell itself constitutes the boundary lines of the restricted area. For this facility, a further "protective" zone is defined. This protective zone is established in the lobby of the reactor cell by locked doors under the operator's control. A locked door at the top of the stairs prevents unauthorized entrance from the laboratory and office facilities upstairs while a locked door downstairs prevents unauthorized entrance to the reactor cell lobby from the rest of the downstairs of the reactor building such as the radiochemistry laboratory.

For the UFTR, the reactor building itself constitutes the boundary lines of an exclusion area, usually thought of as the restricted area, in that personnel can be excluded from this building rapidly during an emergency situation and everyone in the reactor building is under the control of the UFTR operations staff.

The reactor building has five entrances (exits) but only two (one upstairs and one downstairs) leading from the Nuclear Sciences Center, will be in normal use and then only during normal work hours; the other three exits will be used only for emergency conditions or for authorized special circumstances such as off-site refueling, and will be kept secured <u>or</u> under control of a licensed operator at all times. Access to the exclusion area including the restricted area and the protective zone will be controlled according to the facility Security Plan. Only authorized personnel will be allowed to enter the reactor cell without the knowledge and permission of the reactor operator.

During non-use periods, the reactor cell will be kept locked. The construction of the reactor building as a "vault-type room" as defined in 10 CFR Part 73.2(o) means all doors are capable of being locked and the entire facility safeguarded from unauthorized access.

2.1.2 <u>Exclusion Area Authority and Control</u>

2.1.2.1 <u>Authority</u>. The University of Florida is located in the city of Gainesville, at Alachua County, approximately one mile from the center of the city (University Avenue and Main Street). The University of Florida was created by an Act of the Florida Legislature in 1905, and has a winter '80 quarter enrollment of about 30,500. The maximum enrollment at about 32,500 occurs during the fall quarter. Direct supervision over the University of Florida, its policies and affairs, is vested in the Board of Regents. The Board of Regents is a body composed of nine citizens from different regions of the state who are appointed for nine-year terms by the Governor of Florida. All University affairs are administered by the President with the advice and assistance of the Administrative Council. This Council has the authority to determine all activities, including exclusion and removal of personnel and property from the area.

All land within the boundary lines of the campus and the exclusion area of the reactor building, as described in section 2.1.1.2 is owned and controlled by the Administrative Council of the University of Florida. The President and/or the Council of the University of Florida have the authority to determine all activities, including exclusion and removal of personnel and property from any part of the campus including the exclusive mineral rights for the entire campus area.

2.1.2.2 Control of Activities Unrelated to Plant Operation. Since the exclusion area is identified with the reactor cell, no activities unrelated to reactor operation will be permitted within the cell.

2.1.2.3 Arrangements for Traffic Control. Since the campus is not traversed by any major highway, traffic control arrangements will be limited to campus routes only. All ingress and egress roads to the campus (Figure 2-4) will be controlled by campus officials. In the event of difficulties arising from or developed by the reactor, the radiation warning system will sound the evacuation siren for the reactor building. The staff, faculty and students in the building are advised to evacuate the building upon hearing the siren. It is estimated that all uninjured persons can be evacuated from the reactor building in less than two (2) minutes. Evacuation routes lead directly away from the reactor building toward the nearest roads. Evacuation drills for facility personnel shall be conducted quarterly, at intervals not to exceed four months, to assure that facility personnel are familiar with the emergency plan.

2.1.2.4 Abandonment or Relocation of Roads. Since the reactor cell, which encompasses the reactor room and the control room, is defined as the exclusion area, there is no need to consider abandonment or relocation of public roads transversing the exclusion area.

2.1.3 Population Distribution

Population data is based on 1970 census data updated with more recent estimates as available.(8)

2.1.3.1 Population Within 10 Miles. The only significant large permanent population grouping within 10 miles of the reactor site is represented by the city of Gainesville itself (See Figure 2-5). The total city population is about 83,000 and as shown in Figure 2-2;most of the population is to the north and east of the reactor site.

Figure 2-6 illustrates the population density per square mile of the various entities in the State of Florida. As noted, Alachua County has a density of 50-249 persons per square mile. Figure 2-7A illustrates the percentage population change within the years 1960 through 1970 where Alachua County is found in the 40 to 100% category. Figure 2-7B illustrates the projected population for 1978 with Alachua County falling in the 100,000 - 500,000 range. Current estimates in early 1980 are that Alachua County contains about 145,000 residents.

2.1.3.2 Population Between 10 and 50 Miles. The major population centers between 10-50 miles from the reactor site are illustrated in Figure 2-5 where one can find sparsely populated areas with small population concentrations in the cities of High Springs (2871), Alachua (3,015), and Newberry (1,580) found within the 10-20 mile range from the reactor site. These and other less populated urban areas are found in Figure 2-8. Further detailed population information for this research reactor is not considered necessary due to the low power operation, low radioactive inventory and low potential for accidents as compared to a typical power plant.

2.1.3.3 Transient Population. Population variations related to the City of Gainesville are due mainly to the presence of the University of Florida and Santa Fe Community College, both having a great impact on the population composition of the greater Gainesville area.

The University of Florida population is mostly transient in its occupation of the campus buildings denoted in Figure 2-4. Most of the approximately 42,000 students, faculty and staff populate the campus in varying numbers primarily Monday through Friday during the hours from 7:30 a.m., to about 10:00 p.m. As noted previously, this number is a maximum in the fall and diminishes significantly due to reduced enrollment as the academic year progresses. About 6200 persons occupy the campus dormitories while another 1400 occupy the married housing areas on the periphery of the campus. The rest including about 11,600 faculty and staff make up the transient campus population.

The Santa Fe Community College population is completely transient. The Fall, 1979 semester enrollment was 7063 students while the current enrollment is 7216 students. Because of its location about 6 miles northwest of the UF campus, no further consideration is given to the Santa Fe Community College population.

2.1.3.4 Low Population Zone. The low population zone, as defined by 10 CFR Part 100.3(b), includes the University of Florida campus which constitutes a radial distance of approximately 3500 feet from the reactor site. The only significant permanent population concentrations in the low population zone are the dormitory facilities located on the University campus (See Figure 2-4). The closest residence hall is East Hall (#592), shown in Figure 2-3 which is approximately 750 ft. due west of the reactor building. East Hall is part of a series of adjacent buildings referred to as the Tolbert area housing approximately 950 students. The reactor is located about 600 ft. north of the Reitz Union, 100 ft. west of the Journalism Building and 250 ft. due east of the Materials Building. The

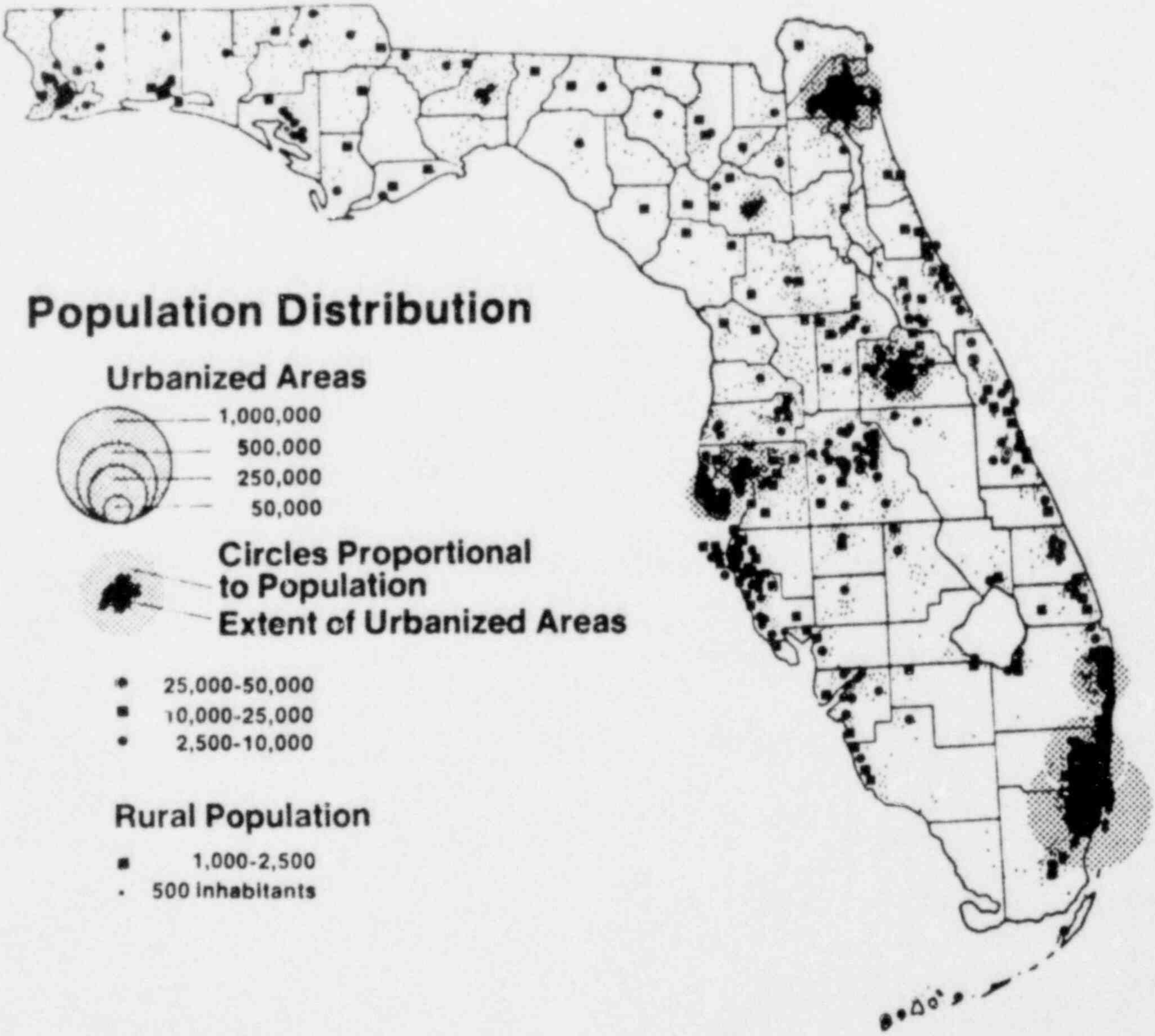


Figure 2-5. Florida 1970 Population Distribution.

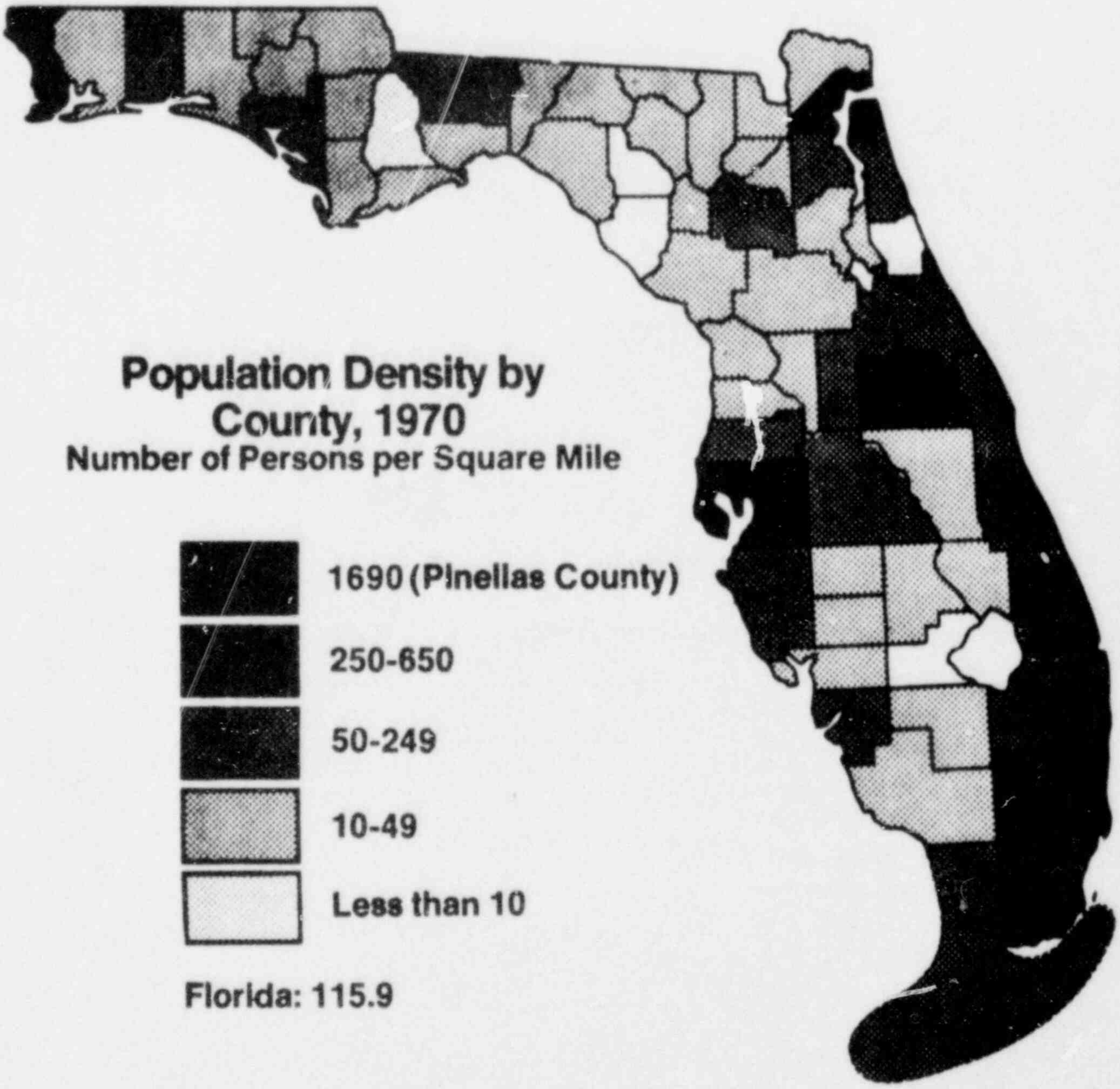


Figure 2-6. Florida Population Density by County for 1970.

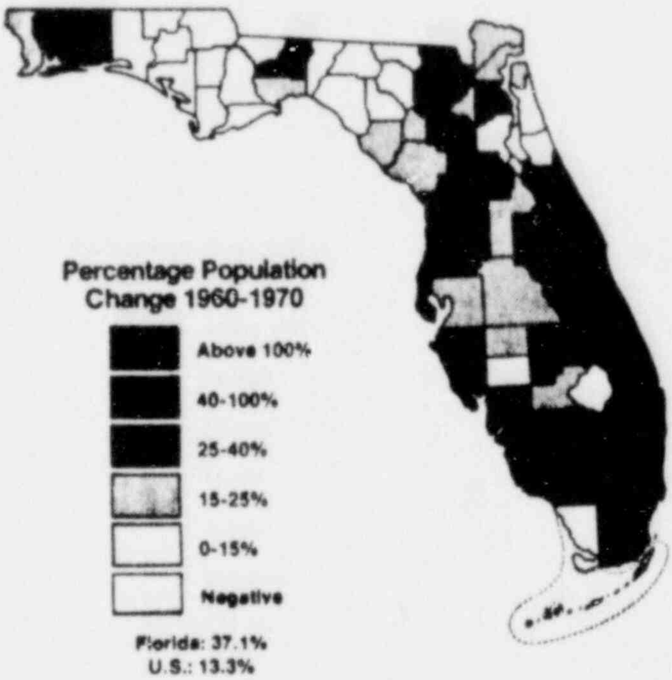


Figure 2-7A. Florida Percentage Population Change by County from 1960 to 1970.

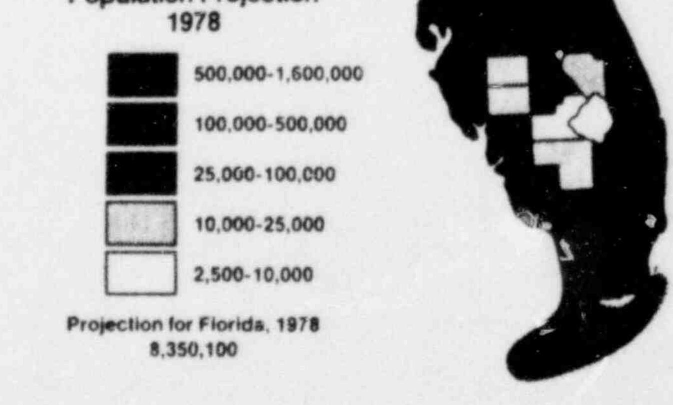


Figure 2-7B. Florida Population Projection by County for 1978.

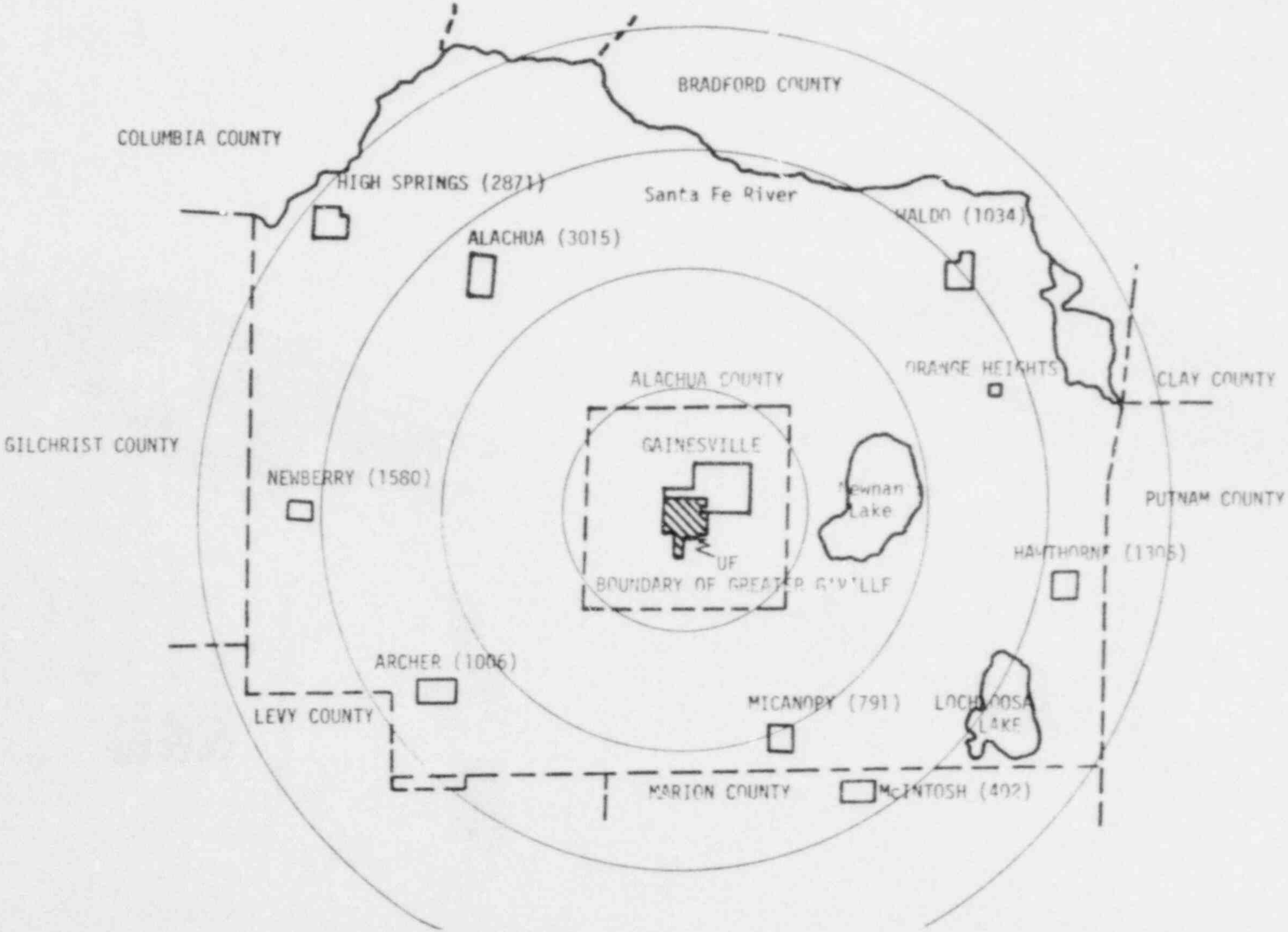


Figure 2-8. Population Concentrations and Locations in Alachua County with Concentric Circles at Five-Mile Intervals.

J. Hillis Miller Health Center is found approximately 3000 ft. southeast of the UFTR. Most of the fraternity houses and other residence halls are found within the 3000 to 4000 ft. range from the UFTR facility (5). The number of students housed within the campus residence areas, excluding the fraternities, is approximately 6,200 as of the fall quarter of 1979.

The transient population concentration within the low population zone is due to the staff, faculty and students who do not reside on campus. As mentioned in Section 2.1.3.3, this number is approximately 36,000.

This low population zone has been selected on the basis of its small easily evacuated, residential population. All of the people within the zone can be notified and evacuated in the event that a significant release of radioactive material occurs at the reactor site.

The dose received by an individual located on the outer boundary of this low population zone for the duration of the postulated fission product release is expected to be well below the preset limits of 25 rem whole body and 300 rem thyroid exposure as specified in 10 CFR 100.11(a)(2).

2.1.3.5 Population Center. The nearest population center as defined by 10 CFR 100.3(a) is the city of Gainesville. It should be noted that the boundary of the densely populated portion of Gainesville is located within approximately 5 miles to the north and northeast of the UFTR campus as shown in Figure 2-3. This distance will exceed the required one and one-third times the distance to the outer boundary of the low population zone as required by 10 CFR 100.11(a)(3).

2.1.3.6 Population Density Around the UFTR Site. Since the UFTR is a small, self-protected reactor presently licensed to operate at 100 Kw (thermal), the usual detailed information on population density out to a 30-mile distance from the reactor is not considered to be necessary. Except for the city of Gainesville, High Springs and Alachua, the rest of Alachua County is found to have a relatively low 50-249 persons per square mile (See Figure 2-6). Figure 2-8 shows the population of various towns around the reactor site, broken down into 5 mile concentric circles.

As indicated in Section 2.1.3, the specific population around the UFTR used for dose assessment calculations was obtained from the document "Characteristics of Housing Units and Population" by Blocks which consists essentially of population data from the 1970 census.(8) This population information is used in siting calculations for this SAR while the other information provided is of a more general and supportive nature. The urban area of Gainesville extends further than 5 miles from the UFTR, but the population was conservatively assumed to be concentrated within a 5 mile radius around the UFTR. Table 2-1 and Figure 2-9 show the population distribution for each sector of the compass for circles with radii 1 and 5 miles. The most significant changes to the Gainesville area population

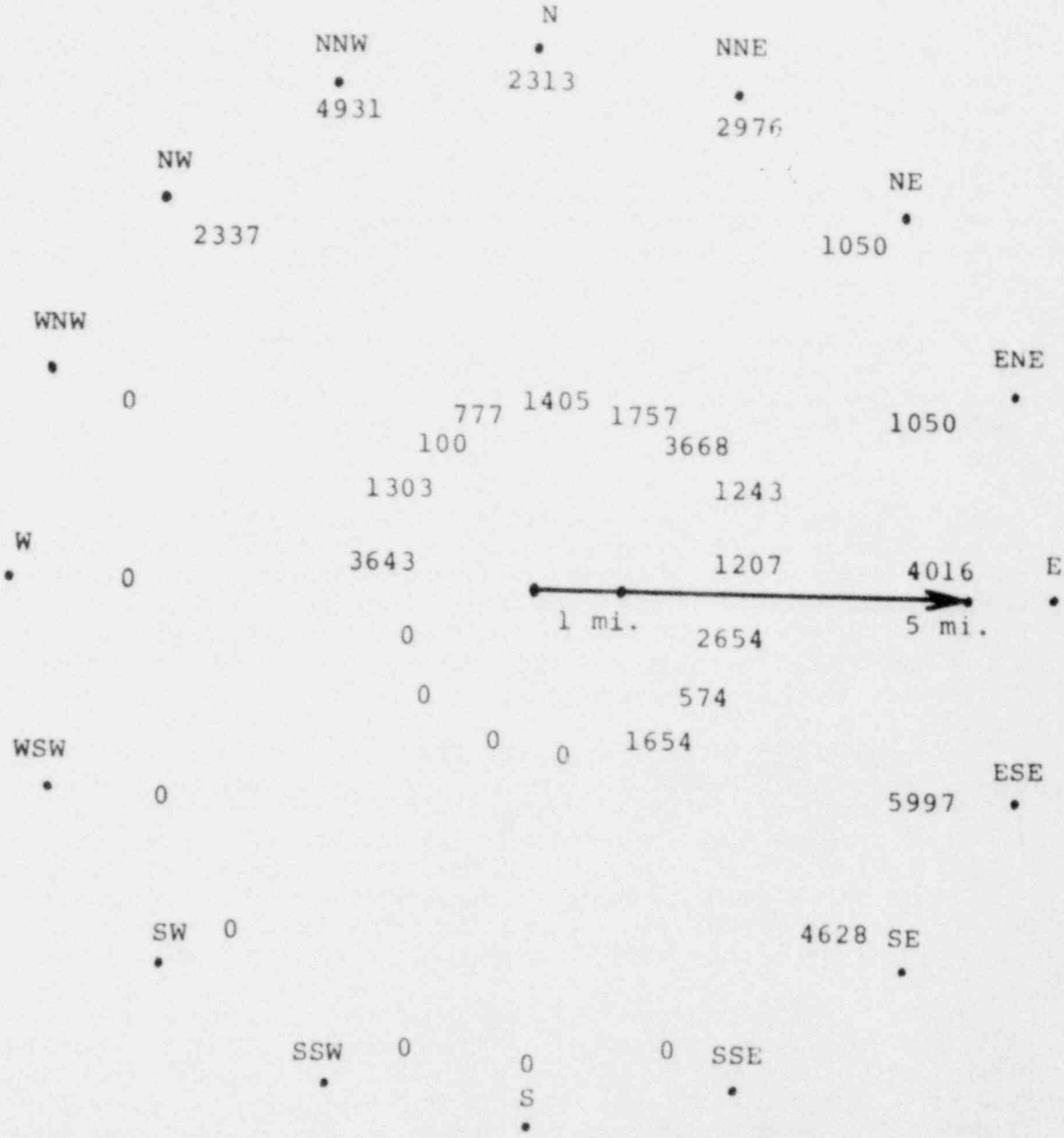


Figure 2-9. Population Distribution Around the UFTR By Sector Based Upon 1970 Census Data.

after 1970 have occurred in the "suburbs", outside the 5 mile area and through annexation, although there has been some buildup of population in the S, SSW, SW and WSW sectors primarily beyond the 1 mile radius which will be reported in the 1980 census not yet available. Since the S,SSW, etc. sectors are newly built up, they are expected to be relatively unimportant in the dose assessment analysis. The 1 and 5 mile radius circles are reported as the basis for establishing the so-called urban boundary addressed in Chapter 15 of this SAR analyzing hypothetical radiation doses following the design basis accident.

Table 2-1

Population Distribution Around the UFTR

Sector	Population Within 0-1 miles	Population Within 1-5 miles
N	1405	2313
NNE	1757	2978
NE	3668	1050
ENE	1243	1050
E	1207	4016
ESE	2654	5997
SE	574	4628
SSE	1654	*
S	*	*
SSW	*	*
SW	*	*
WSW	*	*
W	3643	*
WNW	1303	*
NW	100	2337
NNW	777	4931

2.2 Nearby Industrial, Transportation, and Military Facilities

A study of the area activities has shown that there are no significant industrial activities in the immediate area that could lead to potential accidents having an effect on the UFTR Reactor Building and environs.

2.2.1 Locations and Routes

Gainesville is primarily an education-related, small-business-oriented city. Large-scale industries are not present to any significant extent; the areas surrounding the UFTR site and University of Florida campus are representative of most of Gainesville, consisting primarily of residential areas, apartment complexes and small businesses such as restaurants, stores, etc. A study of area activities shows that there are no significant industrial activities in this immediate area that could lead to potential accidents having an effect on the UFTR Reactor Building.

Transportation routes located close to campus include State Road 26 known as University Avenue which is located approximately 2300 ft. north of the reactor site, U.S. Highway 441 known as 13th Street located about 3800 ft. east of the reactor site, State Road 121 located about 7000 ft. west of the reactor site. The location of all of the above are shown in Figures 2-3 and 2-4. Interstate 75 is located about 3 1/2 miles southwest of the reactor site at its closest approach.

Since the reactor building is located between the Nuclear Sciences Center on the south side and the Reed Laboratory buildings on the north, any explosion of transported materials would first have to exert its effect on both of these buildings. Although not immediately adjacent, the same protection is afforded on the east side by the Journalism Building and on the west side by the unoccupied Chiller Unit Facility. The location of the UFTR building in relationship with all surrounding buildings and the campus in general, provides for shielding and a protective effect from the forces of explosion on all sides.

The Gainesville Regional Airport is the only airport in the vicinity. Although the runway system is essentially unchanged, the airport terminal is a completely new facility to the south of the main runway opposite the old terminal on the north side and about a half mile away. The airport is located approximately five (5) miles northeast of the University of Florida campus as shown in Figure 2-2.

2.2.2 Descriptions

Since there are nearly no industrial or military facilities which are expected to impact upon the safe operation of the UFTR facility, the descriptions in this section are limited to major transportation routes through and around Gainesville and to the only airport in the area, the Gainesville Regional Airport.

2.2.2.1 Description of Transportation Routes. State Roads 26, 121, and 24, U.S. Highway 441 and Interstate 75 are all well-traveled, major transportation routes through and/or around Gainesville. The primary usage of State Roads 26, 121, 24 and U.S. Highway 441 are for commuter travel to the University of Florida and to the center of the city. Interstate 75 is primarily used for commuter travel from surrounding cities and for tourist travel to South and Central Florida. Other uses for all of the above roads include shipment of goods and services but shipment of dangerous, toxic or explosive substances would be minimal, particularly for those roads nearest the UFTR site, i.e., State Roads 26,121, and 24 and U.S. Highway 441.

2.2.2.2. Description of the Gainesville Airport. The Gainesville Regional Airport is located on the northeast edge of Gainesville, Florida, four (4) miles northeast of the center of the city. Primary access from the city center is via excellent four lane routes, East University Avenue and Waldo Road (State Road 24), as seen in Figure 2-2. The former Army Air Corps Bases which is now Gainesville Regional Airport was deeded to the city of Gainesville, the present owner, in 1948.

The Gainesville Regional Airport has a total of 10,650 ft. of runway (compass headings 240° - 280°), as seen in Figure 2-10 The airport provides both air carrier and general aviation facilities for the Gainesville area. Certified air carrier service is provided by Eastern Airlines. Scheduled Interstate air carrier service is provided to Gainesville by Air Florida. In Table 2-2, the Air Traffic Volume Report for the Gainesville Regional Airport for the year 1976 shows the number of operations during that year and compares it to the previous year's figures. Table 2-3 represents the same information for the months of January through June of 1977, also comparing these semi-annual figures with those of 1976. Tables 2-4, 2-5 and 2-6 similarly contain the Air Traffic Volume Reports for the single month of January in 1978, 1979 and 1980 respectively. These reports indicate a steady, relatively fast increase in scheduled air carrier activities and steady decreases in chartered activities. In general, the airport is becoming busier with larger volumes of traffic which is the justification for the new John R. Alison Air Terminal Building opened in January, 1979. Table 2-7 includes the demand allocation for the Gainesville Airport including the mid-1978 demand with projections into the years 1980, 1985 and 1990. It should be noted that these figures do not include the additional operations that will be brought about by the addition of Air Florida Airline passenger services to Gainesville. In spite of this fact, these projections are still deemed accurate since the figures were originally considered an overestimate. Of the toal number of landings, it can be assumed conservatively that approximately 25 percent will cross the University of Florida campus.

Development of the Gainesville Regional Airport, as provided in the current airport map (Figure 2-10)includes the extension of runway 10/28 to 8,500 ft., the construction of a utility runway parallel to runway 10/28, and the expansion of the general aviation terminal facilities. Completion of the above has been delayed several times but is expected by the summer of 1981 (9).

Accidents recorded for the period of January through September 1977, include three (3) forced landings, while during the year of 1976, five (5) forced landings were recorded; an average of 3 to 4 (minor) accident occurrences per year can be assumed. There have been no fatalities reported. An examination of this accident information indicates that there is a very small probability of an aircraft accident such as a crash, affecting the reactor building of the UFTR facility which represents such a small fraction of the possible crash area around the airport and is about five miles removed from the airport.

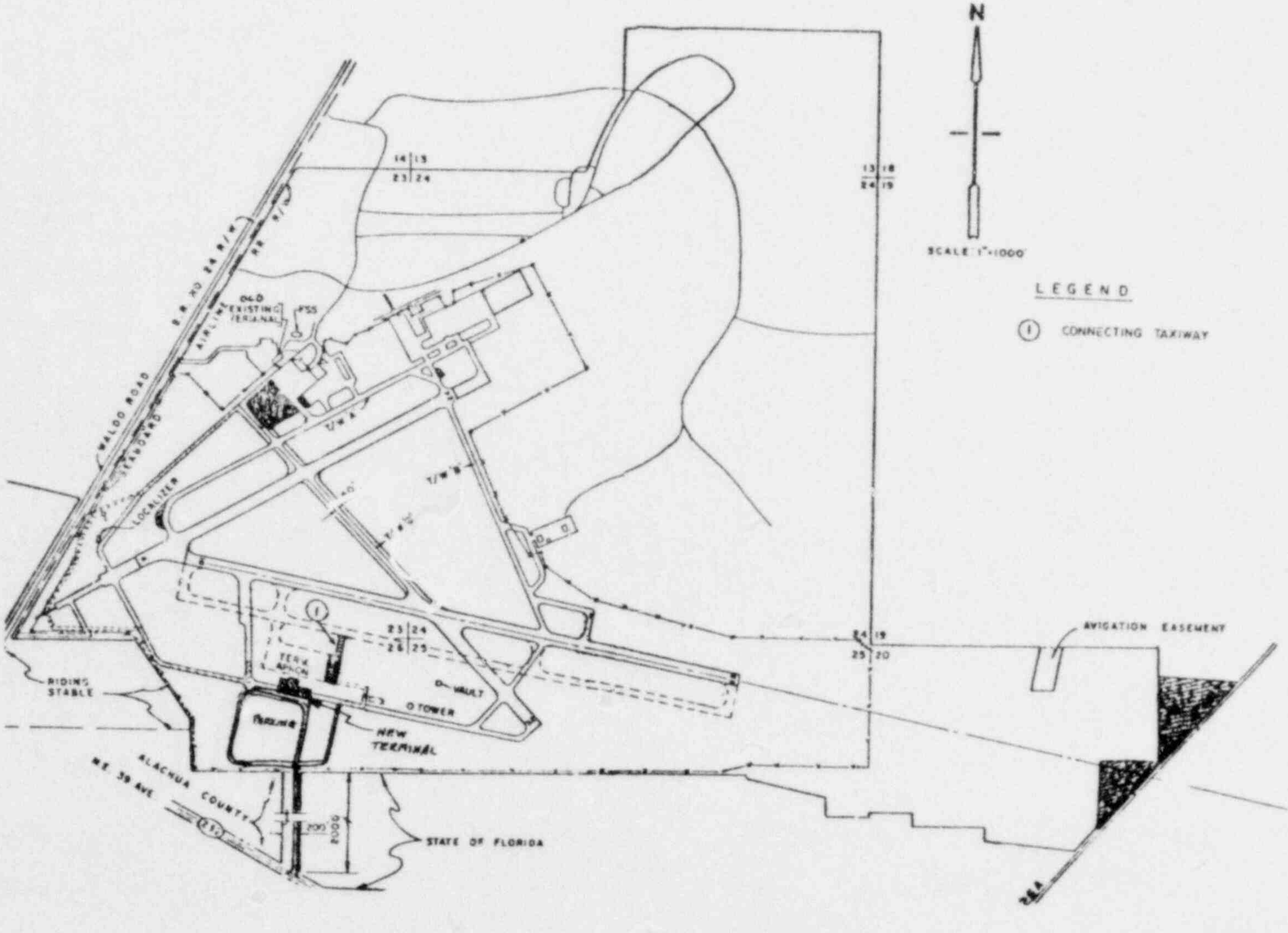


Figure 2-10. Location and Orientation of Gainesville Municipal Airport Runways.

Table 2-2

Gainesville Regional Airport
3901 N.E. 46th Drive, Gainesville, Florida 32601

AIR TRAFFIC VOLUME REPORT

DECEMBER, 1976

	1976	1975	PERCENTAGE 1976 over 1975 INCREASE OR (DECREASE)	1976	1975	PERCENTAGE 1976 over 1975 INCREASE OR (DECREASE)
PASSENGERS (Number)						
Scheduled Air Carrier						
Deplaned	10,321	9,399	9.8	111,824	94,879	17.8
Enplaned	11,108	9,672	14.8	111,495	95,550	16.7
Totals	21,429	19,071	12.4	223,319	190,429	17.3
Commuter/Air Taxi						
Deplaned	58	265	(78.1)	2,050	3,752	(45.4)
Enplaned	64	306	(79.1)	2,399	3,798	(36.8)
Totals	122	571	(78.6)	4,449	7,550	(41.1)
Non-Scheduled (Charter)						
Deplaned	325	124	162.1	5,814	4,347	33.7
Enplaned	525	124	323.4	6,091	4,672	30.4
Totals	850	248	242.7	11,905	9,019	32.0
Total (All Types)						
Deplaned	10,704	9,788	9.4	119,698	102,978	16.2
Enplaned	11,697	10,102	15.8	120,561	104,020	15.9
Totals	22,401	19,890	12.6	240,259	206,998	16.1
TOWER OPERATIONS (Numbers)						
Air Carrier	314	326	(3.7)	3,870	3,656	5.8
Commuter/Taxi	17	240	(93.0)	1,382	3,824	(63.8)
General Aviation	7,584	6,232	21.7	90,117	82,146	9.7
Military	43	245	(82.4)	2,269	2,657	(14.6)
Totals	7,958	7,043	13.0	97,638	92,283	5.8

Table 2-3

Gainesville Regional Airport
3901 N.E. 46th Drive, Gainesville, Florida 32601

AIR TRAFFIC VOLUME REPORT

JUNE, 1977

	1977	1976	PERCENTAGE 1977 over 1976 INCREASE OR (DECREASE)	1977	1976	PERCENTAGE 1977 over 197[illegible] INCREASE OR (DECREASE)
PASSENGERS (Number)						
Scheduled Air Carrier						
Deplaned	11,147	9,428	18.2	60,930	55,048	10.7
Enplaned	11,067	9,778	13.2	60,125	55,098	9.1
Totals	22,214	19,206	15.7	121,055	110,146	9.9
Commuter/Air Taxi						
Deplaned	136	242	(43.8)	789	1,585	(50.2)
Enplaned	183	252	(27.4)	897	1,894	(52.6)
Totals	319	494	(35.4)	1,686	3,479	(51.5)
Non-Scheduled (Charter)						
Deplaned	88	233	(62.2)	2,094	2,625	(20.2)
Enplaned	97	233	(53.4)	1,879	2,675	(29.8)
Totals	185	466	(60.3)	3,973	5,300	(25.0)
Total (All Types)						
Deplaned	11,271	9,903	14.8	63,813	59,258	7.7
Enplaned	11,347	10,263	10.6	62,901	59,667	5.4
Totals	22,718	20,166	12.7	126,714	118,925	6.5
TOWER OPERATIONS (Numbers)						
Air Carrier	297	311	(4.5)	1,851	1,942	(4.7)
Commuter/Taxi	188	165	13.8	680	1,173	(42.0)
General Aviation	9,088	5,915	53.6	52,842	40,167	31.6
Military	166	99	17.2	946	1,411	(33.0)
Totals	9,689	6,490	49.3	56,319	44,693	26.0

Table 2-4

GAINESVILLE REGIONAL AIRPORT
3901 N.E. 46th Drive, Gainesville, Florida 32601

AIR TRAFFIC VOLUME REPORT

JANUARY, 1978

	1978	1977	PERCENTAGE 1978 over 1977 INCREASE OR (DECREASE)
PASSENGERS (Number)			
Scheduled Air Carrier			
Deplaned	12,355	9,579	28.7
Enplaned	11,938	9,158	30.4
Totals	24,293	18,755	29.5
Commuter/Air Taxi			
Deplaned	198	107	85.0
Enplaned	186	100	86.0
Totals	384	207	85.5
Non-Scheduled (Charter)			
Deplaned	254	795	(68.1)
Enplaned	295	520	(43.3)
Totals	549	1,315	(58.3)
Total(All Types)			
Deplaned	12,807	10,499	22.0
Enplaned	12,419	9,778	27.0
Totals	25,226	20,277	24.4
CARGO (Pounds)			
Air Freight			
Deplaned	60,760	49,954	21.6
Enplaned	29,441	22,252	32.3
Totals	90,201	72,206	24.9
Mail			
Deplaned	000	000	000
Enplaned	000	000	000
Totals	000	000	000
Air Express			
Deplaned	000	000	000
Enplaned	000	000	000
Totals	000	000	000
Total Cargo			
Deplaned	60,760	49,954	21.6
Enplaned	29,441	22,252	32.3
Totals	90,201	72,206	24.9
TOWER OPERATIONS (Numbers)			
Air Carrier	409	322	27.0
Commuter/Air Taxi	186	15	1140.0
General Aviation	6,890	7,938	(13.2)
Military	37	111	(66.7)
Totals	7,552	8,386	(10.3)

Table 2-5

GAINESVILLE REGIONAL AIRPORT (New Terminal)
3901 N.E. 46th Avenue, Gainesville, Florida 32601

AIR TRAFFIC VOLUME REPORT

JANUARY, 1979

	1979	1978	PERCENTAGE 1979 over 1978 INCREASE OR (DECREASE)
PASSENGERS (Number)			
Scheduled Air Carrier			
Deplaned	14,724	12,355	19.2
Enplaned	14,302	11,938	19.8
Totals	29,026	24,293	19.5
Commuter/Air Taxi			
Deplaned	233	198	17.7
Enplaned	224	186	20.4
Totals	457	384	19.0
Non-Scheduled (Charter)			
Deplaned	50	254	(80.3)
Enplaned	70	295	(76.3)
Totals	120	549	(78.1)
Total (All Types)			
Deplaned	15,007	12,807	17.2
Enplaned	14,596	12,419	17.5
Totals	29,603	25,226	17.4
CARGO (Pounds)			
Air Freight			
Deplaned	44,440	60,760	(29.9)
Enplaned	30,263	29,441	2.8
Totals	74,703	90,201	(17.2)
Mail			
Deplaned	000	000	000
Enplaned	000	000	000
Totals	000	000	000
Air Express			
Deplaned	000	000	000
Enplaned	000	000	000
Totals	000	000	000
Total Cargo			
Deplaned	44,440	60,760	(26.9)
Enplaned	30,263	29,441	2.8
Totals	74.703	90,201	(17.2)
Tower Operations (Numbers)			
Air Carrier	454	409	11.0
Commuter/Air Taxi	191	186	2.7
General Aviation	8,906	6,890	29.3
Military	70	37	89.2
Totals	9,621	7,522	27.9

Table 2-6

GAINESVILLE REGIONAL AIRPORT
3400 N.E. 39th Avenue, Gainesville, Florida

AIR TRAFFIC VOLUME REPORT

JANUARY, 1980

	1980	1979	PERCENTAGE 1980 over 1979 INCREASE OR (DECREASE)
PASSENGERS (Number)			
Scheduled Air Carrier			
Deplaned	16,080	14,724	9.2
Enplaned	14,893	14,302	4.1
Totals	30,973	29,026	6.7
Commuter/Air Taxi			
Deplaned	-	233	(100.0)
Enplaned	-	224	(100.0)
Totals	-	457	(100.0)
Non-Scheduled (Charter)			
Deplaned	129	50	158.0
Enplaned	128	70	82.9
Totals	257	120	114.2
Total (All Types)			
Deplaned	16,106	15,007	7.3
Enplaned	14,927	14,596	2.3
Totals	31,033	29,603	4.8
CARGO (Pounds)			
Air Freight			
Deplaned	55,373	44,440	24.6
Enplaned	40,943	30,263	35.3
Totals	96,316	74,703	28.9
Mail			
Deplaned	000	000	000
Enplaned	000	000	000
Totals	000	000	000
Air Express			
Deplaned	000	000	000
Enplaned	000	000	000
Totals	000	000	000
Total Cargo			
Deplaned	55,373	44,440	24.6
Enplaned	40,943	30,263	35.3
Totals	96,316	74,703	28.3
TOWER OPERATIONS (Numbers)			
Air Carrier	538	454	18.5
Commuter/Taxi	20	191	(89.5)
General Aviation	7,388	8,906	(17.0)
Military	72	70	2.9
Totals	8,018	9,621	(16.7)

Table 2-7

GAINESVILLE REGIONAL AIRPORT

DEMAND ALLOCATION

COMMERCIAL AND MILITARY

Operations	Base Year 1970	Existing	1980	1985	1990
Cert. Air Carrier	NA	2,782	6,200	8,800	12,700
Intra. Air Carrier*	NA	-	(11,400)	(12,800)	(14,300)
Busy Hour	NA	2	7	8	9
Ann. Instrument App.	1,502	219	634	900	1,299
Military	-	3,643	3,700	3,700	3,700
Passengers					
Enplaned	7,009**	58,757	200,500	305,200	464,600
Typical Peak Hour	NA	153	361	458	511
Based Aircraft		GENERAL AVIATION			
Single-Eng.	49	86	87	119	152
Multi-Eng. 12.5	11	18	20	26	36
Multi-Eng. 12.5	-	5	2	-	-
Turboprop 12.5	-	-	3	3	5
Turboprop 12.5	-	-	-	2	3
Turbojet 12.5	-	-	2	4	8
Rotor	-	-	4	6	9
TOTAL	60	109	118	160	213
Operations					
Single-Eng.	76,500	91,600	143,700	200,400	271,500
Multi-Eng.	5,500	9,500	12,000	16,900	23,800
Multi-Eng. 12.5	-	1,500	600	-	-
Turboprop 12.5	-	-	5,600	5,900	8,700
Turboprop 12.5	-	-	-	1,900	2,900
Turbojet	-	2,200	4,200	5,500	10,000
Rotor	-	-	5,200	9,100	13,700
LOCAL	37,000	35,500	59,400	79,800	109,900
ITINERANT	45,000	68,800	111,900	161,900	220,700
TOTAL	82,000	104,300	171,300	241,700	330,600
Busy Hour VFR	101	102	130	156	196
Busy Hour IFR	24	13	28	40	54
Ann. Instrument App.	856	977	2,238	3,238	4,414
Passengers					
Busy Hour Pilots and Passengers	-	-	164	198	249

*Included in General Aviation Operations Forecast

**Fy 1970.

2.2.2.3 Projections of Industrial Growth. As stated in the Chamber of Commerce Population Estimation Report (10), the Gainesville metropolitan area is a center for health and educational services for both the region and the state. The presence of the University of Florida and the Santa Fe Community College has had a great impact on both the population composition and the economic structure of the area.

Residents of the Gainesville area depend heavily on government institutions, particularly public education institutions, for employment. Unlike the southern Florida metropolitan areas, Gainesville does not depend heavily on tourist trade, the citrus industry or the in-migration of retirees for its economic well-being. The major role accorded to government employment and the relative stability of its economy make the Gainesville area similar to other northern Florida medium-sized metropolitan areas such as Tallahassee and Pensacola. With the attraction of service and recreation-related companies such as Nationwide Insurance and Bear Archery, the Gainesville area is spreading its economic base but not affecting the type of non-polluting, service and light industry base now in existence.

University of Florida economists had projected an annual increase of 4.5 percent in non-farm jobs during 1977 and 1978. Significant employment growth has been realized in retail trade, in the services industries and in contract construction by the end of 1978. The current building slowdown is affecting Gainesville, but less so than most parts of the country. In the last two years Bear Archery and Nationwide Insurance have both located in Gainesville and contributed to its population growth with approximately 800 new jobs. These types of activities do not impact significantly upon the UFTR site.

Although there is significant growth projected for the Gainesville area, for non-farm jobs in retail trade and services industries, there are no new significant types of activities or industrial development currently projected in the UFTR site vicinity in the near future. Consequently, this growth pattern has no direct impact on the UFTR site suitability.

2.2.3 Evaluation of Potential Accidents

2.2.3.1 Determination of Design Basis Events. The effects of potential accidents in the vicinity of the reactor site from present and projected industrial installations and operations are concluded to be insignificant when compared to the accident potential presented by tornadoes in the North-Central Florida region. This same conclusion applies for effects from potential transportation accidents which are also concluded to have minor effects.

Based on the low probability of aircraft accidents, the relatively small areas of aircraft impact, the protected location of the UFTR building in reference to other surrounding buildings, and the relatively small size of most of the aircraft involved, it is concluded that the probability for tornadoes to affect the UFTR site as well as their potential impact, is much greater than the probability of an aircraft crash affecting the site. (See tornado data in Section 2.3.1.2.2). Therefore, potential tornado damage is considered the most probable and most severe, externally-initiated accident possibility. All other effects from potential accidents in the vicinity .. the site due to industrial or transportation operations are considered negligible compared to the potential effects of tornadoes.

2.2.3.2 Effects of Design Basis Events. As the external design basis accident, tornadoes will have no effect on the safety-related components of the UFTR. Since the reactor building is designed as a vaulted structure (See Chapter 3), tornadoes are not expected to affect the UFTR training reactor itself.

2.3 Meteorology

2.3.1 Regional Climatology

2.3.1.1 General Climate. Quoted from Reference 2, the following information is based on local climatological summaries for the Gainesville area prepared by the U.S. Weather Bureau. The proximity of the extensive land mass to the north and northwest gives Gainesville a continental type climate in winter, but the nearness of the ocean area and the direction of prevailing winds cause marine climatic characteristics to prevail in summer. Maximum temperatures in the nineties are common in summer but readings as high as 100°F have been recorded in only eleven of the last thirty-two years prior to 1978. Frequent afternoon thunderstorms and associated showers provide relief from heat in summer. February 14th is the average date for the last occurrence of freezing temperatures in the spring and the average date for the first occurrence of freezing temperatures in the fall is December 6th. Precipitaion varies greatly from year to year for any month but on the basis of mean monthly totals, there is a rainy season of four months, June through September. This four-month period brings about 52 percent of the annual total precipitation, nearly all of which is in the form of rain. Hail falls occasionally but usually covers very small localized areas. The only measurable accumulations of snow recorded at Gainesville was 1.0 inch in February, 1899. On January 18, 1977 there was a trace of snow recorded in Gainesville. There was a trace of snow or sleet in December 1917, February 1951, and January 1958. The major portion of the rain comes from showers that are of relatively short duration and frequently associated with thunderstorms. The greatest precipitation total for any month appearing in the records for this station is 15.78 inches in October, 1941. The longest drought without measurable rainfall was 39 days, October 18 through November 25, 1903. It is not expected that any of these weather extremes would affect the safe operation of the UFTR facilities.

2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases.

2.3.1.2.1 Tropical Storms (Hurricanes). As stated in Reference (11), since 1891 when more complete weather recordkeeping was started, through 1972, a total of 58 tropical storms or hurricane centers have passed within approximately 75 miles of the University of Florida site, only one additional hurricane has come near the UFTR site along the east coast of Florida but much more than 75 miles away at its nearest center. After 1885, weather records differentiated between tropical storms (winds less than 73 mph) and hurricanes (winds more than 73 mph). For 1886 through 1972, there have been 46 passages of tropical storms. Of these a maximum of 13 hurricanes were experienced within 100 miles from the site. The most destructive was probably the hurricane of October 19, 1944. However, relatively few storms have moved inland on Florida's west coast between Cedar Key (directly west from Gainesville) and Fort Myers in the past 88 years. The most recent hurricane to pass near the UFTR site had its center several hundred miles away along the east coast of Florida at its nearest point as it moved on a northerly course along the coast. Most tropical storms have a tendency to move on one of three general courses which prevents them from having a maximum impact on the UFTR area as they move northward. As shown in Figure 2-11A,the typical tropical storm takes one of three routes; either it (1) recurves north and northeast over the Florida coast, (2) moves northward paralleling the west coast, or (3) moves on a north-westerly course across the Gulf of Mexico. As illustrated on the frequency histogram in Figure 2-11B,the highest frequency of tropical storms in the Central Florida area has occurred in September, with October being the month of the second highest frequency. Nevertheless, tropical storms are not considered a great hazard at the University of Florida UFTR site for two reasons. First, the severity of the storm is reduced by the overland movement necessary for a storm from the Gulf of Mexico or the Atlantic Ocean to reach the Gainesville area. Second, tidal flooding is prevented by the inland location of the UFTR site and there are no bodies of water near the UFTR site.

Experience with the passage of past hurricanes indicates maximum gusts of approximately 60 miles per hour around the site. It should be noted that even thunderstorms with accompanying hail, excessive rain, and strong winds, occasionally develop gustiness of this severity.

2.3.1.2.2 Tornadoes. In the period of 1948 through 1958, more than 50% of the waterspouts reported throughout the coastal states of the United States were reported in Florida. Of the 1,264 reported occurrences in Florida from 1948 through 1972, 575 of these were observed on Florida's west coast. Waterspouts have occasionally caused considerable damage to shipping and have become destructive tornadoes as they crossed from water to land.(11)

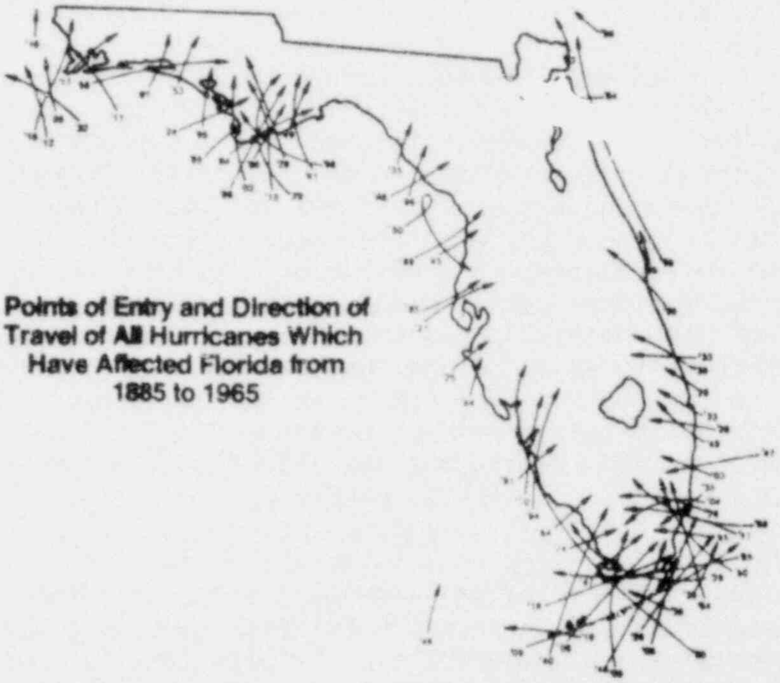


Figure 2-11A. Historical Hurricane Points of Entry for the State of Florida.

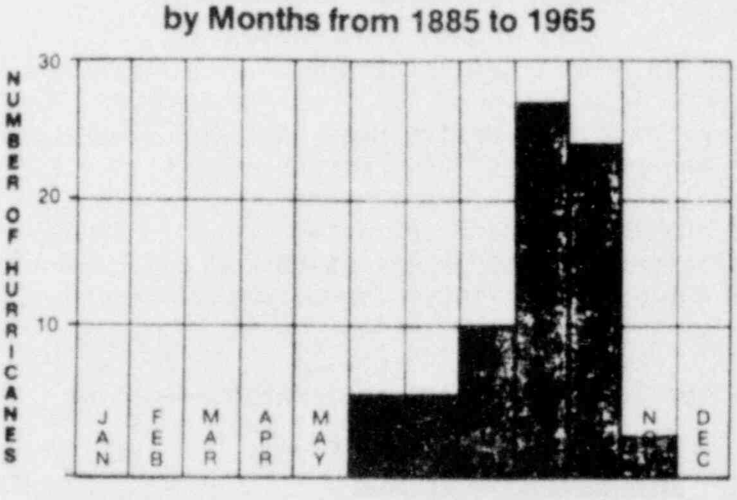


Figure 2-11B. Florida Hurricane Monthly Frequency Histogram.

Property damage due to tornadoes has been reported in Alachua County only six (6) times during the last 62 years. In February 1934, tornadoes caused damage in the eastern part of the county resulting in six persons injured and one death. In July 1946, a tornado hit Stengel airfield causing property damage estimated at $21,500. In June 1957, a small tornado destroyed tobacco barns and caused minor property damage in the vicinity of Newberry, approximately 15 miles from Gainesville. In July 15, 1967, a tornado touched in the North Main Street area of Gainesville injuring two persons and causing extensive damage valued at over $80,000. On May 4, 1978, the latest tornado struck a number of areas in Gainesville spread primarily along a corridor across northwest Gainesville causing considerable property damage through fallen trees and damaged roof structures and motor vehicles. In general, serious building damage was limited to the effects of fallen trees or damage done to mobile homes.

In the period of 1961 through 1972, a total of 776 tornadoes were reported in the State of Florida. Approximately 81 of these tornadoes were associated with the passage of tropical storms. As quoted from Reference 10, statistics compiled by THOM (12) indicate the highest frequency of tornado occurrence is along Florida's southeast coastline, and also south of Tampa. As illustrated in Figure 2-12A, the tornado frequency in Alachua County was between 5 and 9 for the typical period of 1959 through 1971. Figure 2-12B indicates that June is the month in which the highest numbers of tornadoes have occurred in the Florida area.

According to statistical methods proposed by THOM, the probability of a tornado striking a point within a given area may be estimated using the formula (12):

$$P = \frac{ZT}{A} \qquad (2\text{-}1)$$

where symbols are defined as follows:

- P = the mean probability per year of a tornado striking a point within area A
- Z = the geometric mean tornado path area, square miles
- T = the mean number of tornadoes per year in the area
- A = the area concerned, square miles.

The value of T (mean number of tornadoes per year) is taken as 7.0 in 12 years for Alachua County in which the UFTR site is located. Based on data reported by THOM for midwest tornadoes, an average tornado path area is about 2.82 square miles which is the applicable conservative value used for Z. The surface area of Alachua County is approximately 965 square miles which is the value used for A.

Weather bureau records indicate that the average path of the few

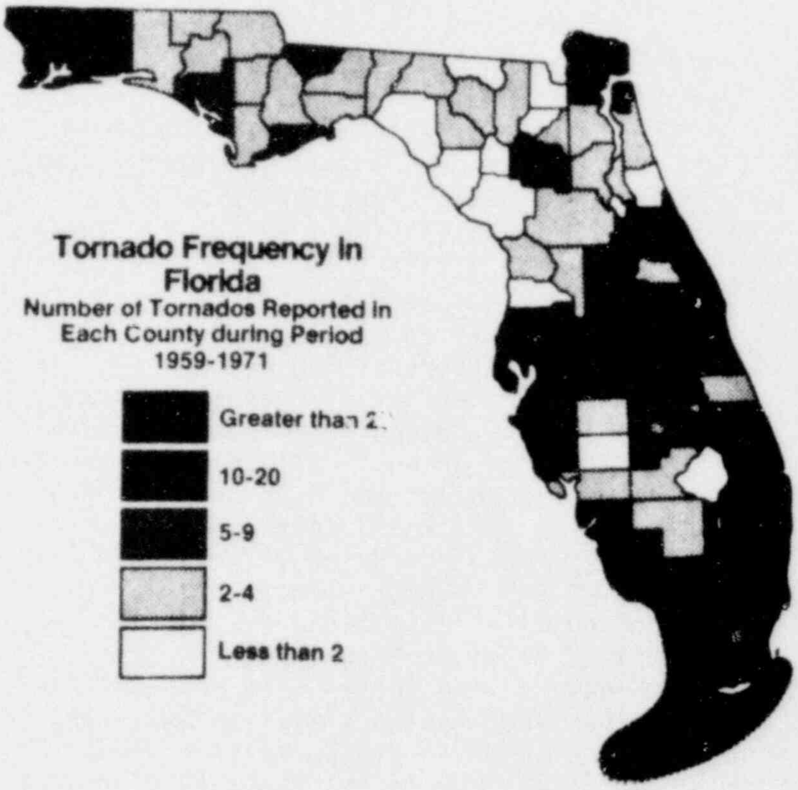


Figure 2-12A. Tornado Frequency by Florida County for Years 1959-1971.

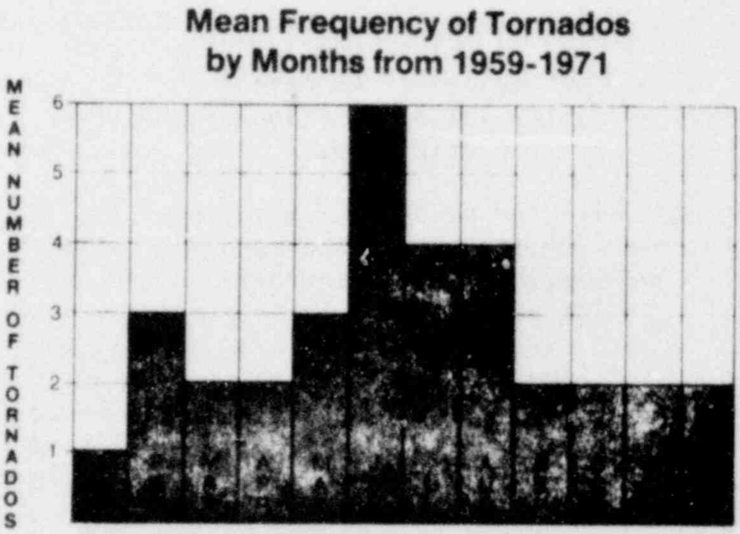


Figure 2-12B. Monthly Frequency of Tornados by Month for the Years 1959-1971.

tornadoes in Florida that actually reach the ground is about 125 yards wide and 4 miles long, (0.284 square miles) as compared to a nationwide average path area measuring 400 yards in width and 16 miles in length. In other words, Florida tornadoes typically affect about 7.8% of the area that a tornado affects on the national average

Using the value of A equivalent to the total land area of Alachua County (965 square miles) in which the UFTR site is located, a value of $P \approx 1.7 \times 10^{-3} \text{year}^{-1}$ is calculated as the mean probability per year of a tornado striking within the UFTR site. Of course this probability of such a tornado striking within the UFTR site (reactor building occupies less than an acre) is very conservative because the mean tornado path area in Florida is so much less than the national average used in the calculation.

The mean recurrence interval, $R = 1/P$, of a tornado striking a point anywhere in the 0.024 degree square in which the site is located is, therefore, about 600 years. However, in the period from 1916 through 1979, only six (6) property-damaging tornadoes have been reported in Alachua County, Florida where the site is located (equivalent to the probability of $P = 2.8 \times 10^{-4} \text{year}^{-1}$). Since the probability value P is greater than 10^{-7}, tornadoes will be considered for the design and operating basis of the UFTR as the most likely natural disaster to affect the UFTR site.

2.3.2 Local Meteorology

2.3.2.1 Normal and Extreme Values of Meteorological Parameters. As quoted from Reference 11, there are two major sets of meteorologically influential features which interact to determine the climate patterns of Alachua County and the UFTR site. The first set of influential features includes the critical surface features of the county as well as its location relative to other significant, climate-influencing geographical properties of the surrounding region. The critical surface features are depicted in Figure 2-13 which shows the generalized topographical map of the State of Florida, and in Figure 2-14 which shows the generalized topographical map of Alachua County. The second set of influential features consists of predominant patterns of zonal atmosphere behavior.

The features which are included in the first set are:

1) latitude,
2) proximity to the Gulf of Mexico and the Atlantic Ocean,
3) presence of inland lakes,
4) strength of surface which depends upon a variety of surface properties, and,
5) the rate of nocturnal cooling.

The second set of influential geographic features include the following

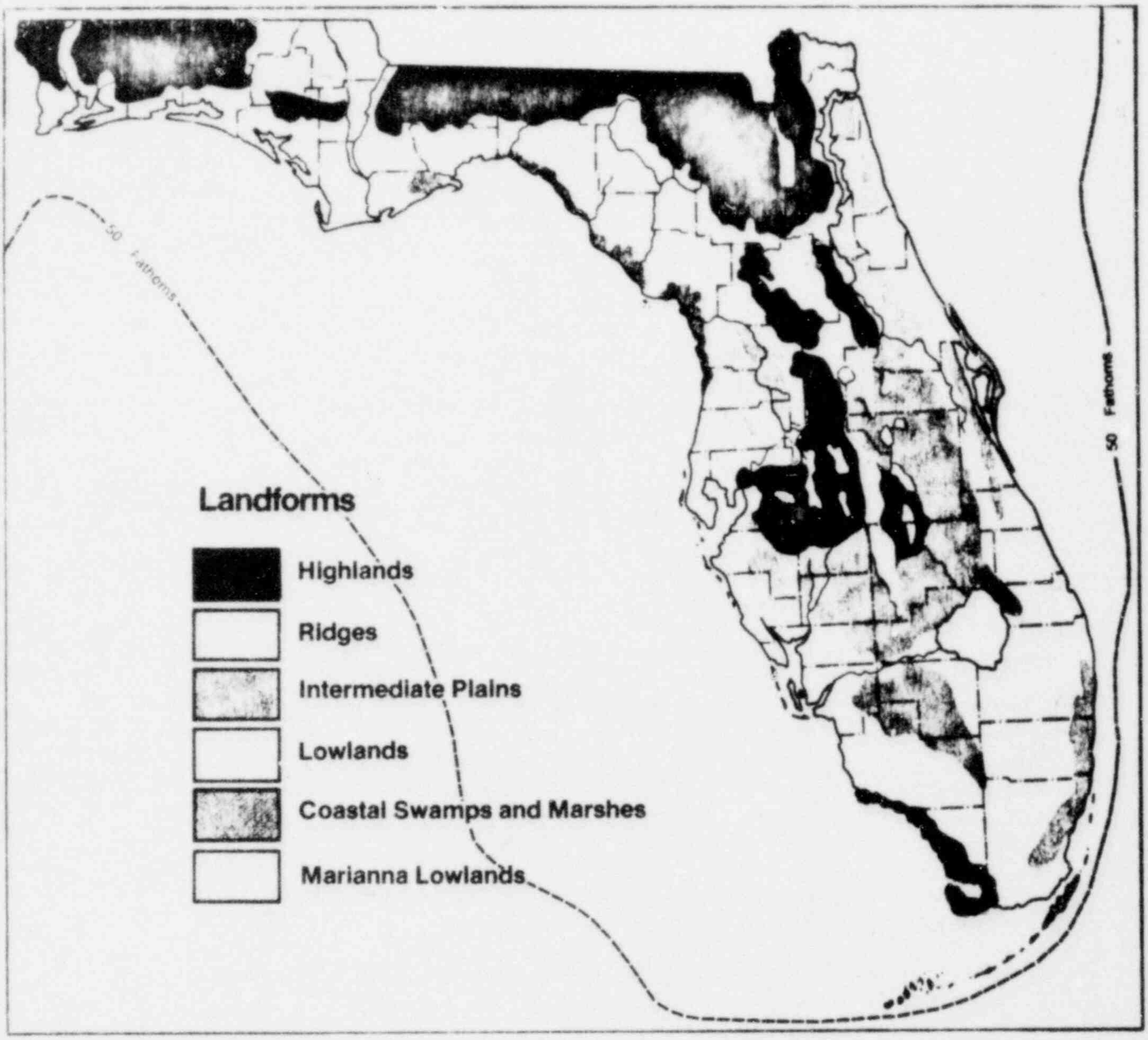


Figure 2-13. Topological Map of Florida.

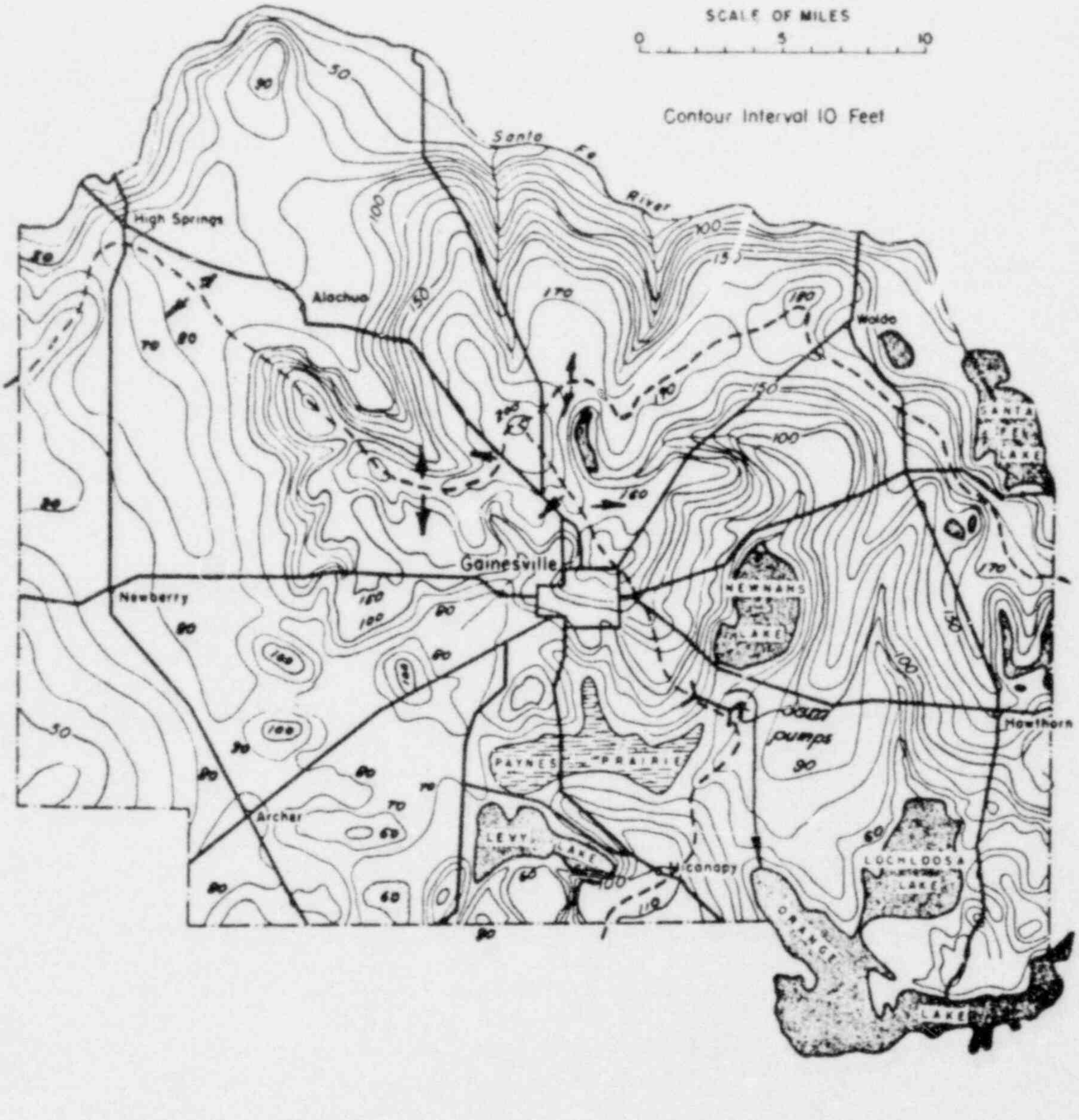


Figure 2-14. Generalized Topographical Map of Alachua County, Florida.

POOR ORIGINAL

properties:

1) sea breeze convergence,
2) frequency of frontal passage,
3) frequency and strength of hurricanes,
4) frequency and duration of anti-cyclonic subsidies condition,
5) frequency and intensity of occurrence of tradewind inversion, and
6) the position and strength of the north atlantic sub-tropical high.

The behavior of this last feature of atmospheric circulation controls the local behavior of most of the remainder of the state.

The average year in Alachua County may be divided into two seasons: The warm, rainy season and a cooler, dry season. The warm, rainy seasc- runs from about the middle of May to the end of September. The cooler, dry season dominates the remainder of the year. Most of the rain (about 60%) falls during the hot summer,occurring as afternoon thunderstorms generated by strong surface heating, and fed by double sea breeze convergence. When high cloud cover inhibits convective development in the afternoon, permitting only formation of small, cumulous clouds, rain may occur at night as a result of instability generated by nocturnal radiative cooling from the top of the small clouds. Precipitation during the summer has a very patchy horizontal distribution for any particular day.

Frontal passage during winter months is the most severe variable rain-producing mechanism for the county. Frontal or low occurrences within Florida averaged 38 for winter, 29 for spring, 19 for summer, and 41 for fall, for the years 1965 through 1967. Shaw's winter and fall are included in the cooler, dry winter seasons defined above. During the winter months, the differential, seasonal cooling of land and sea, the occasional presence of the strengthening high pressure cells, and the formation of low level inversions by the high rate of nocturnal cooling act to maintain a high degree of atmospheric stability. A high percentage occurrence of the tradewinds inversion during these winter months (70% in February) also contributes to this stability. Under these conditions, convective activity is suppressed and the possibility for vertical mixing and ventilation is limited. Frontal passages act to disrupt this stability and generate convective activity and vertical mixing. Usually the entire county will receive rain as a result of a frontal passage. The rain may occur at any time during the day since frontal storms are not dependent upon local land surface heating.

Following the movement of a cold front across northern Florida, the lower troposphere will be dominated by colder air with relatively warmer air (higher potential temperature) aloft. Such a configuration is stable and acts as an additional inhibitor of vertical mixing. A decrease in the frequency of frontal movement across northern Florida is one probable cause of periodic draft occurrences. A reduction of the frequency of frontal storms will reduce total annual precipitation substantially below annual values of the evaporation demand as estimated by pan evaporation. Rain accompanying frontal storms is usually less than that associated with convective activity and will tend to be more effective for the recharge of soil and surface storage. On the other hand, the intensive rainfall associated with late afternoon convective storms will tend more to recharge the limestone aquifer, particularly in the buildup areas where water runs rapidly off to enter the aquifer through solution sinks. The so-called Floridian aquifer lies near the surface under most of Alachua County. A substantial reduction in the number of frontal passages will cause extensive surface drying with concomitant vegetation stress, lowering of lake levels, and the depletion of the shallow wells.

The ridge extension of the Bermuda High is exceedingly common during the summer months and ordinarily would induce very arid conditions within the Florida peninsula. Were it not for the intense surface heating and the presence of large bodies of water on either side of the peninsula, Florida would be as arid as the great sub-tropical deserts, such as the Sahara Desert at the same latitude. The ocean and the gulf provide moisture, and the differential land-sea heating provide a pressure gradient for the development of sea breeze convergence which powers intense afternoon convective storms. (11)

The climatological summary of the Gainesville station temperature data for the years 1886-1970 is summarized in Table 2-8. Examination of detailed climatological data for 1977, 1978 and 1979 contained in Agronomy Research Reports, AG-79-5,79-4, and 80-5 taken by the University of Florida Institute of Food and Agricultural Sciences, Agricultural Experiment Stations within the Agronomy Department show no significant climate changes over the earlier 84 year period; although precipitation does vary greatly from month to month and year to year. Maximum temperatures in the 90's are common but recordings above 100°F are infrequent due to the nearness to ocean areas and winds which cause marine characteristics to prevail during the summer. Table 2-8 includes mean, maximum and minimum monthly temperatures as well as overall monthly extreme temperatures for the years 1886-1970. The yearly averages are also included.

The Gainesville station precipitation data for 1886-1970 is also summarized in Table 2-8, on a monthly basis with annual values also included. Mean, minimum and maximum values are also reported on a monthly basis.

Gainesville data has shown that the relative humidity averages nearly 88 percent late at night. Early afternoon averages range from about 48 percent in April and May to about 61 percent in July, August and September. Heavy fog forms on 30 to 40 days per year, usually forming late at night

TABLE 2-8

GAINESVILLE CLIMATOLOGICAL DATA SUMMARY (1886 - 1970)
NORMALS, MEANS, AND EXTREMES (13)

	Temperature								Precipitation												Relative Humidity				Wind				
	Normal			Extremes												Snow, Ice Pellets											Fastest Mile		
Month	Daily Maximum	Daily Minimum	Monthly	Record Highest	Year	Record Lowest	Year	Normal Heating Degree Days (Base 65°F)	Normal Total	Maximum Monthly	Year	Minimum Monthly	Year	Maximum in 24 hrs	Year	Mean Total	Maximum Monthly	Year	Maximum in 24 hrs	Year	Hour 01 (Local Time)	Hour 07 (Local Time)	Hour 13 (Local Time)	Hour 19 (Local Time)	Mean Speed	Prevailing Direction	Speed	Direction	Year
J	68.7	46.9	57.8	89	1943	15	1928	2.95	2.95	11.79	1904	.11	1950	2.96	1909	O	O		O										
F	70.8	48.0	59.4	88	1929	6	1899	3.02	3.02	9.2	1901	.12	1943	4.40	1939	T	T	1951	T	1951									
M	76.0	52.8	64.4	96		24	1943	3.41	3.41	12.33	1886	T	1945	3.15	1901	O	O		O										
A	81.4	57.8	69.6	95	1914	32	1919	2.89	2.89	8.95	1949	.14	1899	4.90	1944	O	O		O										
M	87.3	63.7	75.5	101	1945	43	1925	3.20	3.20	10.53	1890	.16	1914	4.06	1920	O	O		O										
J	90.3	69.3	79.8	104	1952	54	1913	6.72	6.72	20.19	1892	1.53	1931	3.82	1946	O	O		O										
J	90.5	71.0	80.8	103	1913	60	1897	7.33	7.33	16.41	1909	2.30	1923	5.71	1909	O	O		O										
A	90.9	71.3	81.1	100	1896	60	1930	7.08	7.08	15.15	1945	1.45	1950	5.77	1937	O	O		O										
S	88.8	69.5	79.1	99	1925	48	1897	5.48	5.48	12.49	1893	1.02	1927	5.60	1950	O	O		O										
O	82.6	61.4	72.0	96	1941	33	1937	3.29	3.29	15.78	1941	.09	1940	9.93	1941	O	O		O										
N	74.6	51.9	63.2	90	1935	20	1970	1.85	1.85	7.61	1888	.10	1936	4.08	1921	O	O		O										
D	68.8	47.1	57.9	87	1951	13	1962	3.00	3.00	9.22	1945	0.0	1889	3.80	1940	O	O	1951	O										
YR	80.9	59.2	70.1	104	1952	6	1899	50.22	50.22	20.19	1892	0.0	1889	9.93	1941	T	T		T	1951									

and dissipating soon after sunrise. Most fog occurs during the period of November through March. With the exception of temperatures and total daily precipitation, no meteorological records are available from past years for the Gainesville area. The nearest station keeping micrometeorological records is the Naval Air Station near Jacksonville. This is approximately 70 miles to the northeast but due to its coastal location, it is unlikely that the Jacksonville data would apply for Gainesville. Due to the lack of this data, a program was set up in 1956 to collect this micrometeorological data which was necessary for the licensing of the UFTR in 1959. Figure 2-15 presents a summary of the wind and precipitation data for the University of Florida for the period of July 1957 through June 1958. Wind data were obtained from wind vanes located on the College of Engineering radar tower at elevations of .25 feet and 30 feet above ground. The wind data are divided into five (5) velocity groups, calm-1, calm-2-4, 5-7, 8-12, and 13+ miles per hour. The radial length of direction lines represented by the wind scale indicates the number of hours for which winds of the designated velocity groups prevailed from the point indicated. Shaded areas represent the number of hours in each velocity range during which precipitation occurred. The detailed study leading to the above results is included in Appendix 2A for completeness. This is the data used to obtain the original UFTR R-56 operating license. Section 2.3.4 contains updated data and the results of diffusion calculations for the UFTR. Prevailing wind directions are from the northwest to northeast in the fall and winter and south to southwest in the spring and summer. Wind velocities generally range from 10 to 16 miles per hour during the day and nearly always drop below 10 miles per hour at night. (2)

2.3.2.2 Potential Influence of the UFTR and Its Facilities on Local Meteorology. Based upon evaluation of the small physical size of the UFTR and small thermal output even at full power (100 Kw), it is concluded that there is no potential for UFTR-caused modifications of the normal or extreme values of meteorological parameters described in Section 2.3.2.1 as a result of the presence and operation of the plant.

2.3.2.3 Local Meteorological Conditions for Design and Operating Bases. Since the UFTR is a self-protected and isolated low-power system with negligible environmental interaction, there are no local meteorological or air quality conditions used for design and operating basis considerations except for those associated with diffusion estimates following an accidental or normal operational release of radioactivity. Both short-and-long-term diffusion estimates are presented in Section 2.3.4. Corresponding diffusion estimates applied for the Design Basis Accident are presented in Secton 2.3.5.

2.3.3 Onsite Meteorological Measurements Program

Because of the self-limiting, low power operation of the UFTR, no onsite meteorological measurements program has been conducted following

UPPER ELEVATION

LOWER ELEVATION

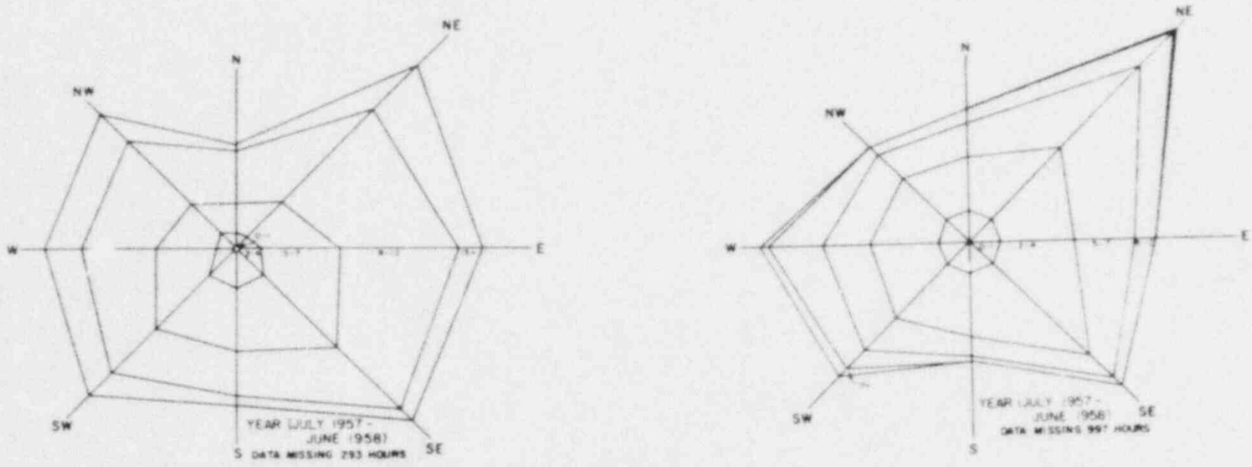


Figure 2-15. Original UFTR Annual Summary of Wind Data Showing Monthly Totals Averaged Over the Year and Used in)riginal UFTR Hazards Summary. (2)

the initial acquisition of meteorological site data for the original UFTR license application. There are meteorological measurement programs in effect at both the Deerhaven plant about 8 miles north of the UFTR site and in a limited way at the Gainesville Regional Airport which is about 5 miles northeast of the UFTR site. It is not felt that any additional measurement programs are needed at this time.

2.3.4 Long-and-Short-Term Diffusion Estimates for the UFTR

The methodology and calculations presented in this section were performed as part of a Master's Thesis project - Reference 14.

2.3.4.1 Objective. This section contains conservative estimates of long and short-term atmospheric diffusion coefficients (χ/Q) for the UFTR site. The atmospheric diffusion model employed in this study is the constant mean wind direction model; the version used is the one recommended by the U.S. Nuclear Regulatory Commission in Regulatory Guide 1.111: "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." The computer code XOQDOQ, developed by the NRC, was used for the calculations (16).

The diffusion of radioactive effluents in the atmosphere is a function of the atmospheric conditions, the topography and the physical and chemical state of the effluents. In the model used for calculations associated with this Safety Analysis Report, the atmospheric conditions were assumed to be defined by the Pasquill stability category as a measure of the atmospheric thermal turbulence, the wind speed and wind direction.

2.3.4.2 Methodology of Calculations of Diffusion Coefficients. There are several equivalent methods to determine the atmospheric thermal turbulence as described in TID-24190:

(i) The combination of insolation and wind speed

(ii) The standard deviations of the azimuthal and polar angles of the wind vector as a function of time as measured by a wind vane with two degrees of freedom

(iii) The temperature gradient, or the measurement of the variation of the temperature with height. This method was used for the compilation of the wind roses used in this study. In practice this variable was determined by measuring the difference in temperatures between two levels of a meteorological tower and later processing the data to obtain hourly averages. This latter procedure was also applied for computing average data for the speed.

One problem sometimes encountered in the acquisition of meteorological data is the existence of wind speeds which are below the anemometer threshold. XOQDOQ distributes these hours within the lowest speed class, with weights in accordance with the direction and stability distribution of the first wind speed class. This distribution was performed for the meteorological data from the Crystal River Nuclear Power Station.

The diffusion coefficient, defined as the atmospheric concentration at a point per unit release, is assumed to follow the pattern of a two dimensional gaussian in the vertical and horizontal directions, and the plume is assumed to be transported along the wind direction. The wind directions considered are the sixteen compass points and the concentrations are averaged within each compass sector by integrating along the horizontal direction. (18) The annual average diffusion coefficient, is the magnitude of concern here. It is calculated in each sector by multiplying the frequency at which the wind blows into this sector times the hourly diffusion coefficient. The resulting equation for the atmospheric diffusion coefficient is (16):

$$X/Q = 2.032\sum_{i,j} f_{ij} (x\bar{u}_i \Sigma_{zj}(x))^{-1} \exp(-h_e^2/2\sigma_{zj}^2 (x)) \qquad (2\text{-}2)$$

where the following definitions apply:

i = wind speed class index;
j = stability class index (usually from 1 to 7 corresponding to Pasquill categories A to G);
f_{ij} = annual frequency that the wind blows into a sector(total number of hours the wind blows into a sector in a year divided by the total number of hours in a year) with speed class "i" and stability class "j";
x = downwind distance from the release point;
$\sigma_{zj}(x)$ = standard vertical plume spread shown in Figure 2-16
$\bar{u}_i$ = wind speed corresponding to wind speed class "i"
h_e = effective stack height (to be defined later)
$\Sigma_{zj}(x)$ = effective vertical plume spread which is given by

$$\Sigma_{zj}(x) = (\sigma_{zj}^2(x) + 0.5\ A_T)^{1/2} \qquad (2\text{-}3)$$

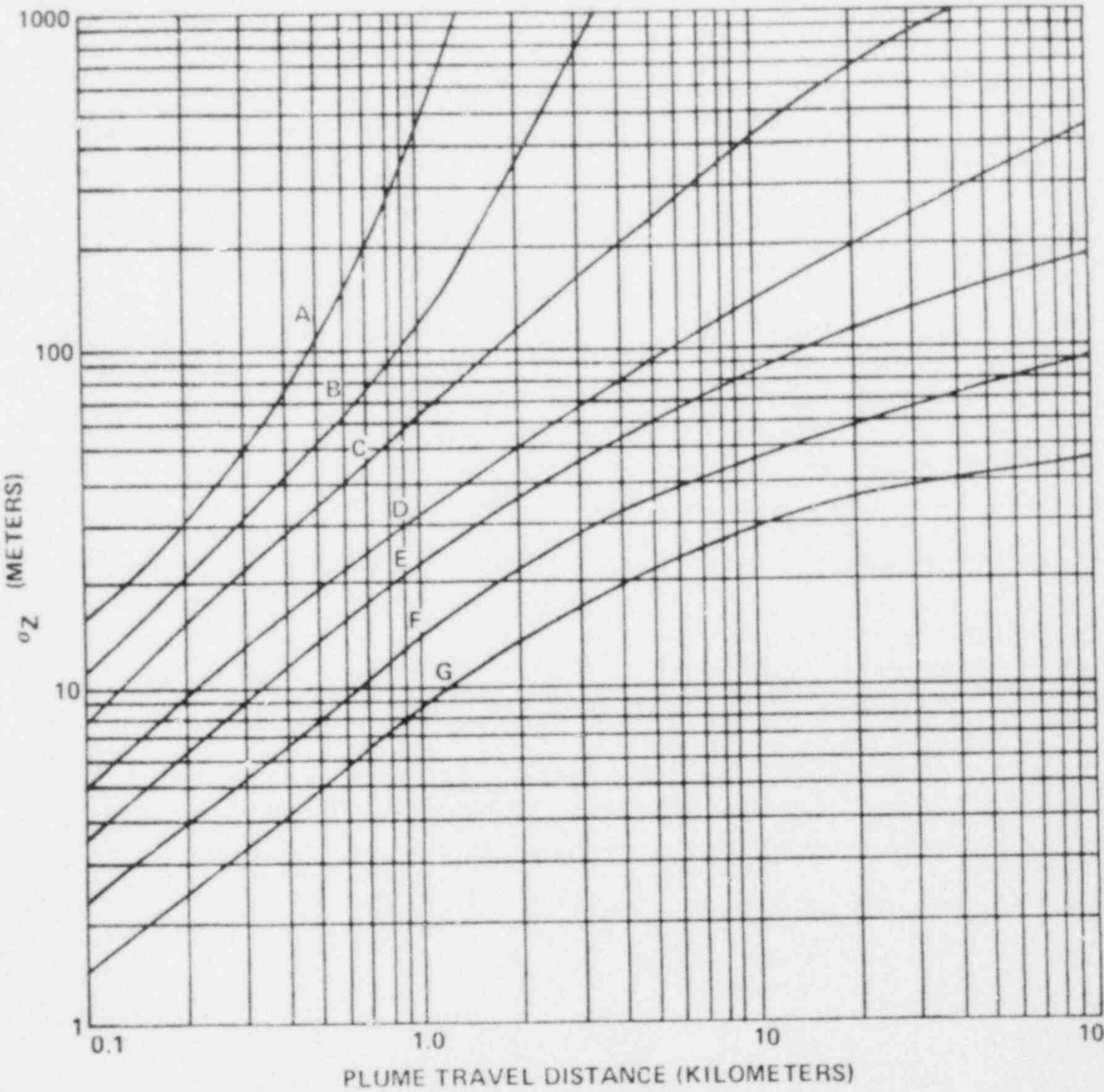
where A_T is the maximum transverse area of the building from which the release takes place (UFTR Reactor Building).

The vertical plume spread is a function of the distance and of the stability class. It increases with distance and with thermal instability. The correction factor shown in Eq. 2.3 accounts for the enhancement in turbulence caused by the building wake.

The effective stack height is given by:

$$h_c = h_g + h_{pr} - c \qquad (2\text{-}4)$$

where the following definitions apply:



Vertical Standard Deviation of Material in a Plume (Letters denote Pasquill Stability Class)

NOTE: THESE ARE STANDARD RELATIONSHIPS AND MAY HAVE TO BE MODIFIED FOR CERTAIN TYPES OF TERRAIN AND/OR CLIMATIC CONDITIONS (E.G., VALLEY, DESERT, OVER WATER).

Figure 2-16. Vertical Standard Deviation of Material in a Plume - Standard Plume Spread as a Function of Downwind Distance, Reference 14.

h_e = effective stack height;
h_g = geometrical stack height;
h_{pr} = plume rise due to the momentum and buoyancy;
c = correction factor for downwash.

The XOQDOQ code contains two kinds of correlations for the plume rise due to momentum and buoyancy, h_{pr}, depending upon the stability class. One correlation applies for stability in Pasquill classes 1 to 4 and the other for stability classes 5 to 7.

First, for neutral and unstable conditions ("j" = 1-4), the following correlation is used for the plume rise in XOQDOQ:

$$h_{pr} = 1.44 \left(\frac{w_o}{u}\right)^{2/3}\left(\frac{x}{D}\right)^{1/3} D \qquad (2\text{-}5)$$

where

w_o = stack exit velocity (meters per second);
x = down wind distance (meters);
u = wind speed at release height (meters per second);
D = internal stack diameter (meters).

The wind speeds at the release height are calculated from the wind speeds at the height which were actually measured, using the following equation:

$$u_r = \left(\frac{h_r}{h_m}\right)^a u_m \qquad (2\text{-}6)$$

where

u_r = wind speed at the release height;
h_r = height of actual release;
h_m = height at which the wind speed was measured;
u_m = measured wind speed.

The "a" is an empirical constant whose value depends on the atmospheric class as follows:

a = 0.24 for "j" = 1 to 4;
a = 0.50 for "j" = 5 to 7.

When the stack exit speed is small compared with the wind speed at the instant of emission, there is a downwash effect, causing the actual effective height of release to be less than the one calculated using the geometrical height corrected by the plume height. The XOQDOQ code uses the Briggs correction when the ratio of exit velocity to wind speed is in the range $\frac{w_o}{u_r} \leq 1.5$; the downwash correction factor becomes:

$$c = 3(1.5 - \frac{w_o}{u_r})D \qquad (2\text{-}7)$$

The plume rise elevation corrected for downwash is then compared with the plume rise due to momentum and buoyancy:

$$h_{pr} = 3\,\frac{w_o}{u} \cdot D \qquad (2\text{-}8)$$

and the smaller value is chosen in the interest of conservatism in the predicted results.

Second, for stable conditions, ("j" = 5-7), two additional correlations are used in XOQDOQ to calculate plume rise as follows:

$$h_{pr} = 283(w_o D)^{1/2} T^{1/4} (g\,\frac{\partial\theta}{\partial z})^{-1/4} \qquad (2\text{-}9)$$

$$h_{pr} = 0.94(w_o D)^{2/3} (gu\,\frac{\partial\theta}{\partial z}/T)^{-1/6} \qquad (2\text{-}10)$$

where

g = acceleration due to gravity (m/sec^2);
T = ambient air temperature (°K);
$\partial\theta/\partial z$ = vertical potential temperature gradient (°K/m).

For stable conditions, the smallest value among the predictions of Eqs. 2-5,2-6,2-7 and 2-8 is chosen for the plume rise by the XOQDOQ code. This selection again assures conservatism in the predicted results.

In research reactors such as the UFTR, operation frequently takes place in short periods of time on the order of several hours as indicated by UFTR operation log books. XOQDOQ accounts for these purge releases by applying the following three formulas for the applicable diffusion coefficient:

$$\chi/Q = (u_i(\pi\sigma_{yj}(x)\sigma_{zj}(x) + 0.5A))^{-1} \qquad (2\text{-}11)$$

$$\chi/Q = (3u_i\pi\sigma_{yj}(x)\sigma_{zj}(x))^{-1} \qquad (2\text{-}12)$$

$$\chi/Q = (u_i\pi\sigma_{yj}(x)\sigma_{zj}(x))^{-1}\exp(-(h_e/\sigma_{zj}(x))^2/2) \qquad (2\text{-}13)$$

The largest diffusion coefficient value predicted in Eqs.(2.11), (2.12), and (2.13) is chosen for each hour. These latter values are then ordered with respect to their frequency of occurrence, and a percentile distribution is obtained.

The output of XOQDOQ is intended to be input to a radiation dose code so each compass section is divided into segments set at different distances and the diffusion coefficients are averaged within each segment. In this fashion the sector-averaged population dose can be calculated more easily.

2.3.4.3 Meteorological Data for Long and Short-Term Diffusion Estimates. Two basic data sets were used for the UFTR diffusion calculations; they are the annual wind rose data obtained from the Gainesville Utilities for the Deerhaven plant and the corresponding wind rose data for the Crystal River Nuclear Power Station. Both wind rose data are included in Appendix 2B. The data from Gainesville Utilities considers only five categories with the standard correspondence of 1 to Pasquill category A, 2 to Pasquill category B, etc., and finally 5 of which includes a combination of categories E,F, and G. Category 5 was distributed into E, F, and G categories, assuming the relative weight corresponding to E, F, and G were the same as for the Crystal River data. The annual wind roses for both sets of data, as given by the output XOQDOQ, are shown in Tables 2-9 and 2-10.

In order to calculate the possible plume rise, building wake and downwash effects, the UFTR release point data contained in Table 2-11 were used.

2.3.4.4 XOQDOQ-Calculated Diffusion Coefficients. As indicated in Table 2-9 and 2-10, the two different computer runs were performed employing the Gainesville and Crystal River wind rose data sets. The annual average diffusion coefficients for the different compass sectors at different distances and the sector averaged diffusion coefficients are shown in Table 2-12 and Table 2-13 for the Gainesville and Crystal River sets of data respectively. The isopleths corresponding to the Gainesville and Crystal River sets of data are shown in Figures 2-17 and 2-18. Due to the small height of the vent (30 feet above mean ground level), the effective release height for each sector as a function of distance is constantly equal to zero.

2.3.4.5 Interpretation of XOQDOQ Results for Diffusion Coefficients. The wind rose data from Gainesville features a relatively isotropic distribution versus the corresponding Crystal River data, which clearly shows the West sector is the one with the worst diffusion characteristics; that is, the diffusion occurs least in the West sector.

Short term radioactivity releases of 8 hours duration were assumed for the analysis of the normal UFTR operations case. This operation time is consistent with normal working periods in the UFTR. The median values for the corresponding short-term UFTR diffusion coefficients to the Gainesville data are plotted in Figure 2-19.

Table 2-9

XOQDOQ-CALCULATED ANNUAL WIND ROSE DATA FOR UFTR USIN' GAINESVILLE DATA

UFTR WITH GAINESVILLE DATA

THE JOINT FREQUENCY DISTRIBUTION, I=WIND SPEED CLASS, J= STABILITY CLASS

DIRECTION =	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
I= 1,J= 1	0.10	0.07	0.07	0.03	0.0	0.0	0.07	0.0	0.03	0.0	0.03	0.0	0.03	0.01	0.07	0.0
I= 2,J= 1	0.21	0.34	0.17	0.31	0.31	0.14	0.17	0.17	0.10	0.03	0.07	0.21	0.24	0.38	0.14	0.14
I= 3,J= 1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 4,J= 1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 5,J= 1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 6,J= 1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 2	0.14	0.07	0.07	0.10	0.14	0.10	0.17	0.14	0.21	0.03	0.17	0.21	0.17	0.07	0.14	0.17
I= 2,J= 2	0.27	0.07	0.41	0.24	0.35	0.34	0.17	0.34	0.10	0.14	0.14	0.17	0.24	0.24	0.21	0.27
I= 3,J= 2	0.27	0.21	0.24	0.34	0.65	0.51	0.41	0.45	0.21	0.21	0.41	0.17	0.72	0.41	0.55	0.51
I= 4,J= 2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 5,J= 2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 6,J= 2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 2,J= 3	0.03	0.10	0.10	0.07	0.07	0.10	0.14	0.14	0.10	0.14	0.0	0.07	0.03	0.17	0.07	0.10
I= 3,J= 3	0.21	0.34	0.48	0.38	0.55	0.68	0.38	0.45	0.38	0.34	0.27	0.27	0.58	0.34	0.31	0.24
I= 4,J= 3	0.07	0.17	0.31	0.31	0.62	0.62	0.31	0.21	0.27	0.21	0.17	0.25	0.56	0.55	0.17	0.10
I= 5,J= 3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.33	0.03	0.0	0.0	0.0
I= 6,J= 3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 4	0.10	0.34	0.38	0.24	0.28	0.24	0.38	0.30	0.42	0.38	0.24	0.34	0.28	0.19	0.34	0.32
I= 2,J= 4	0.92	0.92	1.10	0.82	1.50	1.24	0.86	0.66	0.60	0.40	0.44	0.34	0.86	1.10	0.36	1.16
I= 3,J= 4	0.82	1.20	1.20	1.60	1.58	1.64	0.92	0.76	1.20	0.74	0.86	1.14	2.36	0.72	0.93	1.20
I= 4,J= 4	0.04	0.16	0.72	0.30	0.68	0.48	0.54	0.46	0.66	0.26	0.52	0.73	1.30	0.95	0.39	0.22
I= 5,J= 4	0.0	0.0	0.0	0.0	0.0	0.02	0.0	0.04	0.0	0.0	0.02	0.03	0.06	0.0	0.0	0.0
I= 6,J= 4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.04	0.0	0.0	0.0
I= 1,J= 5	0.48	0.39	0.40	0.46	0.49	0.48	0.32	0.43	0.32	0.41	0.41	0.36	0.41	0.52	0.37	0.41
I= 2,J= 5	0.44	0.31	0.45	0.34	0.57	0.33	0.29	0.16	0.19	0.11	0.12	0.11	0.80	0.81	0.41	0.44
I= 3,J= 5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 4,J= 5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 5,J= 5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 6,J= 5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 6	0.24	0.19	0.20	0.23	0.24	0.24	0.16	0.22	0.16	0.21	0.21	0.18	0.20	0.26	0.19	0.21
I= 2,J= 6	0.22	0.16	0.23	0.17	0.29	0.17	0.13	0.08	0.10	0.06	0.06	0.05	0.40	0.41	0.21	0.22
I= 3,J= 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 4,J= 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 5,J= 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 6,J= 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 7	0.12	0.09	0.10	0.12	0.12	0.12	0.08	0.11	0.08	0.11	0.11	0.09	0.10	0.13	0.09	0.11
I= 2,J= 7	0.11	0.08	0.12	0.09	0.15	0.08	0.08	0.04	0.05	0.03	0.03	0.02	0.20	0.20	0.12	0.11
I= 3,J= 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 4,J= 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 5,J= 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 6,J= 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	4.79	5.21	6.75	6.15	8.87	7.53	5.58	5.16	5.38	3.81	4.28	4.85	9.73	7.46	5.53	5.93

HOURS INPUTTED IN PERCENT.

WIND MEASURED AT 10.0 METERS.
THE MAXIMUM WIND SPEED (METERS/SEC) IN EACH CLASS IS: 1.542 3.085 5.141 8.226 10.796 12.852
THE CONVERSION FACTOR APPLIED TO THE WIND SPEED CLASSES IS 0.447

DISTANCES AND TERRAIN HEIGHTS IN METERS AS FUNCTIONS OF DIRECTION FROM THE SITE:

DIRECTION =	S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
DISTANCE	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.
ELEVATION	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

POOR ORIGINAL

Table 2-10

XOQDOQ-CALCULATED ANNUAL WIND ROSE DATA FOR UFTR USING CRYSTAL RIVER DATA (10/1/76-9/30/77)

UFTR WITH CRYSTAL RIVER DATA OCTOBER 1 76 SEPTEMBER 30 77

THE JOINT FREQUENCY DISTRIBUTION, I=WIND SPEED CLASS, J= STABILITY CLASS

DIRECTION =	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
I= 1,J= 1	0.0	0.01	0.02	0.04	0.04	0.0	0.01	0.03	0.0	0.01	0.03	0.11	0.08	0.04	0.0	0.0
I= 2,J= 1	0.0	0.01	0.04	0.07	0.06	0.0	0.02	0.05	0.0	0.02	0.05	0.19	0.14	0.07	0.0	0.0
I= 3,J= 1	0.0	0.10	0.09	0.34	0.26	0.26	0.11	0.17	0.15	0.16	0.24	1.05	1.42	0.32	0.09	0.01
I= 4,J= 1	0.02	0.29	0.40	0.52	0.52	0.52	0.30	0.29	0.29	0.26	0.73	1.21	1.91	1.25	0.17	0.17
I= 5,J= 1	0.04	0.07	0.10	0.32	0.29	0.16	0.07	0.14	0.11	0.24	0.36	0.06	0.19	0.27	0.12	0.14
I= 6,J= 1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.04	0.09	0.01	0.0	0.0	0.10	0.06	0.01
I= 7,J= 1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.02	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 2	0.0	0.0	0.0	0.00	0.00	0.00	0.0	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.0	0.0
I= 2,J= 2	0.0	0.0	0.0	0.01	0.04	0.04	0.0	0.02	0.05	0.05	0.06	0.14	0.05	0.02	0.0	0.0
I= 3,J= 2	0.0	0.01	0.02	0.16	0.15	0.15	0.09	0.11	0.06	0.10	0.25	0.17	0.24	0.21	0.09	0.0
I= 4,J= 2	0.04	0.07	0.11	0.17	0.17	0.07	0.10	0.07	0.04	0.05	0.11	0.05	0.19	0.26	0.16	0.04
I= 5,J= 2	0.0	0.0	0.01	0.02	0.02	0.02	0.06	0.01	0.05	0.02	0.0	0.05	0.05	0.16	0.06	0.04
I= 6,J= 2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.02	0.01	0.04	0.05	0.06
I= 7,J= 2	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 3	0.0	0.0	0.0	0.00	0.01	0.00	0.00	0.00	0.01	0.00	0.02	0.00	0.00	0.0	0.0	0.0
I= 2,J= 3	0.0	0.0	0.0	0.01	0.04	0.01	0.01	0.01	0.02	0.01	0.07	0.01	0.01	0.0	0.0	0.0
I= 3,J= 3	0.0	0.0	0.06	0.09	0.16	0.04	0.02	0.02	0.05	0.04	0.07	0.24	0.10	0.09	0.04	0.0
I= 4,J= 3	0.0	0.04	0.09	0.06	0.10	0.06	0.05	0.0	0.01	0.04	0.05	0.05	0.06	0.20	0.14	0.06
I= 5,J= 3	0.0	0.0	0.0	0.0	0.01	0.0	0.01	0.0	0.02	0.02	0.02	0.0	0.01	0.07	0.09	0.04
I= 6,J= 3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.01	0.0	0.0	0.05	0.01	0.0
I= 7,J= 3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 4	0.0	0.01	0.03	0.05	0.09	0.04	0.04	0.04	0.05	0.03	0.04	0.08	0.05	0.03	0.03	0.01
I= 2,J= 4	0.0	0.06	0.17	0.30	0.49	0.24	0.21	0.20	0.27	0.17	0.24	0.42	0.29	0.16	0.17	0.04
I= 3,J= 4	0.0	0.39	0.61	0.98	0.70	0.76	0.51	0.37	0.27	0.32	0.57	0.65	0.73	1.07	0.49	0.22
I= 4,J= 4	0.05	0.55	1.73	1.81	0.59	0.50	0.40	0.30	0.42	0.22	0.45	0.56	0.65	1.16	0.95	0.33
I= 5,J= 4	0.02	0.15	0.05	0.31	0.04	0.02	0.04	0.07	0.21	0.24	0.36	0.22	0.14	0.59	0.59	0.10
I= 6,J= 4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.04	0.06	0.01	0.0	0.0	0.06	0.09	0.01
I= 7,J= 4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 5	0.01	0.06	0.05	0.15	0.17	0.15	0.12	0.11	0.10	0.05	0.08	0.06	0.11	0.11	0.06	0.02
I= 2,J= 5	0.02	0.27	0.22	0.70	0.78	0.67	0.55	0.50	0.45	0.22	0.35	0.27	0.47	0.49	0.27	0.09
I= 3,J= 5	0.22	0.98	0.60	1.81	1.92	1.11	0.96	0.85	0.54	0.40	0.36	0.73	[illegible]	0.96	0.50	0.34
I= 4,J= 5	0.20	0.41	0.76	0.72	0.50	0.56	0.34	0.36	0.55	0.50	0.41	0.42	0.64	0.54	0.37	0.30
I= 5,J= 5	0.0	0.0	0.02	0.01	0.0	0.07	0.02	0.10	0.16	0.30	0.12	0.04	0.0	0.10	0.01	0.02
I= 6,J= 5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 7,J= 5	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 6	0.00	0.10	0.07	0.13	0.39	0.29	0.20	0.12	0.04	0.06	0.02	0.04	0.02	0.05	0.04	0.02
I= 2,J= 6	0.01	0.26	0.19	0.35	1.03	0.78	0.52	0.32	0.11	0.16	0.05	0.10	0.06	0.12	0.10	0.06
I= 3,J= 6	0.04	0.41	0.17	0.95	1.23	0.60	0.54	0.12	0.12	0.09	0.06	0.02	0.14	0.04	0.07	0.12
I= 4,J= 6	0.05	0.09	0.14	0.21	0.0	0.02	0.0	0.01	0.02	0.04	0.05	0.0	0.0	0.0	0.01	0.0
I= 5,J= 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.01	0.0	0.0	0.0	0.0	0.0	0.0
I= 6,J= 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 7,J= 6	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 1,J= 7	0.02	0.05	0.09	0.15	0.22	0.15	0.05	0.02	0.00	0.01	0.01	0.01	0.0	0.0	0.01	0.0
I= 2,J= 7	0.06	0.19	0.32	0.52	0.78	0.54	0.17	0.06	0.01	0.05	0.04	0.04	0.0	0.0	0.02	0.0
I= 3,J= 7	0.07	0.41	0.15	0.31	0.36	0.20	0.11	0.12	0.05	0.02	0.02	0.01	0.01	0.0	0.01	0.01
I= 4,J= 7	0.02	0.11	0.05	0.11	0.0	0.0	0.0	0.01	0.04	0.01	0.0	0.0	0.0	0.0	0.0	0.0
I= 5,J= 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 6,J= 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
I= 7,J= 7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.29	0.0	0.0	0.0	0.0	0.0	0.0	0.0
TOTAL	0.91	5.11	6.38	11.42	11.17	8.06	5.65	4.64	4.69	4.11	5.36	7.04	8.60	9.13	4.89	2.24

TOTAL HOURS CONSIDERED ARE 8029

WIND MEASURED AT 10.0 METERS.
THE MAXIMUM WIND SPEED (METERS/SEC) IN EACH CLASS IS: 1.341 2.682 4.470 7.153 9.388 11.176 0.0
THE CONVERSION FACTOR APPLIED TO THE WIND SPEED CLASSES IS 0.447

DISTANCES AND TERRAIN HEIGHTS IN METERS AS FUNCTIONS OF DIRECTION FROM THE SITE:

DIRECTION =	S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
DISTANCE	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.	300.
ELEVATION	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.

POOR ORIGINAL

Table 2-11

UFTR RELEASE POINT SUMMARY DATA

Average vent exit velocity*	0.15 m/sec
Vent inside diameter	0.86 m
Height of the vent release point	8.25 m
Height of the building vent	6.75 m
Minimum UFTR building cross sectional area	165 m^2
Building vent air flow	0.087 m/sec

*The average vent exit velocity was obtained by dividing the air flow by the cross sectional area.

Table 2-12. Annual Average UFTR Diffusion Coefficients Based Upon Gainesville Data.

ONE VENT RELEASE
NO DECAY, UNDEPLETED

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	1.146E-05	2.442E-06	1.693E-06	1.045E-06	5.458E-07	3.530E-07	2.499E-07	1.904E-07	1.519E-07	1.251E-07	1.055E-07
SSW	1.084E-05	3.226E-06	1.593E-06	9.826E-07	5.116E-07	3.268E-07	2.321E-07	1.767E-07	1.404E-07	1.153E-07	[illegible]
SW	1.270E-05	3.767E-06	1.864E-06	1.149E-06	5.969E-07	3.806E-07	2.704E-07	2.051E-07	1.631E-07	1.339E-07	1.127E-07
WSW	1.211E-05	3.624E-06	1.763E-06	1.099E-06	5.725E-07	3.660E-07	2.605E-07	1.981E-07	1.576E-07	1.295E-07	1.092E-07
W	1.510E-05	4.563E-06	2.243E-06	1.391E-06	7.169E-07	4.570E-07	3.245E-07	2.467E-07	1.956E-07	1.605E-07	1.350E-07
WNW	1.324E-05	3.956E-06	1.945E-06	1.198E-06	6.215E-07	3.961E-07	2.812E-07	2.134E-07	1.694E-07	1.390E-07	1.169E-07
NW	1.028E-05	3.049E-06	1.496E-06	9.201E-07	4.766E-07	3.032E-07	2.159E-07	1.633E-07	1.293E-07	1.059E-07	8.890E-08
NNW	1.051E-05	3.134E-06	1.539E-06	9.473E-07	4.916E-07	3.134E-07	2.227E-07	1.691E-07	1.344E-07	1.103E-07	9.270E-08
N	1.011E-05	3.001E-06	1.474E-06	9.073E-07	4.696E-07	2.985E-07	2.114E-07	1.601E-07	1.269E-07	1.039E-07	8.724E-08
NNE	9.511E-06	2.864E-06	1.412E-06	8.716E-07	4.540E-07	2.932E-07	2.065E-07	1.571E-07	1.249E-07	1.025E-07	8.641E-08
NE	9.387E-06	2.811E-06	1.392E-06	8.517E-07	4.432E-07	2.832E-07	2.016E-07	1.513E-07	1.220E-07	1.003E-07	8.443E-08
ENE	9.233E-06	2.746E-06	1.340E-06	8.298E-07	4.302E-07	2.739E-07	1.943E-07	1.474E-07	1.170E-07	9.593E-08	[illegible]
E	1.505E-05	4.490E-06	2.207E-06	1.359E-06	7.056E-07	4.498E-07	3.195E-07	2.425E-07	1.925E-07	1.579E-07	1.329E-07
ESE	1.478E-05	4.450E-06	2.192E-06	1.353E-06	7.062E-07	4.525E-07	3.228E-07	2.459E-07	1.960E-07	1.613E-07	1.359E-07
SE	1.122E-05	3.346E-06	1.644E-06	1.013E-06	5.266E-07	3.361E-07	2.393E-07	1.815E-07	1.443E-07	1.185E-07	9.972E-08
SSE	1.241E-05	3.712E-06	1.826E-06	1.126E-06	5.850E-07	3.713E-07	2.654E-07	2.019E-07	1.562E-07	1.315E-07	1.117E-07

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES

BEARING	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	9.074E-08	5.133E-08	3.451E-08	1.964E-08	1.352E-08	1.005E-08	7.931E-09	6.455E-09	5.462E-09	[illegible]	[illegible]
SSW	8.321E-08	4.653E-08	3.101E-08	1.756E-08	1.132E-08	9.312E-09	6.806E-09	5.512E-09	4.696E-09	[illegible]	[illegible]
SW	[illegible]	5.305E-08	3.594E-08	2.046E-08	1.312E-08	1.021E-08	7.997E-09	[illegible]	[illegible]	[illegible]	[illegible]
WSW	[illegible]	[illegible]	3.515E-08	2.011E-08	[illegible]	[illegible]	7.913E-09	6.451E-09	[illegible]	[illegible]	[illegible]
W	1.138E-07	6.480E-08	4.331E-08	2.464E-08	1.665E-08	1.232E-08	9.652E-09	7.469E-09	6.345E-09	5.335E-09	[illegible]
WNW	1.002E-07	5.604E-08	3.735E-08	2.128E-08	1.435E-08	[illegible]	8.335E-09	[illegible]	5.654E-09	4.872E-09	[illegible]
NW	7.675E-08	4.253E-08	2.834E-08	1.612E-08	1.089E-08	8.043E-09	6.294E-09	[illegible]	4.369E-09	[illegible]	[illegible]
NNW	7.965E-08	4.471E-08	2.904E-08	1.734E-08	[illegible]	[illegible]	[illegible]	[illegible]	4.608E-09	[illegible]	[illegible]
N	[illegible]	4.155E-08	2.769E-08	1.556E-08	[illegible]	7.757E-09	6.059E-09	[illegible]	4.115E-09	[illegible]	[illegible]
NNE	7.417E-08	4.157E-08	2.779E-08	1.594E-08	[illegible]	7.914E-09	6.217E-09	[illegible]	[illegible]	[illegible]	[illegible]
NE	7.251E-08	4.073E-08	2.734E-08	1.566E-08	1.063E-08	7.387E-09	6.176E-09	[illegible]	[illegible]	[illegible]	3.166E-09
ENE	6.914E-08	3.867E-08	2.577E-08	1.468E-08	9.914E-09	7.332E-09	5.743E-09	[illegible]	4.012E-09	[illegible]	[illegible]
E	1.140E-07	6.381E-08	4.255E-08	2.426E-08	1.643E-08	1.214E-08	9.500E-09	[illegible]	[illegible]	[illegible]	4.934E-09
ESE	1.169E-07	6.587E-08	4.415E-08	2.535E-08	1.722E-08	1.279E-08	1.005E-08	[illegible]	[illegible]	[illegible]	[illegible]
SE	8.563E-08	4.803E-08	3.209E-08	1.834E-08	1.241E-08	9.190E-09	7.204E-09	5.870E-09	4.920E-09	4.213E-09	3.664E-09
SSE	9.497E-08	5.327E-08	3.557E-08	2.031E-08	1.373E-08	1.016E-08	7.963E-09	6.485E-09	5.434E-09	4.651E-09	[illegible]

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT — SEGMENT BOUNDARIES IN MILES

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	1.754E-06	5.657E-07	2.529E-07	1.527E-07	1.059E-07	[illegible]	[illegible]	[illegible]	6.576E-09	[illegible]
SSW	1.697E-06	5.741E-07	[illegible]	1.412E-07	9.735E-08	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
SW	1.574E-06	6.273E-07	2.773E-07	1.640E-07	1.129E-07	5.540E-08	2.295E-08	1.323E-08	[illegible]	[illegible]
WSW	1.883E-06	5.974E-07	2.637E-07	1.587E-07	[illegible]	5.136E-08	2.356E-08	1.316E-08	6.471E-09	4.641E-09
W	2.376E-06	7.450E-07	3.306E-07	1.957E-07	1.354E-07	6.653E-08	2.922E-08	1.241E-08	[illegible]	[illegible]
WNW	2.060E-06	6.494E-07	2.847E-07	1.704E-07	1.173E-07	5.765E-08	2.175E-08	1.072E-08	[illegible]	[illegible]
NW	1.505E-06	4.981E-07	2.177E-07	1.300E-07	9.932E-08	4.375E-08	1.651E-08	[illegible]	[illegible]	[illegible]
NNW	1.671E-06	5.117E-07	2.255E-07	1.351E-07	9.311E-08	4.548E-08	[illegible]	[illegible]	[illegible]	3.456E-09
N	1.562E-06	4.903E-07	2.141E-07	1.276E-07	8.751E-08	4.271E-08	1.551E-08	7.514E-09	4.963E-09	[illegible]
NNE	1.495E-06	4.740E-07	2.091E-07	1.259E-07	8.668E-08	[illegible]	1.621E-08	7.999E-09	[illegible]	3.641E-09
NE	1.459E-06	4.679E-07	2.047E-07	1.226E-07	8.462E-08	[illegible]	1.603E-08	7.943E-09	[illegible]	3.617E-09
ENE	1.428E-06	4.435E-07	1.968E-07	1.177E-07	8.090E-08	3.971E-08	[illegible]	7.341E-09	4.686E-09	3.354E-09
E	2.370E-06	7.372E-07	3.235E-07	1.937E-07	1.334E-07	[illegible]	[illegible]	1.222E-08	7.763E-09	5.563E-09
ESE	2.321E-06	7.371E-07	3.266E-07	1.970E-07	1.364E-07	6.785E-08	[illegible]	1.287E-08	8.725E-09	5.912E-09
SE	1.742E-06	5.501E-07	2.419E-07	1.451E-07	1.001E-07	[illegible]	1.776E-08	9.254E-09	5.839E-09	4.221E-09
SSE	1.934E-06	6.110E-07	2.697E-07	1.611E-07	1.110E-07	[illegible]	2.378E-08	1.024E-08	6.507E-09	4.660E-09

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	8.26		REP. WIND HEIGHT	(METERS)	10.0
DIAMETER	(METERS)	0.96		BUILDING HEIGHT	(METERS)	6.8
EXIT VELOCITY	(METERS)	0.15		BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	163.0
				HEAT EMISSION RATE	(CAL/SEC)	0.0

AT THE RELEASE HEIGHT:
VENT RELEASE MODE — WIND SPEED (METERS/SEC)
ELEVATED — LESS THAN 0.030
MIXED — BETWEEN [illegible]

AT THE MEASURED WIND HEIGHT (10.0 METERS):
VENT RELEASE MODE — WIND SPEED (METERS/SEC) STABLE CONDITIONS — WIND SPEED (METERS/SEC) UNSTABLE/NEUTRAL CONDITIONS
ELEVATED — LESS THAN [illegible] — LESS THAN 0.030
MIXED — BETWEEN [illegible] — [illegible] 0.030 AND 0.15[illegible]

POOR ORIGINAL

Table 2-13. Annual Average UFTR Diffusion Coefficients Based Upon Crystal River Data.

ONE VENT RELEASE
NO DECAY, UNDEPLETED

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) DISTANCE IN MILES

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	1.422E-06	4.540E-07	2.315E-07	1.462E-07	7.874E-08	5.156E-08	3.740E-08	2.889E-08	2.328E-08	1.935E-08	1.646E-08
SSW	8.204E-06	2.635E-06	1.339E-06	8.429E-07	4.511E-07	2.939E-07	2.123E-07	1.634E-07	1.313E-07	1.088E-07	9.231E-08
SW	8.215E-06	2.631E-06	1.347E-06	8.504E-07	4.556E-07	2.965E-07	2.138E-07	1.644E-07	1.319E-07	1.092E-07	9.253E-08
WSW	1.644E-05	5.252E-06	2.678E-06	1.686E-06	9.013E-07	5.862E-07	4.227E-07	3.248E-07	2.606E-07	2.158E-07	1.828E-07
W	2.149E-05	7.107E-06	3.668E-06	2.325E-06	1.252E-06	8.180E-07	5.919E-07	4.562E-07	3.669E-07	3.044E-07	2.584E-07
WNW	1.492E-05	4.963E-06	2.568E-06	1.628E-06	8.771E-07	5.730E-07	4.146E-07	3.194E-07	2.569E-07	2.130E-07	1.806E-07
NW	7.717E-06	3.193E-06	1.634E-06	1.033E-06	5.516E-07	3.583E-07	2.531E-07	1.982E-07	1.589E-07	1.314E-07	1.113E-07
NNW	6.873E-06	2.225E-06	1.135E-06	7.113E-07	3.777E-07	2.443E-07	1.754E-07	1.343E-07	1.074E-07	8.866E-08	7.494E-08
N	5.717E-06	1.792E-06	9.003E-07	5.599E-07	2.944E-07	1.892E-07	1.352E-07	1.031E-07	8.219E-08	6.765E-08	5.704E-08
NNE	4.389E-06	1.390E-06	7.047E-07	4.403E-07	2.328E-07	1.501E-07	1.075E-07	8.211E-08	6.555E-08	5.401E-08	4.559E-08
NE	4.680E-06	1.422E-06	7.097E-07	4.392E-07	2.294E-07	1.466E-07	1.042E-07	7.911E-08	6.282E-08	5.154E-08	4.333E-08
ENE	5.729E-06	1.705E-06	8.449E-07	5.222E-07	2.735E-07	1.751E-07	1.246E-07	9.474E-08	7.531E-08	6.183E-08	[illegible]
E	6.258E-06	1.845E-06	9.098E-07	5.603E-07	2.530E-07	1.675E-07	1.334E-07	1.014E-07	8.063E-08	6.620E-08	5.569E-08
ESE	6.832E-06	2.061E-06	1.026E-06	6.337E-07	3.367E-07	2.111E-07	1.499E-07	1.138E-07	9.027E-08	7.400E-08	6.217E-08
SE	4.491E-06	1.337E-06	7.037E-07	4.373E-07	2.208E-07	1.463E-07	1.040E-07	7.897E-08	6.271E-08	5.143E-08	4.322E-08
SSE	2.121E-06	6.587E-07	3.309E-07	2.055E-07	1.070E-07	6.912E-08	4.929E-08	3.753E-08	2.988E-08	2.456E-08	2.068E-08

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) DISTANCE IN MILES

BEARING	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	1.425E-08	8.260E-09	5.646E-09	3.329E-09	2.303E-09	1.731E-09	1.374E-09	1.132E-09	9.577E-10	8.267E-10	7.251E-10
SSW	7.976E-08	4.501E-08	3.111E-08	1.819E-08	1.248E-08	9.343E-09	7.388E-09	6.064E-09	5.114E-09	4.403E-09	3.853E-09
SW	7.587E-08	4.570E-08	3.096E-08	1.803E-08	1.235E-08	9.235E-09	7.295E-09	5.983E-09	5.043E-09	4.340E-09	3.796E-09
WSW	1.575E-07	9.013E-08	6.119E-08	3.562E-08	2.440E-08	1.824E-08	1.440E-08	1.191E-08	[illegible]	[illegible]	7.464E-09
W	2.234E-07	1.276E-07	8.746E-08	5.122E-08	3.521E-08	2.640E-08	2.085E-08	1.716E-08	1.448E-08	1.249E-08	1.092E-08
WNW	1.563E-07	8.951E-08	6.112E-08	3.574E-08	[illegible]	1.839E-08	1.455E-08	1.194E-08	1.007E-08	8.675E-09	[illegible]
NW	9.559E-08	5.478E-08	3.699E-08	2.145E-08	1.464E-08	1.091E-08	8.566E-09	7.033E-09	5.914E-09	5.079E-09	[illegible]
NNW	6.455E-08	3.660E-08	2.462E-08	1.419E-08	9.643E-09	7.164E-09	5.628E-09	4.594E-09	3.855E-09	3.305E-09	[illegible]
N	4.903E-08	2.758E-08	1.845E-08	1.056E-08	7.140E-09	5.286E-09	4.141E-09	3.372E-09	2.824E-09	2.417E-09	2.104E-09
NNE	3.922E-08	2.215E-08	1.486E-08	8.535E-09	5.784E-09	4.295E-09	3.370E-09	2.748E-09	2.305E-09	1.975E-09	1.722E-09
NE	3.715E-08	2.074E-08	1.380E-08	7.838E-09	5.285E-09	3.903E-09	3.052E-09	2.482E-09	2.076E-09	1.775E-09	1.544E-09
ENE	4.464E-08	2.498E-08	1.665E-08	9.486E-09	6.439E-09	4.739E-09	3.711E-09	3.021E-09	2.530E-09	2.165E-09	1.885E-09
E	4.778E-08	2.672E-08	1.780E-08	1.012E-08	6.820E-09	5.035E-09	3.936E-09	3.199E-09	2.676E-09	2.287E-09	1.989E-09
ESE	5.326E-08	2.961E-08	1.964E-08	1.109E-08	7.441E-09	5.471E-09	4.263E-09	3.455E-09	2.883E-09	2.455E-09	2.133E-09
SE	3.704E-08	2.060E-08	1.366E-08	7.717E-09	5.174E-09	3.804E-09	2.963E-09	2.401E-09	2.002E-09	1.707E-09	1.481E-09
SSE	1.775E-08	9.935E-09	6.620E-09	3.764E-09	2.534E-09	1.869E-09	1.459E-09	1.185E-09	9.898E-10	8.451E-10	7.342E-10

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT

SEGMENT BOUNDARIES IN MILES

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	2.439E-07	8.166E-08	3.777E-08	2.339E-08	1.650E-08	8.430E-09	3.387E-09	1.740E-09	1.135E-09	8.276E-10
SSW	1.407E-06	4.683E-07	2.145E-07	1.319E-07	9.255E-08	4.682E-08	1.652E-08	9.397E-09	6.060E-09	4.410E-09
SW	1.412E-06	4.726E-07	2.161E-07	1.325E-07	9.278E-08	[illegible]	1.638E-08	9.290E-09	[illegible]	4.347E-09
WSW	2.709E-06	9.356E-07	4.271E-07	2.619E-07	1.833E-07	9.238E-08	3.631E-08	1.835E-08	1.184E-08	8.572E-09
W	3.835E-06	1.297E-06	5.979E-07	3.686E-07	2.591E-07	1.314E-07	5.216E-08	2.655E-08	1.721E-08	1.250E-08
WNW	2.163E-06	9.089E-07	4.188E-07	2.580E-07	1.813E-07	9.187E-08	3.641E-08	1.850E-08	1.198E-08	8.685E-09
NW	1.715E-06	5.726E-07	2.609E-07	1.596E-07	1.116E-07	5.602E-08	2.187E-08	1.098E-08	7.053E-09	[illegible]
NNW	1.109E-06	3.926E-07	1.774E-07	1.080E-07	7.516E-08	3.749E-08	1.449E-08	7.211E-09	4.608E-09	3.311E-09
N	9.472E-07	3.066E-07	1.368E-07	8.263E-08	5.722E-08	2.829E-08	1.079E-08	5.322E-09	3.383E-09	[illegible]
NNE	7.395E-07	2.422E-07	1.087E-07	6.588E-08	4.573E-08	2.271E-08	8.721E-09	4.323E-09	2.757E-09	1.978E-09
NE	7.478E-07	2.392E-07	1.055E-07	6.318E-08	4.347E-08	2.130E-08	8.027E-09	3.931E-09	2.490E-09	1.779E-09
ENE	8.925E-07	2.810E-07	1.261E-07	7.573E-08	5.219E-08	2.565E-08	9.711E-09	4.773E-09	3.031E-09	2.165E-09
E	9.622E-07	3.055E-07	1.350E-07	8.108E-08	5.588E-08	2.743E-08	1.036E-08	5.071E-09	3.210E-09	2.292E-09
ESE	1.082E-06	3.449E-07	1.518E-07	9.078E-08	6.237E-08	3.043E-08	1.137E-08	5.513E-09	3.456E-09	2.463E-09
SE	7.394E-07	2.365E-07	1.053E-07	6.306E-08	4.336E-08	2.117E-08	7.908E-09	3.833E-09	2.410E-09	1.711E-09
SSE	3.480E-07	1.123E-07	4.988E-08	3.004E-08	2.074E-08	1.020E-08	3.852E-09	1.882E-09	1.169E-09	8.468E-10

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	10.00
DIAMETER	(METERS)	0.60
EXIT VELOCITY	(METERS)	1.00
REP. WIND HEIGHT	(METERS)	10.0
BUILDING HEIGHT	(METERS)	5.0
BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	250.0
HEAT EMISSION RATE	(CAL/SEC)	0.0

AT THE RELEASE HEIGHT:

VENT RELEASE MODE	WIND SPEED (METERS/SEC)
ELEVATED	LESS THAN 0.200
MIXED	BETWEEN 0.200 AND 1.000
GROUND LEVEL	ABOVE 1.000

AT THE MEASURED WIND HEIGHT (10.0 METERS):

VENT RELEASE MODE	WIND SPEED (METERS/SEC) STABLE CONDITIONS	WIND SPEED (METERS/SEC) UNSTABLE/NEUTRAL CONDITIONS
ELEVATED	LESS THAN 0.200	LESS THAN 0.200
MIXED	BETWEEN 0.200 AND 1.000	BETWEEN 0.200 AND 1.000
GROUND LEVEL	ABOVE 1.000	ABOVE 1.000

POOR ORIGINAL

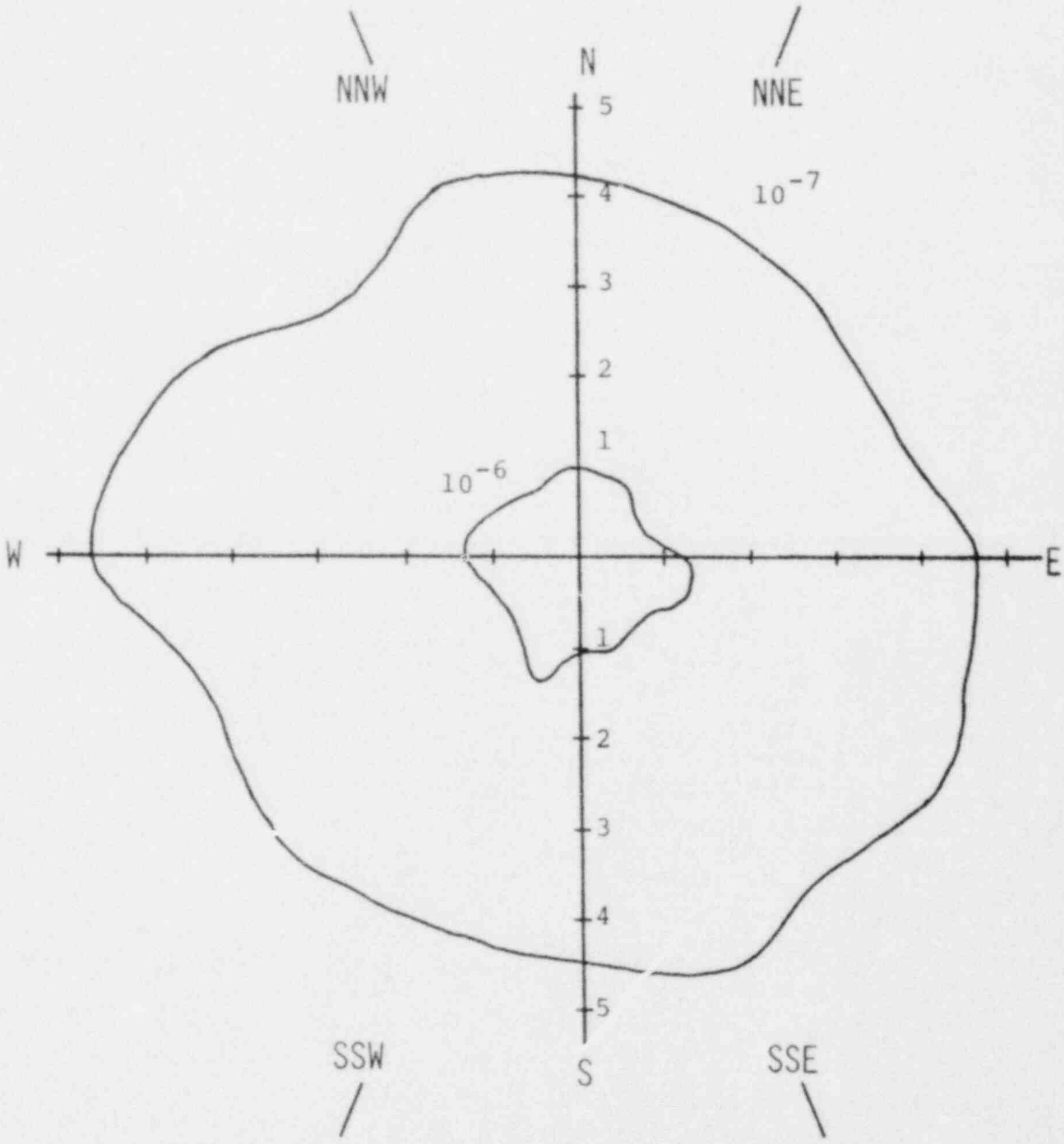


Figure 2-17. Annual Average Isopleths Obtained with Gainesville Data.

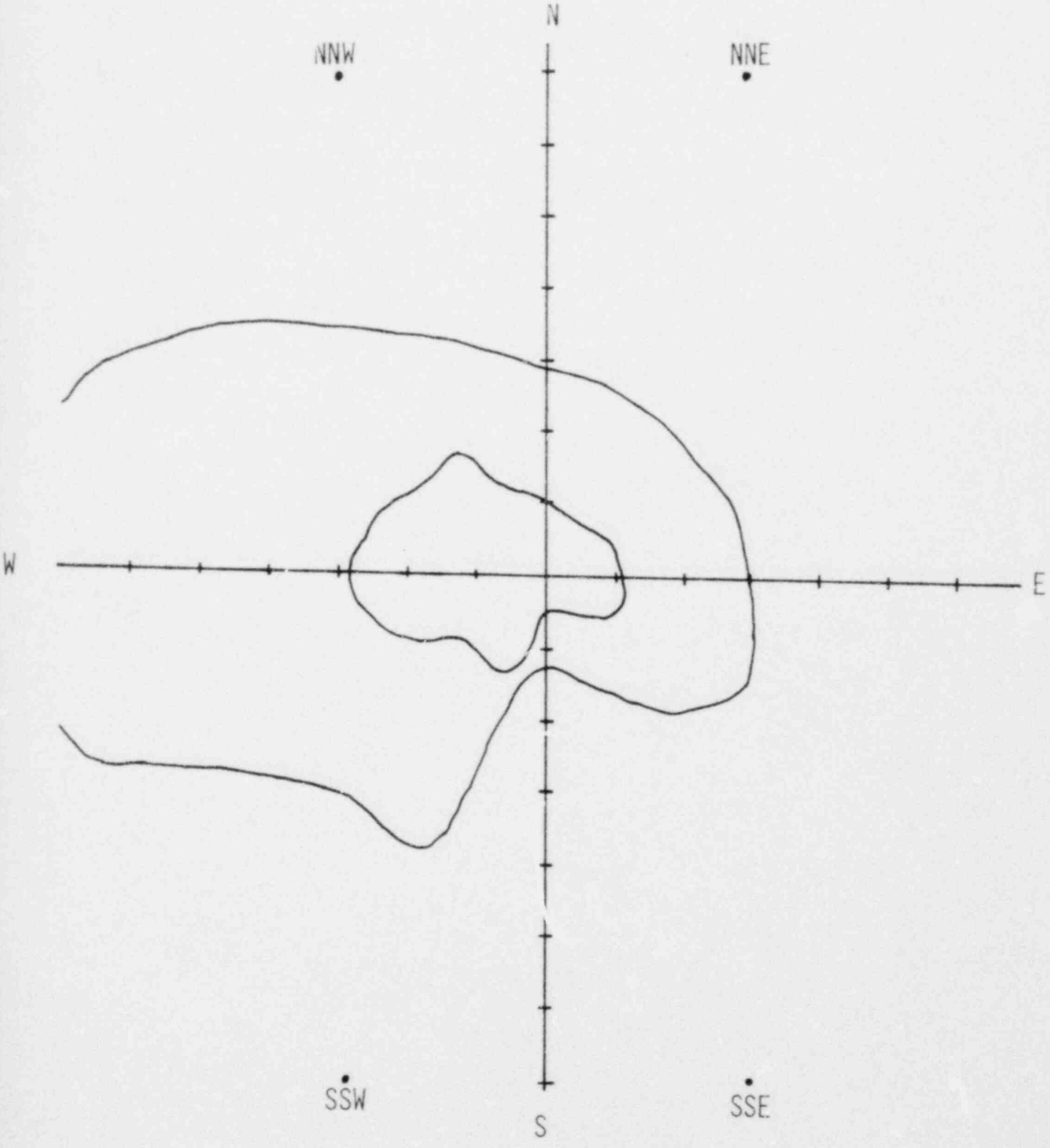


Figure 2-18. Annual Average Isopleths Obtained With Crystal River Data.

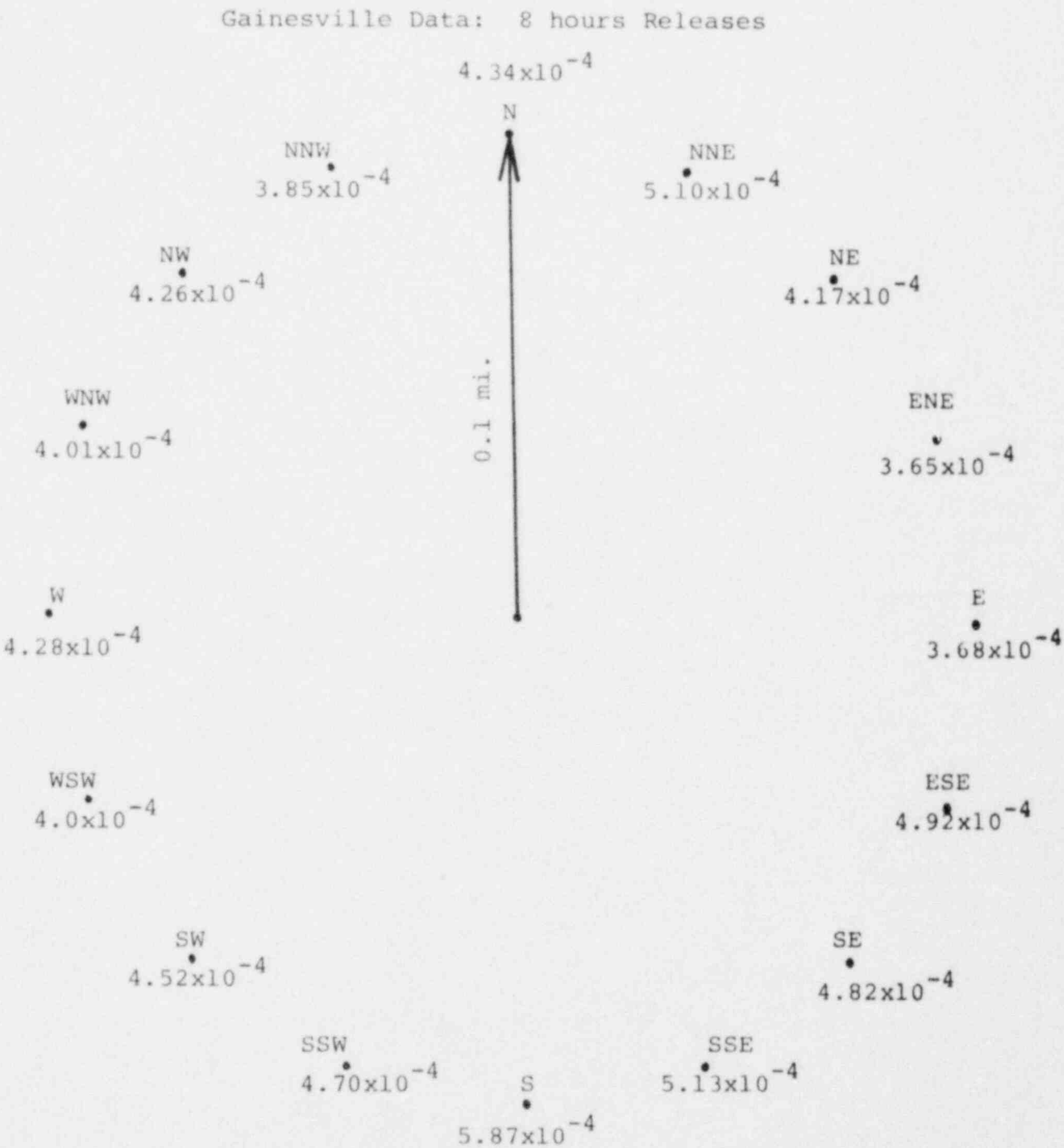


Figure 2-19. Median Values of Short Term UFTR Diffusion Coefficients.

The diffusion coefficients were also calculated at special locations, intended to represent the highest exposed individuals for the study of radiation dose in normal operation. A distance of 0.10 miles was selected as a limit for the model used. See Figure 2-20. The highest diffusion coefficient was only 7.2 E-05 sec/m^3 corresponding to the West sector as indicated by the results shown in Figure 2-20.

2.3.4.6 Experimental Verification of XOQDOQ Results. Regarding the experimental verification of the model employed in XOQDOQ for an urban area, a recent diffusion experimental study performed at the University of California in Los Angeles gives evidence of conservation in the Gaussian model used to calculate diffusion coefficients for an urban area (18). However, the mathematical model which was employed in California did not consider the building wake and the downwash effects. With this simplified model, the predicted diffusion coefficients were more than ten times above those experimentally determined. Although the UCLA and UFTR cases cannot be compared on an absolute basis, a relative comparison should be valid. Therefore, because of the large conservative discrepancy between predicted diffusion coefficients and those actually measured, the calculated results for the UFTR shown in Figure 2-19 and Figure 2-20 are also expected to be very conservative.

2.3.5 Diffusion Estimates for the Design Basis Accident

The methodology and calculations presented in this section were performed as part of a Master's Thesis project - Reference 14.

2.3.5.1 Objective. This section contains conservative estimates of atmospheric diffusion coefficients (χ/Q) for the UFTR site following a design basis accident. Since those coefficients are for times following an accident, they represent short-term diffusion estimates for the UFTR.

2.3.5.2 Methodology for Calculation of Diffusion Coefficients for the Design Basis Accident. There are two approaches, both conservative, recommended by the Nuclear Regulatory Commission in Regulatory Guide 1.111: "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors." (15) The first one uses generic meteorological conditions, and is the more conservative; the second method allows the use of local meteorological conditions and is less conservative since credit for increased diffusion is possible in some regions. The results of calculations for both methods are presented in Section 2.3.5.3.

For generic conservative NRC conditions, the three cases described in the next three sections are considered for different exposure times ranging from initiation of the release up to 30 days.

2.3.5.2.1 Case 1: Exposure Times Less Than Two Hours. For exposure times of less than two hours, the following equation is used:

$$\chi/Q = (\bar{u}\pi\Sigma_y\Sigma_z)^{-1} \qquad (2\text{-}13)$$

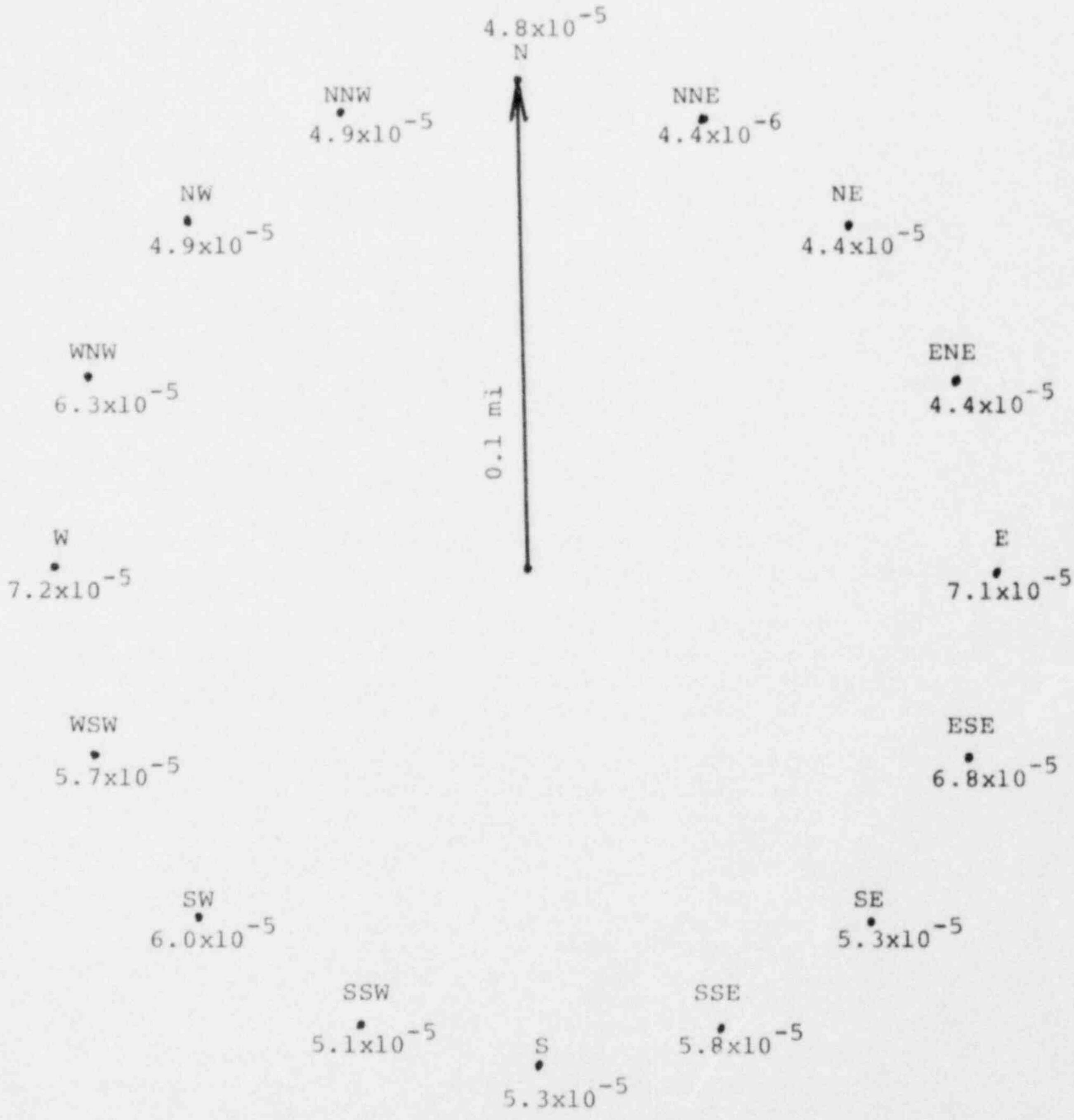


Figure 2-20. Directional Variation of Annual Average Diffusion Coefficients at 0.1 Mile Distance from UFTR.

with

$$\Sigma_y = (\sigma_y^2 + 0.5\ A_T/\pi)^{1/2} \qquad (2\text{-}14)$$

$$\Sigma_z = (\sigma_z^2 + 0.5\ A_T/\pi)^{1/2} \qquad (2\text{-}15)$$

where

χ/Q = diffusion coefficient for the period 0 to 2 hours (sec/m^3)

$\bar{u}$ = windspeed assumed to be 1 m/sec

σ_y = horizontal standard deviation of the plume corresponding to Pasquill category F (m)

σ_z = vertical standard deviation of the plume corresponding to Pasquill category F (m)

A_T = the minimum cross sectional area for the vent's building (m^2).

2.3.5.2.2 Case 2: Exposure Times from 2 to 24 Hours. For exposure times from 2 to 24 hours, the diffusion coefficient corresponds to the sector-averaged model obtained by integrating along the horizontal direction, in the same way as was done for the normal operations case. The following expression is obtained for the diffusion coefficient from 2 to 24 hours.

$$\chi/Q = \frac{2.032}{\bar{u}\sigma_2 x} \qquad (2\text{-}16)$$

where:

x = downward distance (m)

Again from 2 to 24 hours, Pasquill category F, and a windspeed of 1 meter/second are assumed.

2.3.5.2.3 Case 3: Exposure Times from 1 to 30 Days. For exposure times from 1 to 30 days, Eq. (2-16) is still applied; however, in this time range the following atmospheric conditions are assumed:

Time	Atmospheric Condition
1 to 4 days	40% Pasquill Category D, wind speed of 3 m/sec
	60% Pasquill Category F, wind speed of 3 m/sec
4 to 30 days	33% Pasquill Category C, wind speed of 3 m/sec
	33% Pasquill Category D, wind speed of 2 m/sec

Figure 2-21 shows the diffusion coefficients for these sets of conditions as presented in NRC Regulatory Guide 1.4 (20).

For the less conservative local meteorological conditions, the NRC-recommended procedure is to use the 15th percentile value of the hourly diffusion coefficients and the annual average diffusion coefficient for each sector. These two values of diffusion coefficients are plotted on a log-log graph with the times 1 and 8760 hours ($\sim$ 1 year) as abscissas values. These two points are then joined by a straight line. The diffusion coefficient corresponding to any period of duration is then obtained by simply reading the coefficient from a log-log graph.

2.3.5.3 Calculation of the Site Specific Diffusion Coefficients for the Design Basis Accident. Figure 2-21 from Reference 14 shows the variation of DBA diffusion coefficients with distance from the reactor vent starting at 100 meters. Several runs of the computer code XOODOO were performed in order to calculate the short-term diffusion coefficients. The locations were selected at 0.10 mile intervals from the reactor vent as shown in the UFTR environs diagram in Figure 2-22 and the 16 sectors examined (See Figure 2-9). Note that the distance to the Shands Teaching Hospital, selected and supported as the urban boundary in Section 12.4 on dose assessment, is nearly 0.5 miles from the UFTR. The releases were assumed to be purges of 2 hours, 6 hours, 16 hours, 3 days and 26 days duration, corresponding to the periods 0-2 hours, 2-8 hours, 8 hours to 1 day, 1-4 days, 4-30 days respectively. The resultant site specific diffusion coefficients for the worst sector for each time period are shown in Table 2-14 and graphically summarized in Figure 2-23. In general, the diffusion coefficient for the worst sector decreases with duration of the interval and with distance from the release point as expected. The decrease with increasing vent distance greatly reduces maximum doses for a design basis accident.

Table 2-15 shows the Design Basis Accident diffusion coefficients obtained using the NRC standard meteorology at 0.1 mile intervals from the reactor vent. These diffusion coefficients are much larger and hence more conservative than those coefficients obtained using local meteorology as presented in Table 2-14.

2.4 Hydrologic Engin ering

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities. The information in this secton is taken from Reference 2 (the original UFTR Hazards Summary Report which served as the SAR for original operation) with some changes to indicate alterations in the site environs and facilities since the first licensing of the UFTR.

The terrain in the vicinity of Gainesville is gently rolling and the soil is sandy with the exception of relatively small areas of muckland along the shorelines of the fresh water lakes and ponds which are numerous to the east and south of Gainesville.

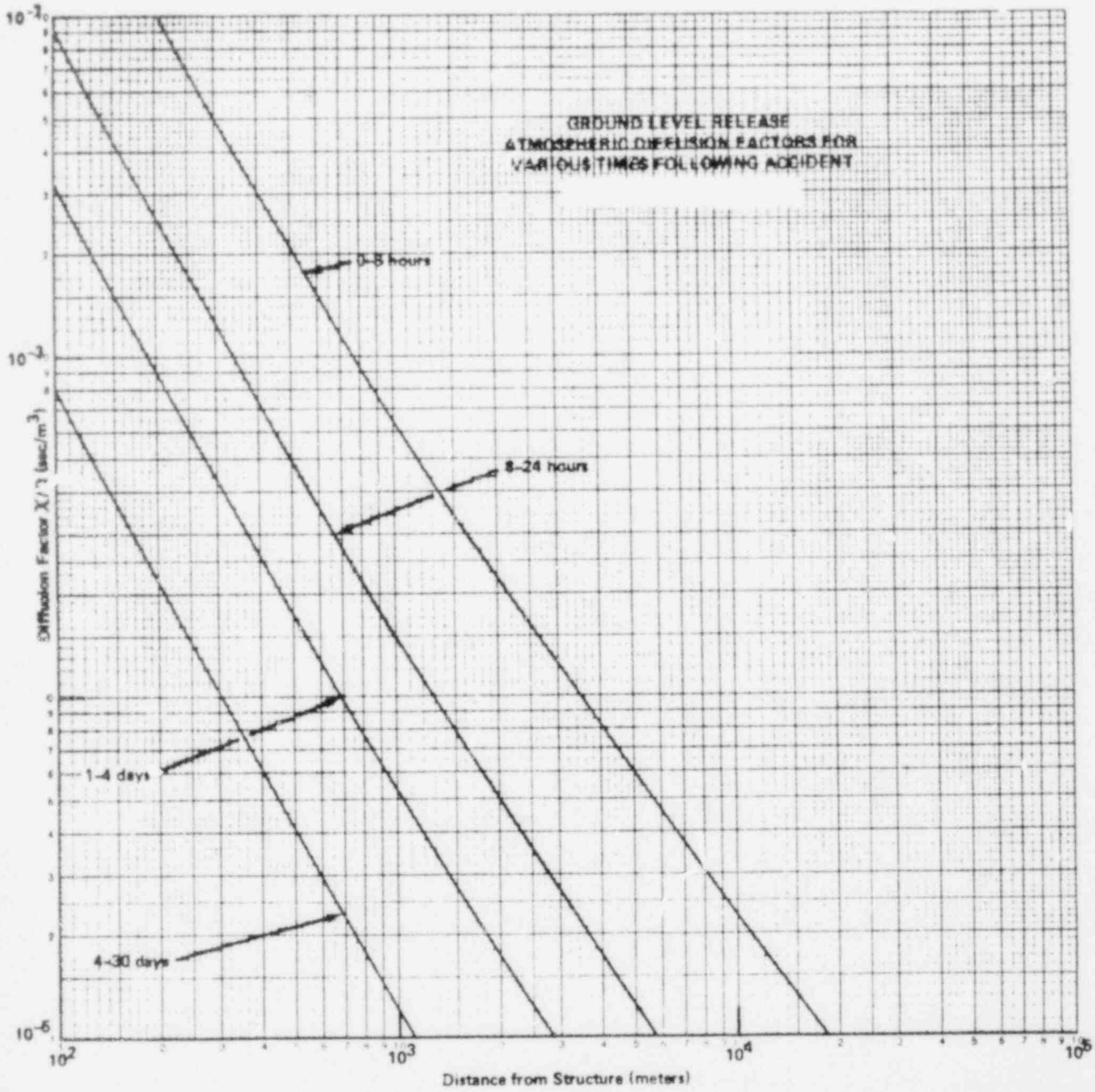


Figure 2-21. Design Basis Accident Diffusion Coefficients With NRC Standard Methodology, Reference 20.

POOR ORIGINAL

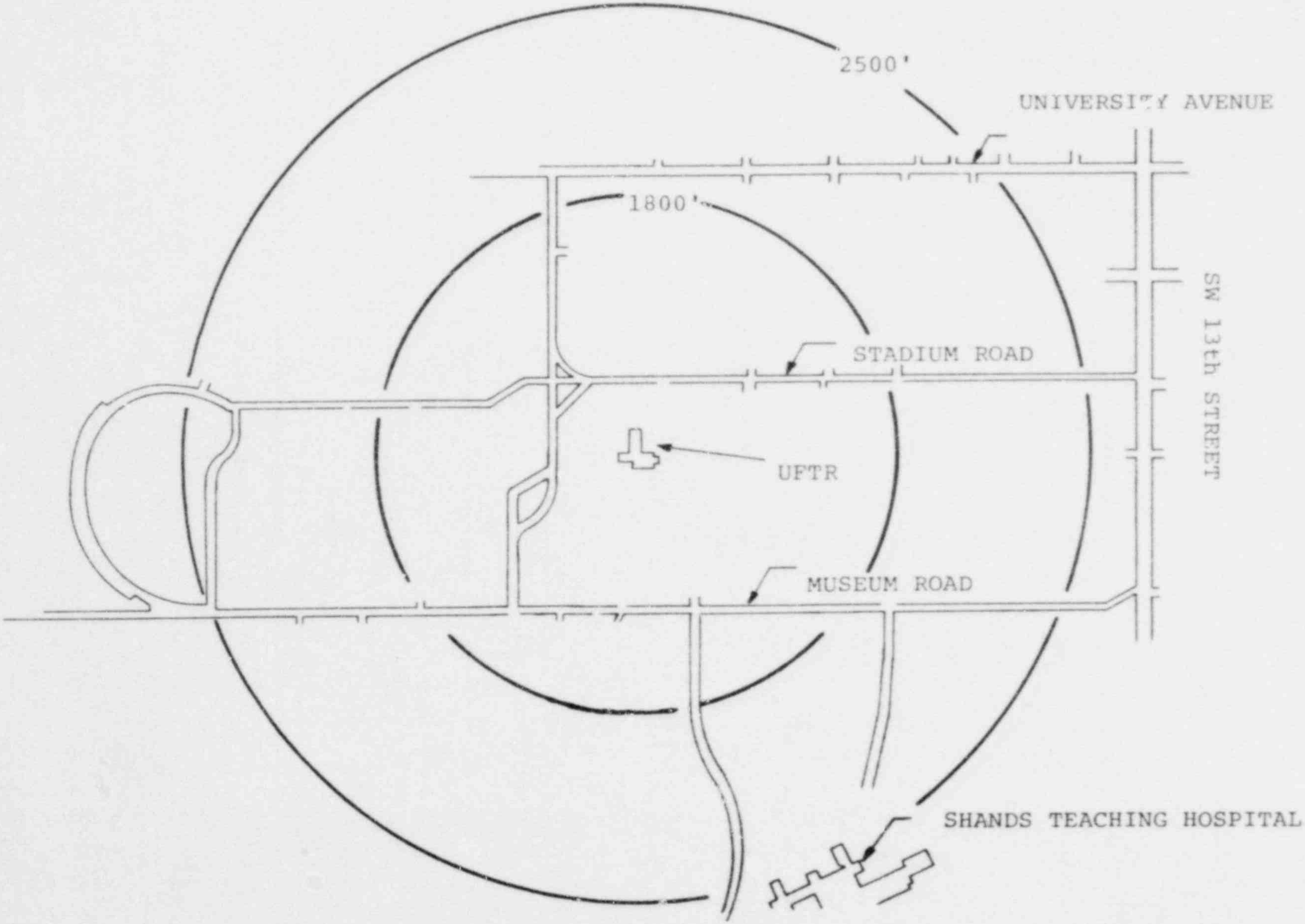


Figure 2-22. UFTR Environs Showing Distance to the Urban Boundary.

Table 2-14

DESIGN BASIS ACCIDENT DIFFUSION COEFFICIENTS WITH SITE SPECIFIC METEOROLOGY

VENT DISTANCE (miles)	PERIOD	DURATION	WORST SECTOR	DIFFUSION COEFFICIENT (sec/m^3)
0.1	0-2 hours	2 hours	S	1.3 E-03
0.1	2-24 hours	22 hours	S	5.1 E-04
0.1	1-4 days	3 days	ESE	3.4 E-04
0.1	4-30 days	26 days	ESE	1.6 E-04
0.2	0-2 hours	2 hours	NNE	4.2 E-04
0.2	2-24 hours	22 hours	NNE	1.6 E-04
0.2	1-4 days	3 days	S	1.0 E-04
0.2	4-30 days	26 days	ESE	5.2 E-05
0.3	0-2 hours	2 hours	WSW	1.9 E-04
0.3	2-24 hours	22 hours	S	7.8 E-05
0.3	1-4 days	3 days	S	5.0 E-05
0.3	4-30 days	26 days	ESE	2.6 E-05
0.4	0-2 hours	2 hours	S	1.2 E-04
0.4	2-24 hours	22 hours	S	4.9 E-05
0.4	1-4 days	3 days	ESE	3.2 E-05
0.4	4-30 days	26 days	ESE	1.6 E-05
0.5	0-2 hours	2 hours	S	8.4 E-05
0.5	2-24 hours	22 hours	S	3.4 E-05
0.5	1-3 days	3 days	ESE	2.2 E-05
0.5	4-30 days	26 days	ESE	1.1 E-05

Table 2-15

DESIGN BASIS ACCIDENT DIFFUSION COEFFICIENTS WITH NRC STANDARD METEOROLOGY

DISTANCE (miles)	DIFFUSION COEFFICIENTS (sec/m^3) 0-8 hours	8-24 hours	1-4 days	4-30 days
0.1	1.0 E-02	3.0 E-03	1.3 E-03	3.5 E-04
0.2	4.5 E-03	1.0 E-03	5.6 E-04	8.5 E-05
0.3	2.2 E-03	6.4 E-04	2.7 E-04	4.4 E-05
0.4	1.4 E-03	4.0 E-04	1.0 E-04	2.5 E-05
0.5	8.0 E-04	2.6 E-04	7.0 E-05	1.6 E-05

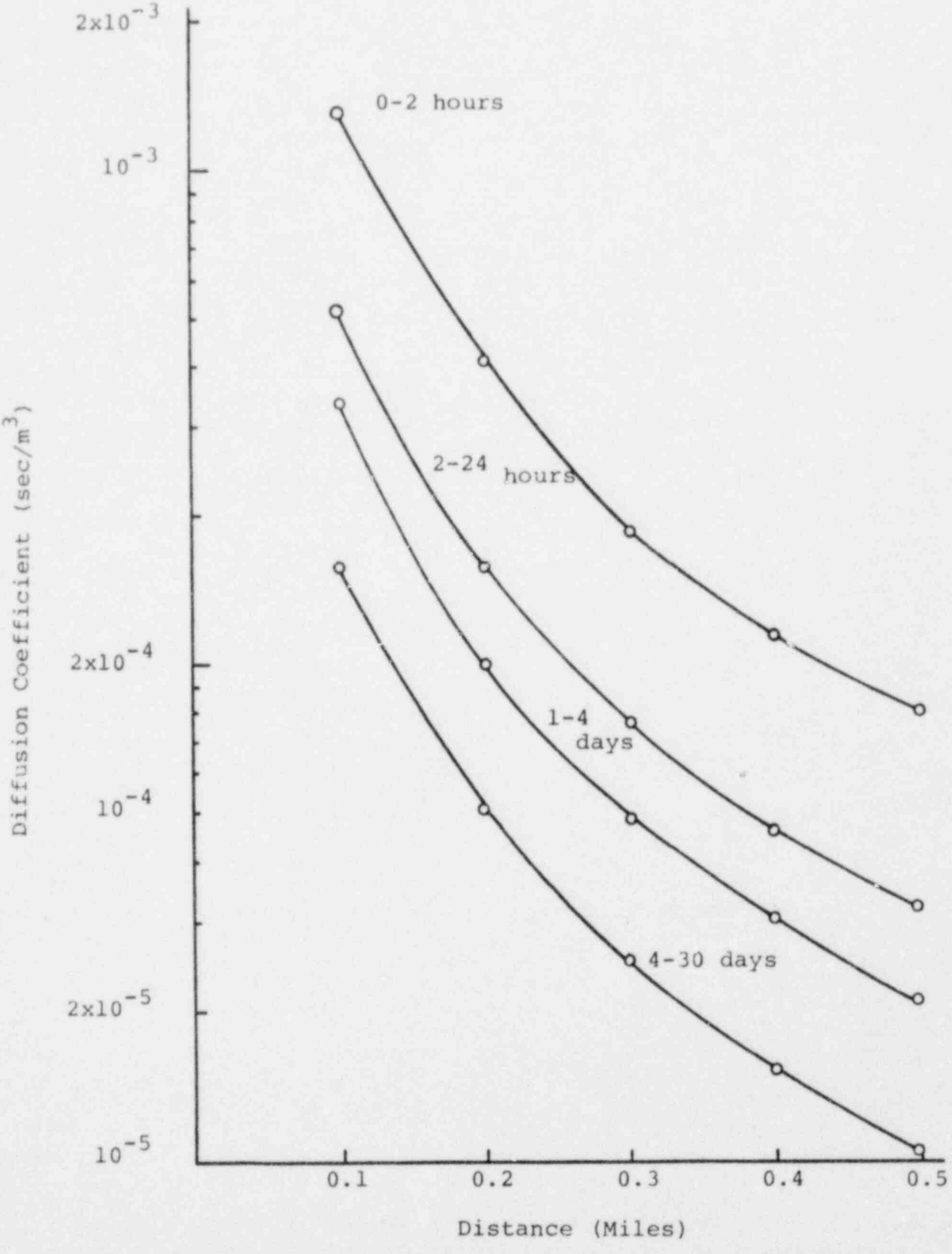


Figure 2-23. Design Basis Accident Diffusion Coefficients with Site-Specific Meteorology for Several Short-Term Time Periods.

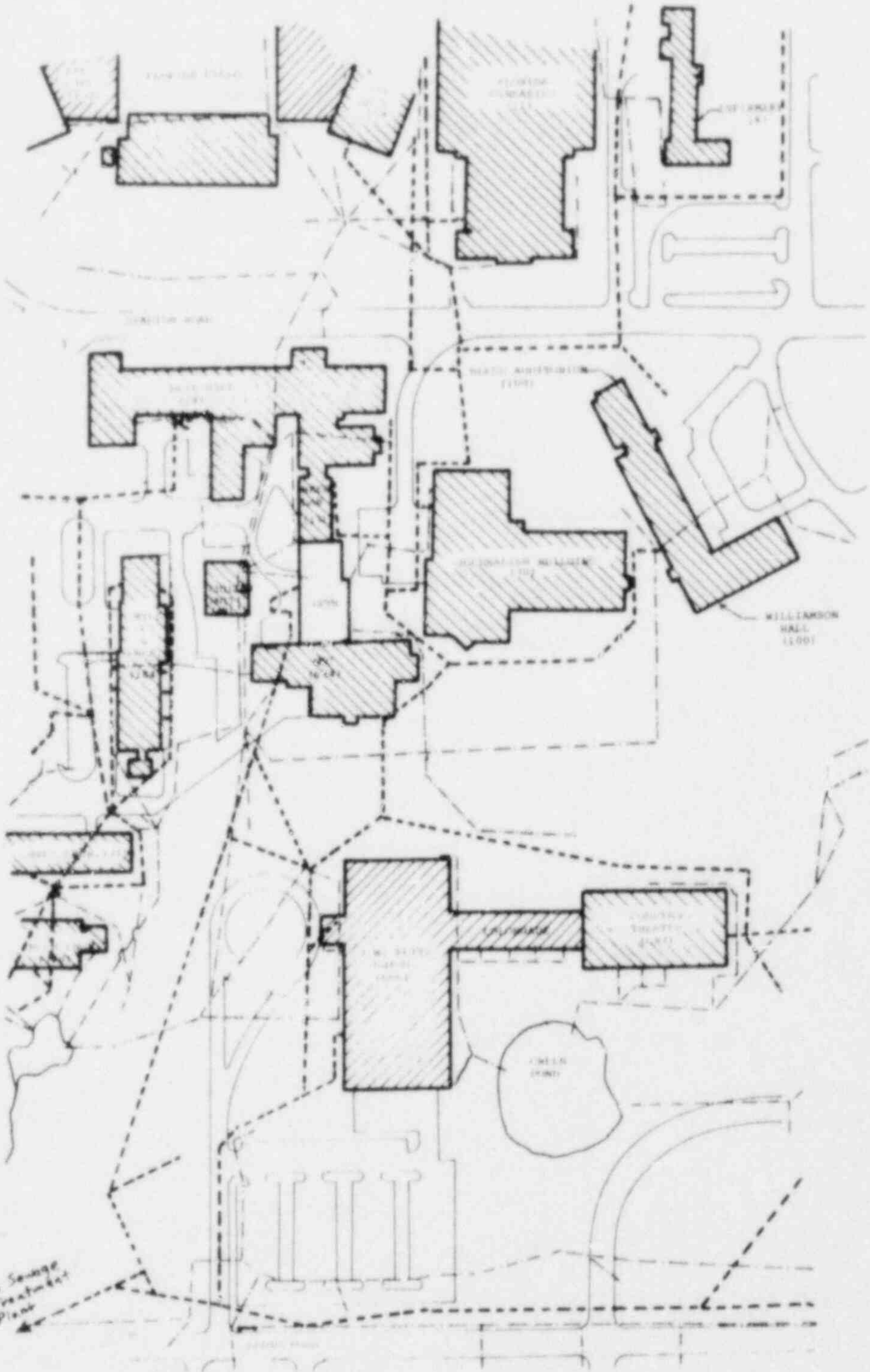
The site selected for the reactor rises to the east. At the base of the rise on the west is a small valley running south and terminating in the vicinity of two sinkholes. Although the valley is mostly landscaped grass and driveways, the basic land features are still present today. Thus, the surface drainage of the site would be to the west and then south to these sinkholes as shown in Figure 2-14. The surface water enters the underground aquifer through these sinks. Therefore, it is anticipated that no meteorological extremes that will cause blockage of tne current ingress or egress features will ever be possible.

2.4.1.2 Hydrosphere. The University of Florida is located in the southwestern quadrant of the greater Gainesville area. Gainesville is in the Central Highlands of Florida in the nortern portion of the Florida peninsula. The nearest approach of the Gulf of Mexico is about 50 miles to the southwest. The Atlantic Ocean is about 65 miles to the east.

Figure 2-14, a generalized topographic map of Alachua County, shows that there are three (3) watersheds in the county. The largest watershed which drains the Gainesville, Micanopy, Archer and Newberry area is believed to contribute surface water through sinkholes and solution caverns in the limestone bedrock to the underground aquifer which eventually feeds Wacasassa River, which flows into the Gulf of Mexico near Cedar Key to the west of Gainesville.

While the storm sewer system of the city of Gainesville is not indicated, it would follow much the same pattern as the existing and proposed sewage lines for the University of Florida shown in Figure 2-24. In general, there are two natural drainage zones for the greater Gainesville area. The dividing line between these zones follows very closely the line formerly occupied by the Seaboard Railway roadbed running diagonally from the southwest to the northeast. With the exception of a small portion in the northeast corner of gainesville, the area to the north and west of the former railroad bed, containing approximately 31.5 square miles, drains toward Hogtown Creek and its tributaries which flow into Lake Kanapaha located in the southwest corner of greater Gainesville. The drainage pattern of the zone laying south and east of the railway is not as clearly defined as the northwest zone, but, in general, is east and south. Water falling on the eastern portion drains eventually into Newnan's Lake and water falling in the southern portion drains into Sweetwater Creek, Biven's Arm and Payne's Prairie which are shown in Figure 2-24. Figure 2-25 shows qualitatively the average volume flow of surface streams in Florida. Since Gainesville is at the headwaters of the St. Johns, Suwannee and Wacassassa River Systems, it has a very small average surface stream flow. There are no surface streams of any consequence in the Gainesville area. During the dry season, which is generally March, April and May, the surface flow of the creeks in the area decreases to nearly zero although there is still a small subsurface flow. The water table is close to the surface and the movement of the ground water is very rapid because of the high porosity and permeability of sandy soil and cavernous limestone bedrock.

The city of Gainesville and vicinity receives its water supply from the municipal water treatment plant. All of the water entering the treatment plant is obtained from seven wells ranging from 367 to 750 ft. in depth. Spring or surface water is not used for the municipal supply but several springs supply water for agriculture and industry.



---- Sanitary Sewage   -·-·- Storm Sewage

Figure 2-24. Current Updated Sanitary and Storm Sewage Systems on University of Florida Campus Around UFTR Site.

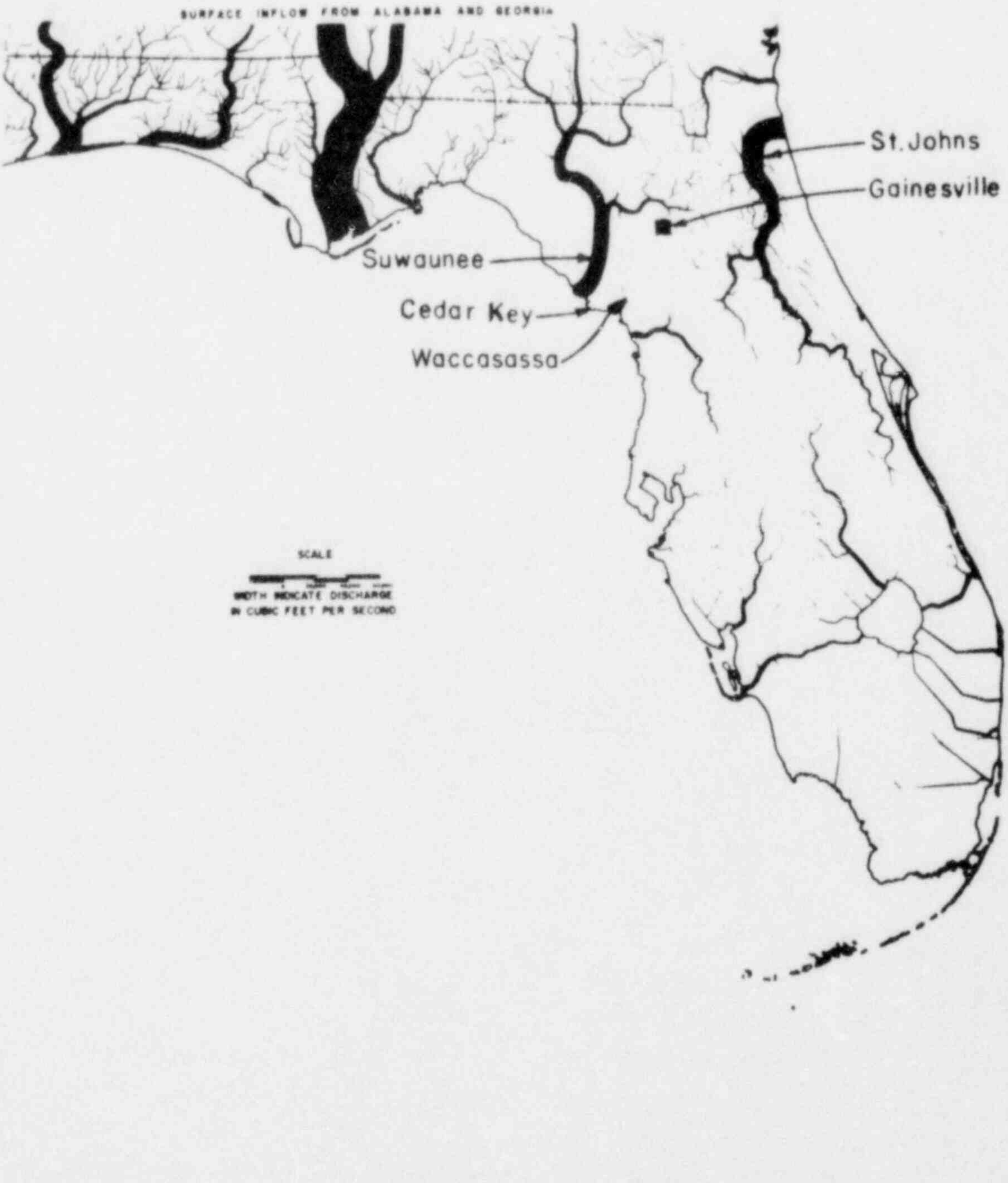


Figure 2-25. Average Flow of Surface Streams in Florida.

The interrelationship between rainfall, evapotranspiration, deficit and percolate is known as the agrohydrologic balance. Figures 2-26A and 2-26B show this interrelationship for Gainesville during 1953 and 1954. From this relationship it can be observed that the amount of water percolating through the surface soil varies from year to year in a complex manner. The amount of percolating water will determine the soil water dilution factor in the event of accidental release of radioactivity from the UF Training Reactor. It should be noted that the amount of percolating water in Gainesville is always relatively small and often there are months when it drops to zero, generally in the spring and Summer. (2)

2.4.2 Floods

2.4.2.1 Flood History. Exhaustive studies have indicated no record of any major flood in the general UFTR site area during the past 100 years.

2.4.2.2 Flood Design Considerations. Because of its inland position which removes the potential for tidal flooding and because of the well-drained location of the UFTR site, no special consideration is given to floods in the UFTR design. At any rate, the self-contained design of the UFTR makes it more resistant to any hypothetical flood condition. At any rate, emergency flood procedures are considered in the UFTR Standard Operating Procedures so no further consideration is necessary here (3).

2.4.2.3 Effects of Local Intense Precipitation. As discussed earlier in Section 2.4.2.2, the location of the reactor site in reference to the drainage system,including the University of Florida storm sewage system provides sufficient drainage to all runoff water likely to occur due to rain: therefore, it is virtually impossible for local precipitation, (at most 9.93 inches in one day, see Table 2-8) ever to affect the reactor building.

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers.

Since the UFTR is an essentially self-contained reactor design requiring minimal cooling by an ultimate heat sink, since no major streams or rivers run near the site area, and since the location itself is well-drained, it is felt that the PMF on streams and rivers in North Central Florida has no potential effects on the UFTR facility and its operation. For these same reasons, probable maximum precipitation, precipitation losses, runoff and stream course models, probable maximum flood flow, water level determinations due to PMF and coincident wind wave activity need not be considered further.

2.4.4 Potential Dam Failures, Seismically Induced

There are no dams in the University of Florida - Gainesville area which could affect the reactor site in case of failure. Therefore, dam failures and attendant water levels and effects need not be considered further.

2.4.5 Probable Maximum Surge and Seiche Flooding

Because of the UFTR site location, there are no surface bodies of water close enough to affect the UFTR site via seiche flooding or surges of any kind.

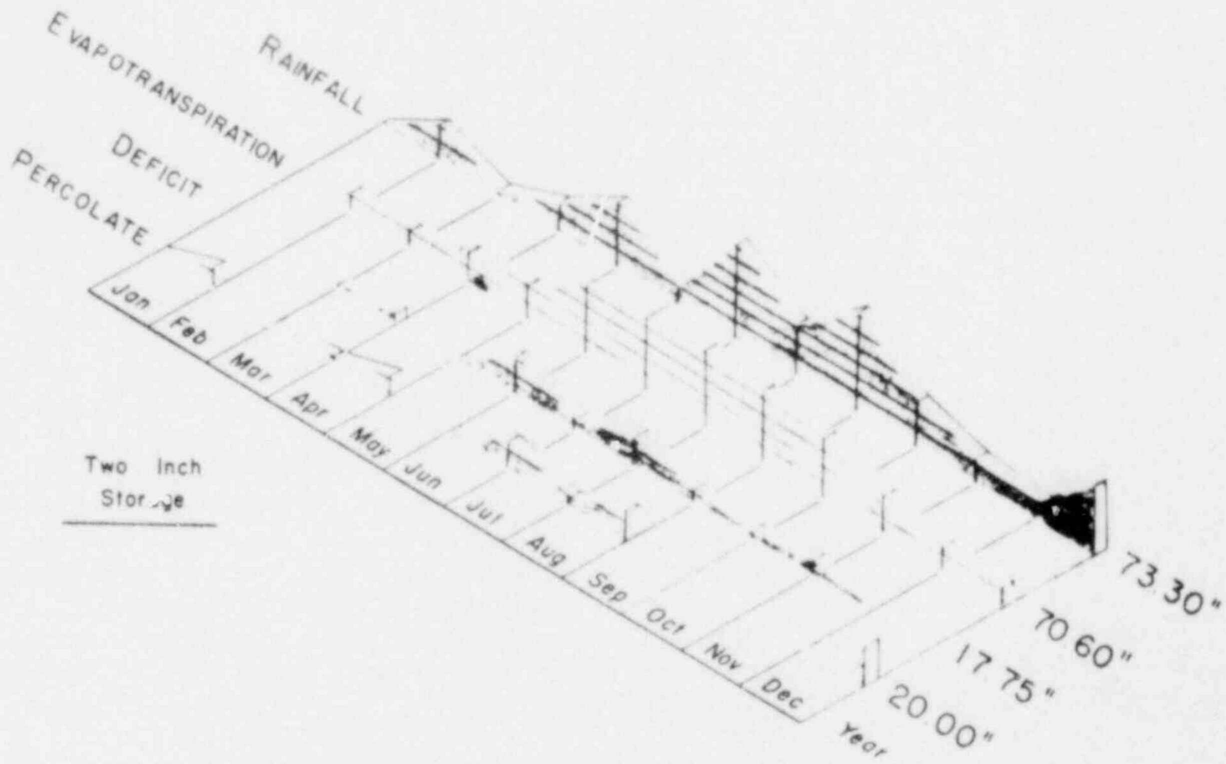


Figure 2-26A. Agrohydrologic Balance, for Gainesville, Florida, 1953.

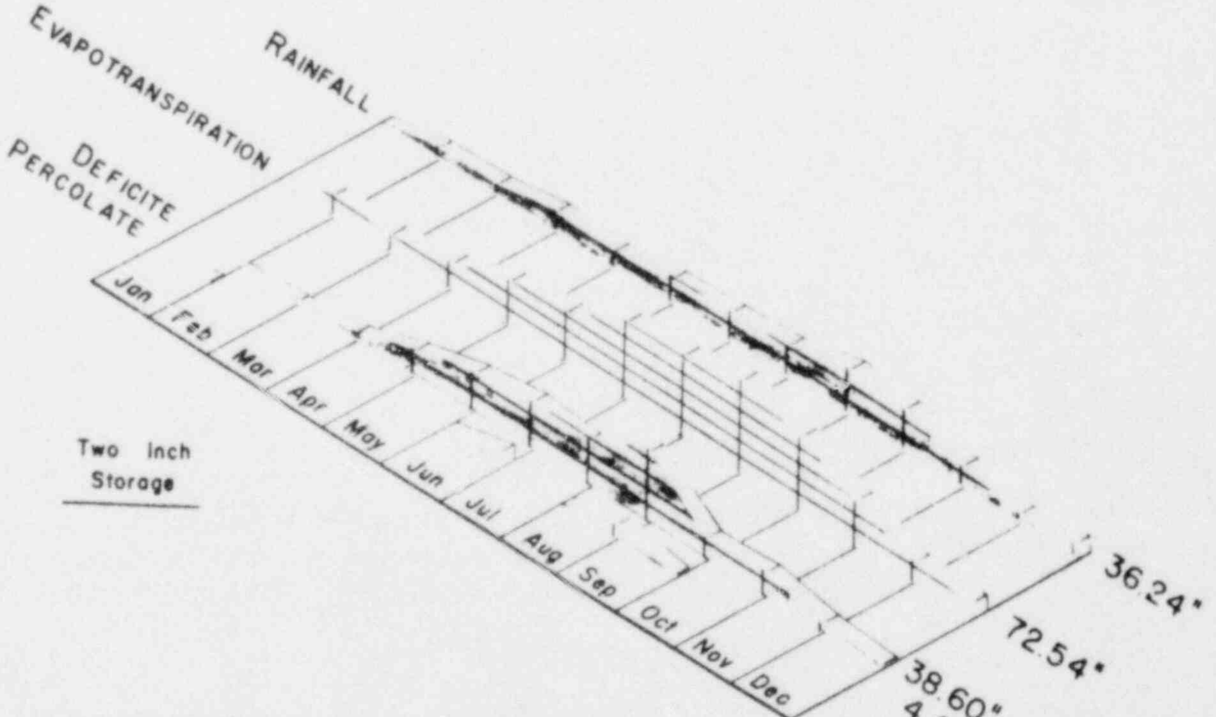


Figure 2-26B. Agrohydrologic Balance, for Gainesville, Florida, 1954

2.4.6 Probable Maximum Tsunami Flooding

Due to its inland location, approximately 50 miles from the Gulf of Mexico, tsunami flooding is predicted to have no effect on the UFTR site. Only one tsunami, or seismic sea wave, has ever been noted along the Gulf Coast of the United States. This wave was caused by the Puerto Rican earthquake of October 11, 1918 and was very small as recorded on the tide gauge of Galveston, Texas.

2.4.7 Ice Effects

Since the site has no surface water bodies on it and since the climate makes the formation of significant amounts of ice extremely unlikely, there are no ice effects to be considered for the UFTR site from ice jam floods, wind-drive ice ridges or other ice-produced effects and forces which could affect safety-related UFTR facilities.

2.4.8 Cooling Water Canals and Reservoirs

Due to low power UFTR operation, there are no cooling water canals or reservoirs, so this section is not required - not applicable.

2.4.9 Channel Diversions

This section is also not applicable to the UFTR facility.

2.4.10 Flooding Protection Requirements

The self-protected, self-controlling design of the UFTR along with its location in a flood-free area make additional flood protection requirements unnecessary.

2.4.11 Low Water Considerations

This section is also not applicable because low water levels and flow rates have no effect on the UFTR facility. The system can withstand a complete loss of cooling water even at full power.

2.4.12 Dispersion, Dilution and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters

This section is not applicable to the UFTR facility because there are no surface waters on the UFTR site to disperse, dilute or concentrate liquid releases of radioactive effluents.

2.4.13 Groundwater

Groundwater information for the UFTR site and environs is contained in Section 2.5.4.

2.4.14 Technical Specifications and Emergency Operations Requirements

Detailed procedures designed to minimize the impact of floods, and protective measures to be taken in case of floods are outlined in the UFTR SOP B.4 - Emergency Flood Procedures.

2.5 Geology, Seismology and Geothechnical Engineering

2.5.1 Basic Geology and Seismic Information

2.5.1.1 Regional Geology. The regional geology of the Gainesville area is represented by the Florida Geological Survey data found in Figure 2-27. Cross Section B-3 of Figure 2-27 shows the general geology of the Gainesville area. The solid bedrock in this area is porous and cavernous Ocala limestone which occurs in a broad truncated dome with its crest in Levy County southwest of Gainesville. The Ocala formation is overlain by other porous limestones and semipermeable sandy clays (Hawthorne formation). This is capped by loose surface sands. In general, all the formations are quite porous and permeable. Locally, however, the Hawthorne sandy clays confine the ground water in the underlying porous limestones under artisian pressure (2). Because of the porous nature of these formations and their relation to the hydrologic description of the region, some information on the geological description of this area has been included in Section 2.4.1.2.

2.5.1.2 Site Geology. The specific site geology is very similar to that of the region as a whole. The physical and chemical properties of the soil, sub-soil and bedrock are such that negligable radioactive decontamination or absoprtion can be expected.

Studies have shown that the soils are sandy and possess very little ion-exchange capacity. The calcium carbonate (limestone) bedrock has virtually no capacity for preventing the rapid movement of radioactive products toward the ground water table. It would only react chemically to neutralize acid solutions and precipitate insoluable carbonates. It has virtually no ion-exchange capacity and is highly porous and permeable so that any chemical precipitates formed would only slightly retard the flow of radioactivie liquids through the bedrock.

Most of the Gainesville area and that part of the campus of the University of Florida north of Radio Road, including the UFTR site, is underlain by a loamy fine-sand type of soil. This was derived from residual Hawthorne formation and is characterized by a typical slope of 2 to 7 percent, light brown or brownish grey surface soil, light yellowish brown or pale brown subsoil, nearly loose to loose with good natural drainage.(2) The soil data for all the test borings undertaken on the site are summarized in Tables 2-16A through 2-16D. Additional test boring data was obtained as a result of construction of the 6 inch water well which is the source of the secondary water supply of the UFTR cooling system. The following data is available as a result of test borings: Limestone: 75' depth, Water Table: 89' depth.

Florida is a relatively inactive area for seismic activity. Due to its compact size and few auxiliary systems, the UFTR is much less susceptible to earth movement problems than large power reactors or facilities with systems spread over larger areas. There is no effect on the system due to geological conditions affecting other situations on the University of Florida campus. Earthquakes are not a serious threat but data on their occurrence and other possible effects are presented in Sections 2.5.2 to 2.5.6.

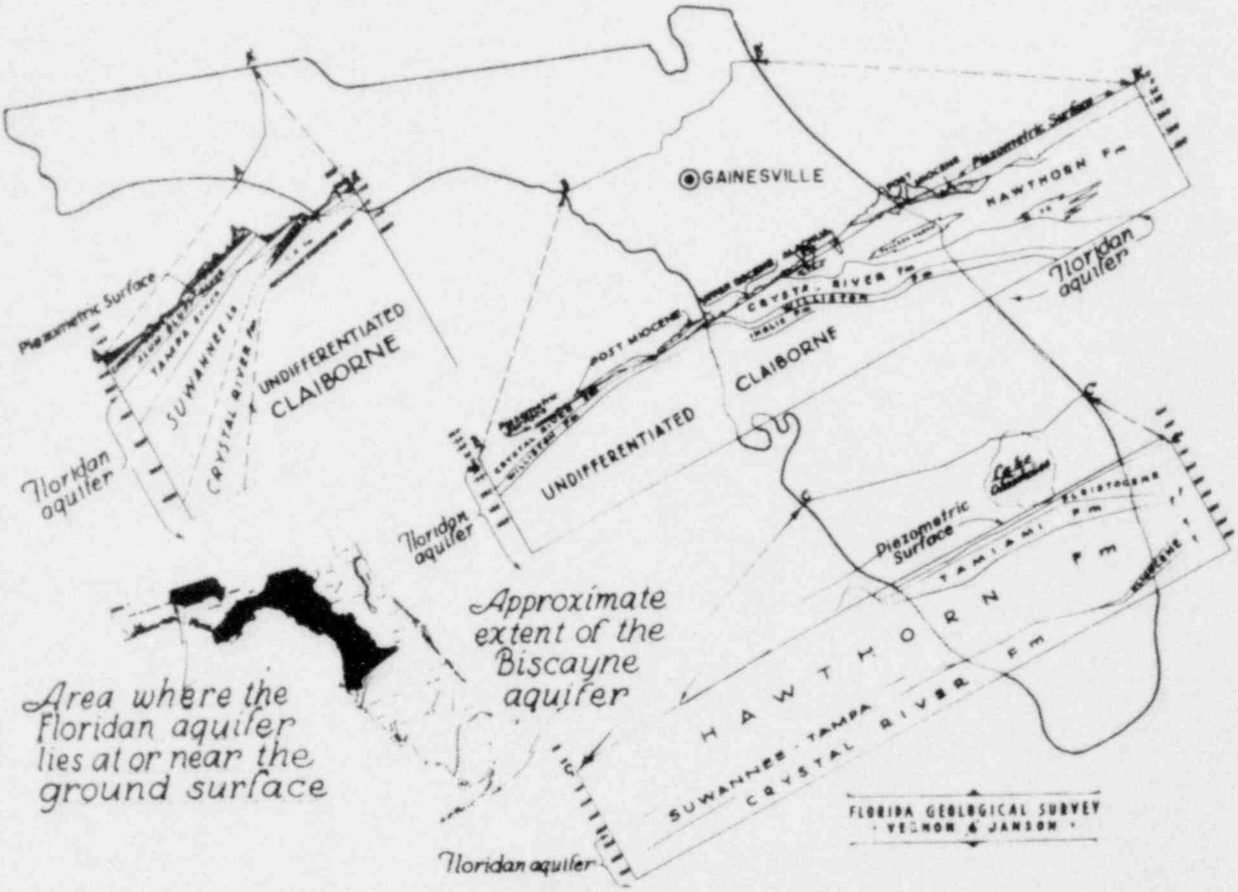


Figure 2-27. Florida Geological Survey Data for North and Central Florida Area Including Gainesville and Alachua County.

POOR ORIGINAL

Table 2-16A

TEST BORING DATA FOR THE UFTR

Boring Record - Project 5111 - EIES

Hole No.....1.................................Sheet...........of.........Sheets

Location....Between Reid Lab and E&I Building.................................

Started.......................................Completed...5/10/57..............

Ground Water Depth....5'...

Hammer Wt.....300 lbs..................Drop....18"..............Sampler Size 2-1/2.

Depth	Number	Blows	Description
5	1	4	Medium soft grey and brown sandy clay
10	2	13	Stiff greyish sandy clay
15	3	28	Stiff blue and grey sandy clay
20	4	57	Stiff tan sandy clay with rock frag.
25	5	100&Core	Stiff blue rocky clay
30	6	Core	Stiff blue clay with sandy layer
35	7	45	Stiff blue rocky clay
40	8	Core	Stiff blue rocky clay
45	9	63	Stiff blue rocky clay

Bottom of Hole 45' No Cavity

Table 2-16B

Hole No...2...............................Sheet...........of.........Sheets

Location...Between Reid Lab and E&I Building.......................................

Started..............................Completed..5/10/57.................

Ground Water Depth...5'...

Hammer Wt....300 lbs....................Drop...18"...............Sampler Size 2-1/2.

Depth	Number	Blows	Description
5	1	3	Soft greyish sandy clay and Phosphate
10	2	8	Medium greyish sandy clay
15	3	22	Stiff light blue and grey sandy clay
20	4	35	Stiff sandy clay and Phosphate
25	5	57	Stiff grey sandy clay with rock frag.
30	6	47	Stiff blue and grey sandy clay
35	7	Core	Stiff blue rocky clay

Bottom Hole 37' No Cavity

Table 2-16C

Hole No...3...Sheet...of...Sheets

Location...Between Reid Lab and E&I Building...

Started...Completed...5/10/57...

Ground Water Depth...7...

Hammer Wt...300 lbs...Drop...Sampler Size...2-1/2

Depth	Number	Blows	Description
5	1	8	Medium brown sandy clay
10	2	13	Stiff brown sandy clay
15	3	7	Soft grey sandy clay with pebble
20	4	7	Soft brown and grey sandy silty clay
25	5	20	Stiff grey sandy clay with phosphate
30	6	32	Stiff grey sandy clay with phosphate
35	7		Stiff grey sandy rocky clay
40	8	Core	Stiff grey sandy rocky clay

Bottom of Hole 40' No Cavity

Table 2-16D

Hole No....4....Sheet....of....Sheets

Location....Between Reid Lab and E&I Building....

Started....Completed..5/13/57....

Ground Water Depth....7....

Hammer Wt....300 lbs....Drop....18"....Sampler Size..2-1/2..

Depth	Number	Blows	Description
5	1	8	Medium grey sandy clay with pebble
10	2	6	Medium brown and grey sandy clay and peb.
15	3	27	Stiff greyish sandy clay
20	4	25	Stiff greyish sandy clay
25	5	37	Medium tan silty sandy clay and rock frag.
30	6	20	Stiff grey sandy clay with rock frag.
35	7	15	Stiff blue rocky clay

Bottom of Hole 37' No Cavity

2.5.2 Vibratory Ground Motion

As reported in Reference 11, seismic analyses to obtain response spectra were conducted by Weston Geophysical Research, Inc. for Florida Power Corporation's Crystal River Site. The Reverend Daniel Linehan, S.J., Director of Weston Observatory, acted as a consultant on the seismic analysis. The response spectra were completed by Dr. C. Allen Cornell, Department of Civil Engineering, Massachusetts Institute of Technology. Although these data are presented for the Crystal River site, they are very similar to and can relate directly to the UFTR site because the soil strata conditions are similar and all of Central Florida has a seismically stable history, relatively free of earthquakes.

The State of Florida is an area which is considered seismically inactive; there is no record of a severe earthquake in Florida. There is ample evidence that Florida has been remarkably stable and free of earthquakes for about one million years, and is considered to be one of the most stable areas in the United States. Only eight (8) earthquakes of Intensity IV (Modified Mercalli Scale) or greater have had their epicenter within 50 miles of the Crystal River plant site. Only one tsunami, or seismic sea wave, has ever been noted along the Gulf Coast of the United States. This wave was caused by the Puerto Rican earthquake of October 11, 1918, and was very small as recorded on the tide guage of Galveston, Texas. There is no record of a tsunami or seismic sea wave ever having affected the Crystal River area. It is highly unlikely that, if a tsunami did occur, it would exert its effects inland as far as Gainesville, Florida, which is over 50 miles inland.

The two strongest earthquakes to have affected the site area in north central Florida were the northern Florida earthquake of January 12, 1879, which was listed as Modified Mercalli IV, and the Charleston, South Carolina earthquake of 1885 which had an epicentral Intensity X, Modified Mercalli. There is no evidence that seismic activity in the southern appalachians or in the greater Antilles Islands of the West Indies had any effect on the Crystal River site, and consequently the UFTR site.

An attenuation curve of earthquake intensity with distance is shown in Figure 2-28 for the Atlantic and Gulf Plains indicates a rather slow attenuation of intensity with distance, due apparently to the deep Cutaceous sediment areas of the Coastal Plain Regions. Based upon this attenuation information, the Florida earthquake of 1879 would have had an intensity no higher than V at the Crystal river site.

Based upon the relationship between earthquake intensity and ground acceleration given in Nuclear Reactors and Earthquakes, TID-7024, U.S. Atomic Energy Commission, the Charleston, South Carolina earthquake would have resulted in a ground acceleration of about .025g at the UFTR site. Based on this data and previous historical data, no special consideration was given in the design of the reactor building beyond making it a "vault-type" building as defined in 10 CFR 73.2(o).

2.5.3 Surface Faulting

There is ample evidence that Florida has been stable and free of earthquakes for about one million years, and it is considered to be one of

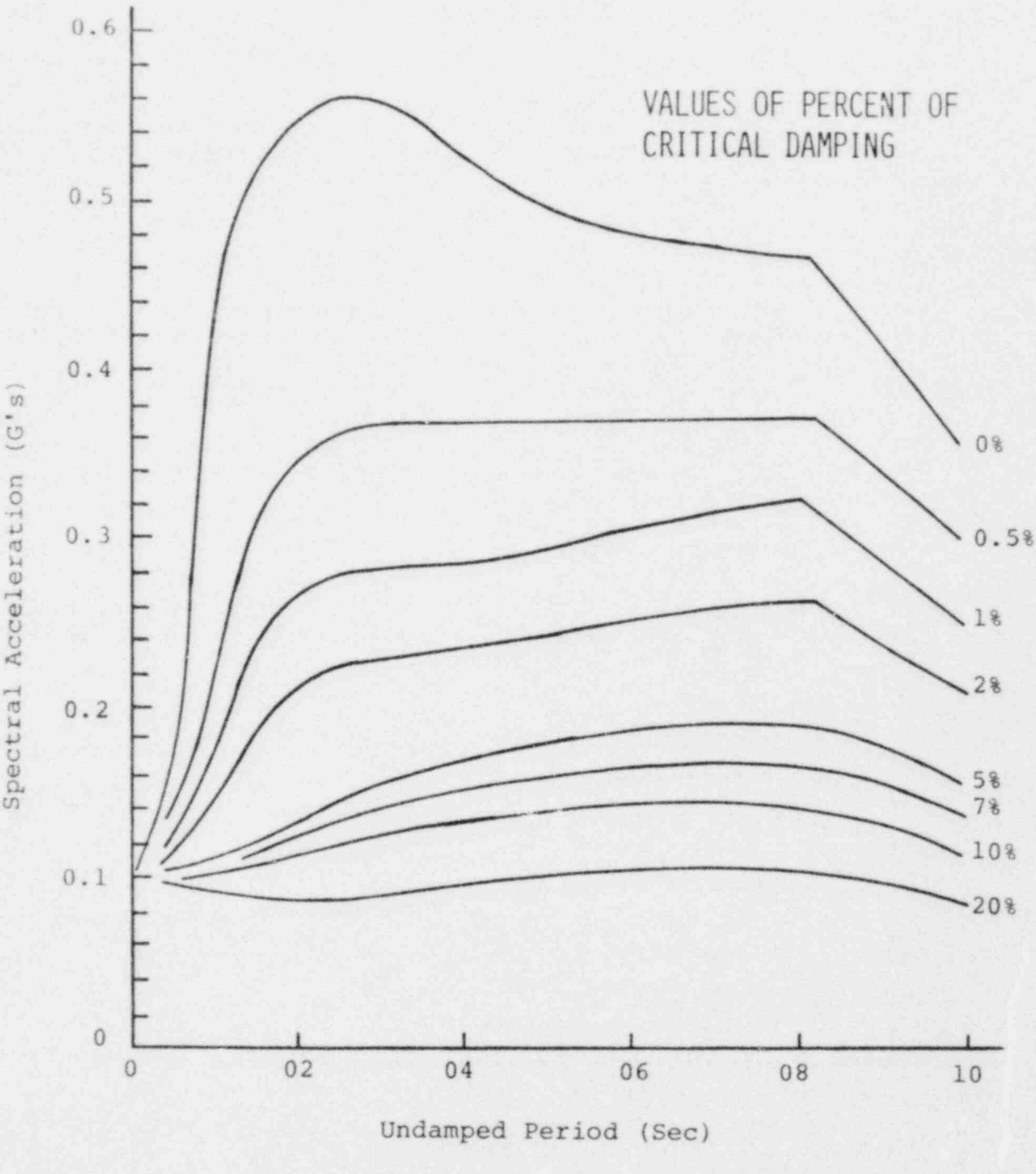


Figure 2-28. Acceleration Spectra (Maximum Hypotheti.al).

the most stable areas in the entire United States. (11) There have, however, been several small earth tremors which has caused slight damage such as small cracks in plaster wall in some areas of the state(2).

2.5.4 Stability of Subsurface Materials and Foundations

The information defining the conditions of the strata supporting the reactor building foundations was included in Section 2.5.1.2 - Site Geology along with the test records and summaries of soil strata compositions. The limerock formations are very stable geologically as indicated by the relative absence of earth movement activity in Florida over the past million years.

2.5.5 Stability of Slopes

There are no rock or soil slopes of concern for the UFTR site. The general incline toward the west and south eliminates the possibility of drainage or flooding problems. The test boring data in Section 2.5.1.2 and the general site and area topography have shown that this area is very stable. There is no danger of landslides since the general slope of the land is a gradual incline with no sharp contours. The test borings also indicate there is no concern with sinkholes affecting the topography of the UFTR site.

2.5.6 Embankments and Dams

This section does not apply to the UFTR site since these facilities are not needed for the UFTR facility and are not present in the UFTR site.

APPENDIX 2A

ORIGINAL UFTR METEOROLOGICAL DATA

(Reference 2)

A. ORIGINAL DETERMINATION OF UFTR WIND ROSE DATA

2A.1 Wind Direction and Velocity

Due to the lack of available local data regarding atmospheric stability, wind direction and velocity, and the relationship of precipitation and wind, a program was started to collect these micrometeorological data. The information as reported in Reference 2 is presented here for completeness.

A Bendix-Friez aerovane was installed on the radar tower at the University of Florida in October, 1956. The instrument is located approximately 125 ft. above ground level (272 ft. above mean sea level) in an area reasonably free of disturbing structures about 1500 ft. from the reactor site. From the latter part of October, 1956, wind direction and velocity were recorded continuously for this station. A second aerovane was installed early in May, 1957, on the same tower about 30 ft. above ground level (177 ft. above mean sea level). Since the elevation of the reactor stack outlet 164 ft. above mean sea level, data taken at this second station should be fairly representative of the undisturbed conditions at the points of gas discharge.

Figure 2A-1 gives an annual comparison of the wind data at the two elevations for the year from July, 1957, through June, 1958. More detailed data are presented as monthly wind roses in Figures 2A-2 and 2A-3. Figure 2A-2 covers the period January, 1957, through June, 1957, for the upper and lower elevations for the period July, 1957, through June, 1958.

In constructing the wind roses, five air velocity groups were used--cal - 1,2-4, 5-7, 8-12 and 13+ miles per hour. Winds of velocity greater than 13 m.p.h. occurred so seldom and for such short duration that it was considered unnecessary to indicate separate groups above 13 m.p.h. The greatest hourly movement of winds recorded during each month and the time of occurrence are given in Table 2A-1.

A wind direction and speed frequency distribution is given in Table 2A-2 for the upper and lower stations for this period, June, 1957, through May, 1958. The prevailing winds at the upper station fall in the range of 5 - 12 m.p.h., while those at the lower station fall in the range of calm to 4 m.p.h. Winds at both elevations show a slight preference for the quadrant from NE to SE.

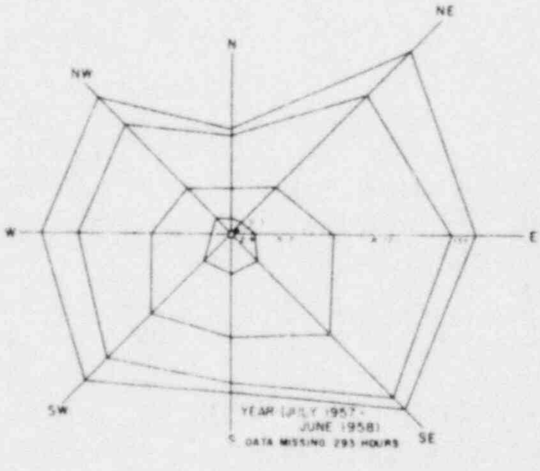
The persistance of wind direction at 30-ft. level for the period June, 1957 through May, 1958, is indicated in Table 2A-3.

2A.2 Precipitation

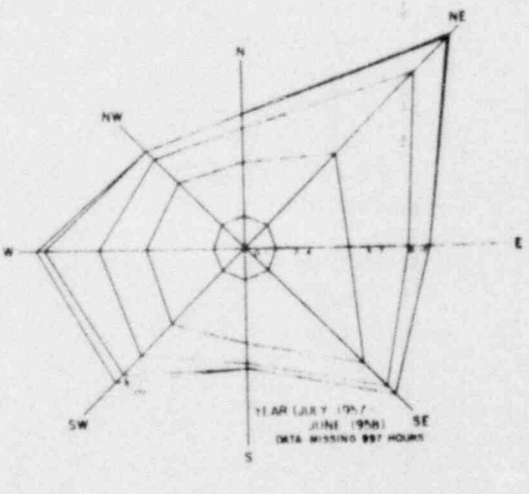
An automatic rain gage is located on the University Campus. The data from this station are available to this project through the U.S. Weather Bureau but the hourly precipitation data have been distributed only through the month of December, 1957, at the present time. The daily rainfall for 1957 at Gainesville, Florida is shown in Table 2A-4.

ORIGINAL UFTR WIND AND PRECIPITATION DATA

The following three figures, 2A-1, 2A-2, and 2A-3 summarize the wind and precipitation data for the University of Florida as wind roses for the period January, 1957 through June, 1958. Wind data were obtained from aerovanes located on the College of Engineering radar tower at elevations of 125 feet and 30 feet above ground. The wind data were divided into five velocity groups, calm-1, 2-4, 5-7, 8-12, and 13+ miles per hour. The radial length of direction lines represented by the windscale indicates the number of hours for which winds of the designated velocity group prevailed from the point indicated. Shaded areas represent the number of hours in each velocity range during which precipitation occurred.



UPPER ELEVATION



LOWER ELEVATION

Figure 2A-1. Original UFTR Annual Summary of Wind Data Showing Monthly Totals Averaged Over the Year, Reference 2.

POOR ORIGINAL

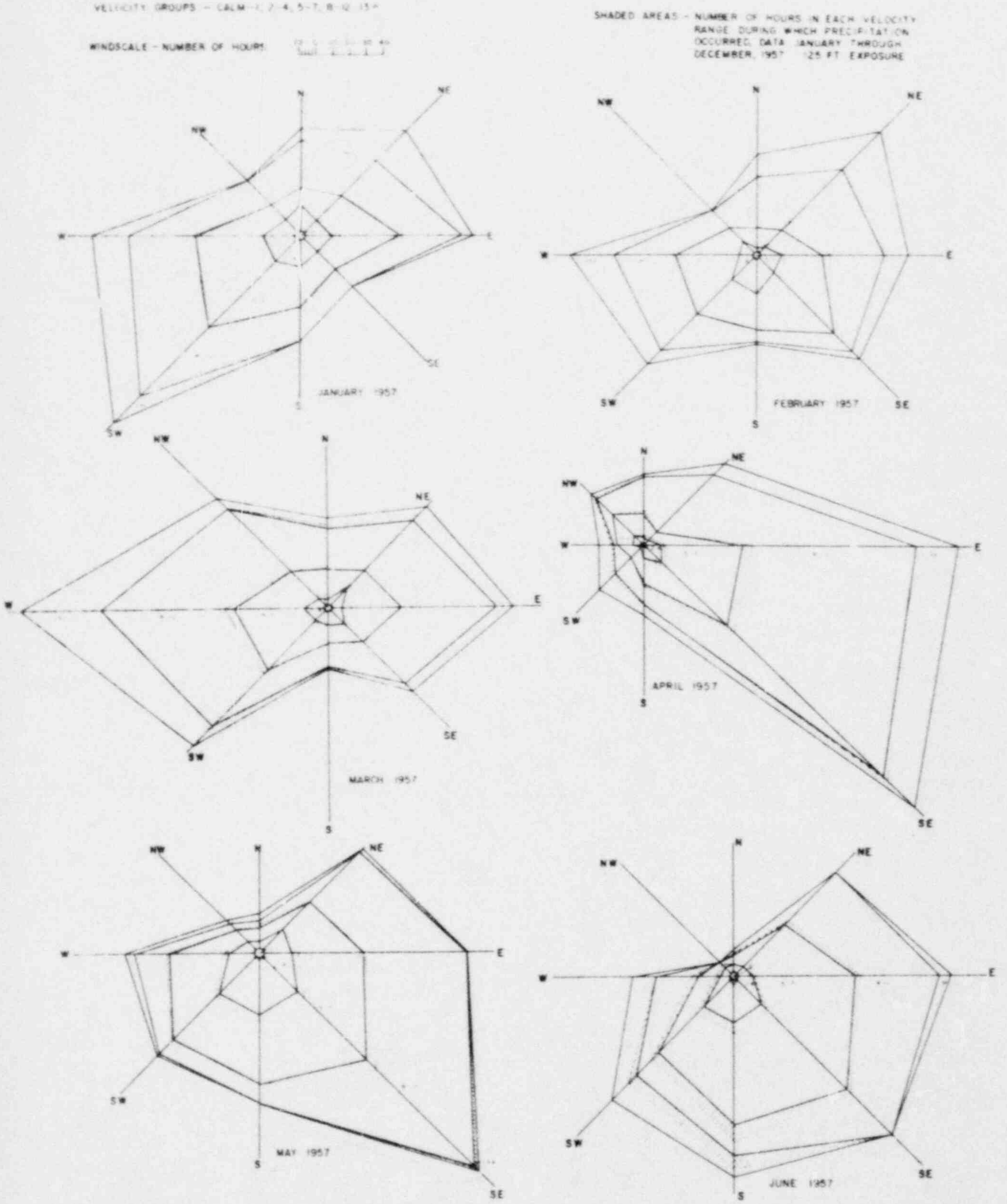


Figure 2A-2. Monthly Wind Roses, January-June, 1957 (Upper Elevation Only).

UPPER ELEVATION LOWER ELEVATION

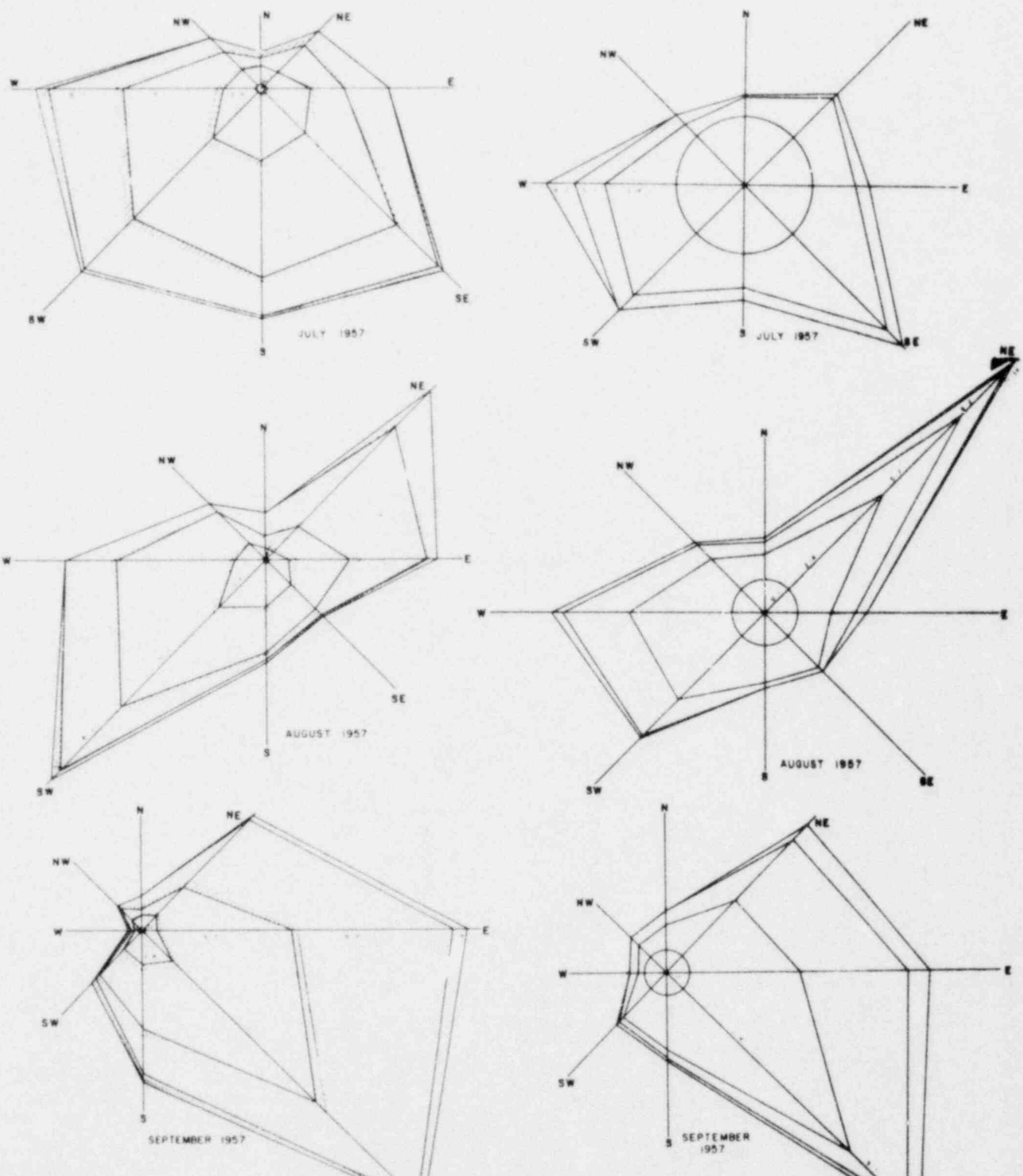


Figure 2A-3. Comparison of Upper and Lower Elevation Monthly Wind Roses, July, 1957-June, 1958.

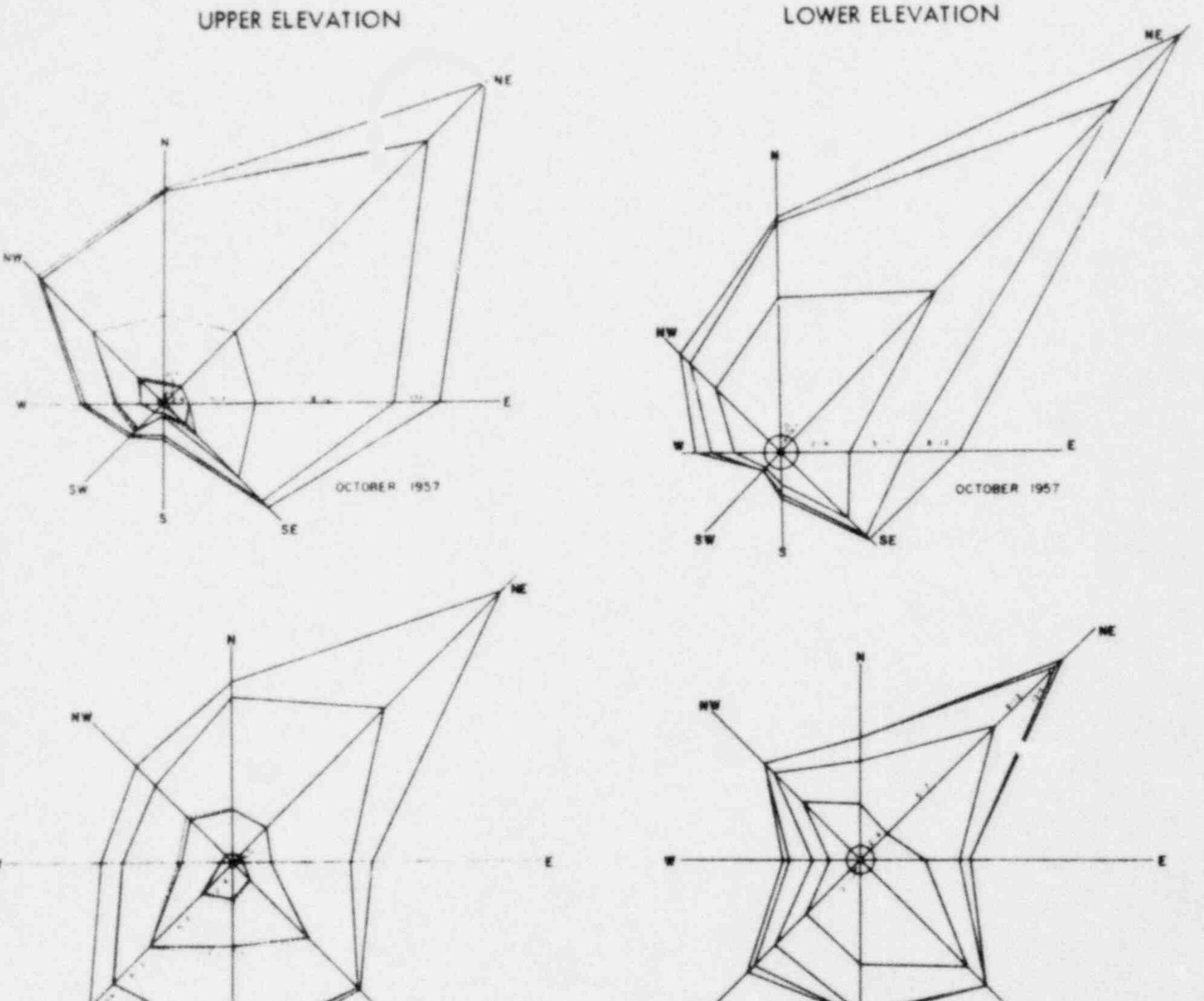


Figure 2A-3. (Continued)

UPPER ELEVATION LOWER ELEVATION

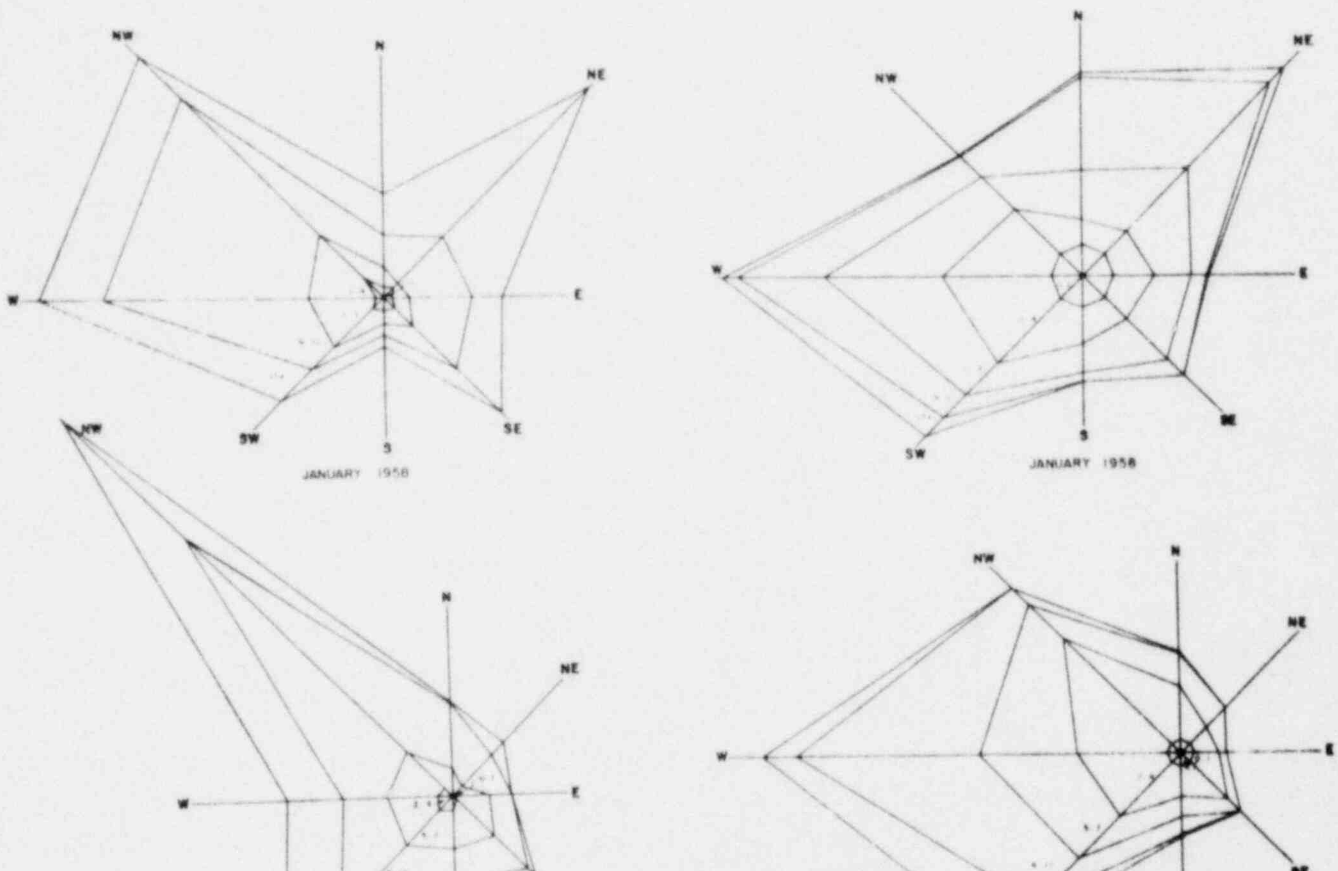


Figure 2A-3. (Continued)

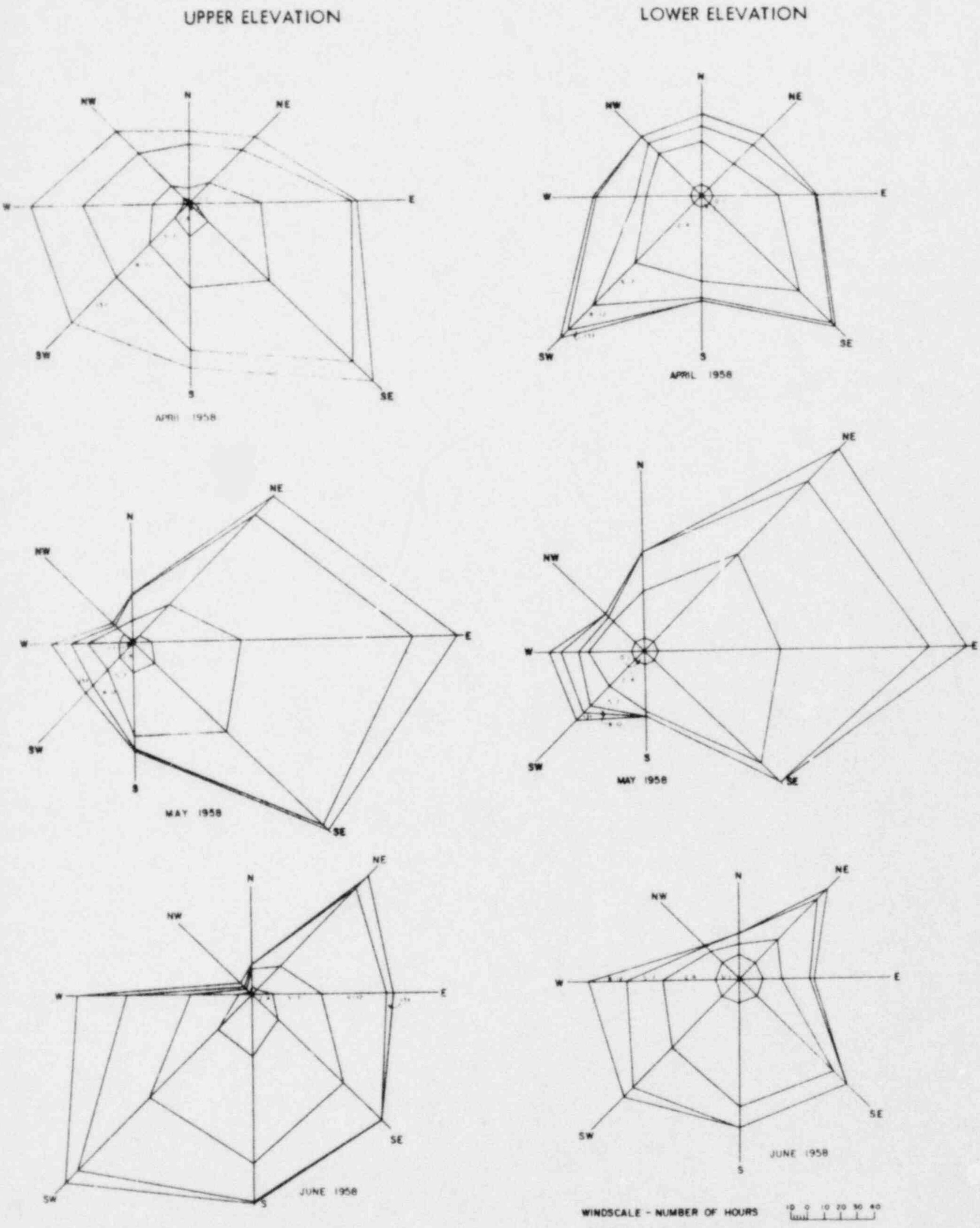


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Figure 2A-3 (Continued)

Table 2A-1

MAXIMUM HOURLY AIR MOVEMENT

UF SITE DATA

Date	Time	Wind Velocity (mph)
January 10	12 - 1 p.m.	W-20
February 19	11 - 12 a.m.	W-25
March 26	2 - 3 p.m.	W-24
April 5	12 - 1 p.m.	SW-22
May 4	12 - 1 p.m.	W-22
June 28	2 - 3 p.m.	SW-20
July 1	3 - 4 p.m.	W-15
August 21	2 - 3 p.m.	NE-18
August 22	3 - 4 p.m.	NE-18
August 23	4 - 5 p.m.	NE-18
September 8	2 - 3 p.m.	S-18
October 9	4 - 5 a.m.	NE-16
November 11	11 - 12 p.m.	NE-20
November 25	12 - 1 p.m.	W-20
December 11	4 - 5 p.m.	NW-21

Table 2A-2

WIND DIRECTION AND SPEED
Per cent of total number of hourly occurrences for each direction and speed group. Wind speed in miles per hour.

	125 Ft. Level (June, 1957 - May, 1958)							30 Ft. Level (June, 1957 - May, 1958)						
Calm (0-1)	Calm	2-4	5-7	8-12	13+	Missing	Total %	Calm	2-4	5-7	8-12	13+	Missing	Total %
Calm (0-1)	1.1					4.0	5.1	17.0					9.6	26.6
N		0.8	1.7	3.2	0.5		6.2		3.4	2.1	0.8	0.0		6.3
NE		0.9	3.2	7.6	3.6		15.3		6.4	6.6	2.9	0.2		16.1
E		1.2	5.2	7.3	1.4		15.1		4.8	3.7	1.1	0.0		9.6
SE		2.2	6.4	5.4	1.0		15.0		8.0	2.4	0.4	0.0		10.8
S		2.2	3.9	2.7	0.7		9.5		3.6	1.1	0.3	0.0		5.0
SW		2.1	4.3	3.3	2.0		11.7		4.4	2.6	1.4	0.6		9.0
W		1.1	3.5	4.3	2.0		10.9		3.9	2.7	3.1	0.5		10.2
NW		1.2	2.4	5.3	2.3		11.2		3.7	2.0	0.7	0.0		6.4
TOTAL %	1.1	11.7	30.6	39.1	13.5	4.0	100.0	17.0	38.2	23.2	10.7	1.3	9.6	100.0

Table 2A-3

PERSISTENCE OF WIND DIRECTION
June 1957 - May 1958 (30 ft. Level)

Hours	Calm	N	NE	E	SE	S	SW	W	NW	Missing	Total Frequency	Total Hours
1	122	107	140	135	137	117	136	132	123	5	1154	1154
2	51	38	63	65	70	63	46	64	30	4	494	988
3	31	25	39	39	49	16	26	26	21	4	276	828
4	23	14	22	28	27	18	19	26	15	0	192	688
5	20	8	16	11	16	5	12	21	9	1	119	595
6	13	5	8	9	18	5	11	7	4	2	82	492
7	17	6	8	6	7	1	3	4	6	1	59	413
8	16	5	3	4	6	3	3	6	4	2	52	416
9	5	1	5	2	2	1	6	3	1	0	26	234
10	16	1	4	2	5	2	3	4	1	0	38	380
11	6	2	5	2	4		4	3	0	6	32	352
12	5	1	5	3	3		2	2	1	0	22	264
13	5	2	1	2	3		5	1	1	1	21	273
14	4		1	1	1		1	2	2	0	12	168
15	2		3		0				0	1	6	90
16			1		1		0		1	2	5	80
17			4	1	1				1	0	7	119
18		1	2				2		0	1	6	108
19			5						1	0	6	114
20			1				1			2	4	80
21								1		2	3	63
22								2		1	3	66
23										0	0	0
24			1					1		2	4	96
25			1							0	1	25
26										1	1	26
27			1					1		0	2	54
28			1							0	1	28
29										1	1	29
30			1							0	1	30
33			1					1		0	2	66
35										1	1	35
36										1	1	36
40										1	1	40
41			1	1						0	2	82
47										1	1	47
63			1							0	1	63
138										1	1	138

TOTAL 8,760

Table 2A-4

DAILY RAINFALL FOR 1957

Gainesville, Florida

Date	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.
1			.01	.85	.06			1.23		1.25		
2				.31	.07		.40			.47		
3			.04		.20	.03	.54			.08		
4			1.91		.06	.12	.60		.26	.01		.15
5			.01	.78	.01	.97	T	.09	1.00			
6			.09	.03		.95	.03	3.07	.05			
7						.19	.57	.35	.40			
8			.13			2.25	.07	.85	.32			
9		.03				1.19		.04	.29		.12	.61
10					.20	.07		1.84	.37			
11		.15		.03	.53		.39		.14			.07
12				.89			.39					
13	.02				.79		.02					
14					T		.53			.01	.62	
15					.12		.01		.23		.12	
16	.33				3.00		T		.60			
17	T				.11	.01	1.44		1.45			
18					.02	.30	.05	.16	.35			
19		.21	.03	.55	.08		1.16	.90	.14		.24	
20		.54		.05	.02				.15			
21	.04	.33										
22	.20		1.25			.55						
23					.02	.03		.51				
24			.74		.55					.10		
25		.10	1.14		.01	.19	.15			.02	.49	
26		.44				.32	T					.04
27					.05	.02	.43		.03			
28		.57				.07	.42	.07	.33			
29					.68	.25	.03	1.11	.39			
30				.45			1.20	.11			.53	
31					.11		.29					
Total	0.59	2.37	5.35	3.94	6.69	7.51	8.72	10.33	6.50	1.94	2.12	0.87

Total for the Year 56.93

Precipitation data obtained from the Weather Bureau's hourly totals is presented on the monthly wind roses for the upper station for the period January through December, 1957, in Figures 2A-2 and 2A-3. The shaded areas indicate the number of hours in each velocity range during which precipitaition occurred. Additional months will be analyzed as information becomes available from the U.S. Weather Bureau.

An analysis of the frequency of wind direction by velocity groups during precipitation is given in Table 2A-5.

2A.3 Inversion and Atmospheric Stability

In May 1957, equipment was installed on a 400 ft. radio tower about two miles west of campus to obtain vertical temperature data. Three days later this installation was destroyed by lightning. Continued attempts to install equipment on this tower met with difficulties, so another location was selected. This new location on the College of Engineering radar tower has now been instrumented. The installation consists of shielded thermocouples, exposed at elevations of 130 ft. and 5 ft. above the ground, connected to a recording potentiometer. Stability conditions of the atmosphere will be determined in this manner and inversion data computed from the temperature profiles obtained.

Due to the lack of temperature lapse rate data, a study was made to estimate the relative frequency of turbulent and stable conditions using the wind speed ratio obtained from readings at the 125 ft. and 30 ft. levels as the criterion of turbulence.

As discussed in the Summary Report for the Argonaut Reactor, the British Chemical Warfare Service has used ratios as a measure of turbulence very successfully. The same range of n, a parameter related to wind ratio by the equation

$$R = \frac{U}{U_o} = \frac{Z}{Z_o}^{\frac{n}{2-n}} \qquad (2A\text{-}1)$$

U = wind speed at upper level

U_o = wind speed at lower level

Z = height at upper level

Z_o = height at lower level

was used to define the same three classifications of turbulence as used in the Argonne Report.

It should be recognized that no long-range conclusions can be drawn from this study regarding turbulent and stable conditions since insufficient data were available.

The results of this study are presented in Table 2A-6.

*"Summary Report on the Hazards of the Argonaut Reactor," D.H. Lennox and C.N. Kelber, ANL 5647.

Table 2A-5

WIND FREQUENCY DURING PRECIPITATION

University of Florida, Gainesville, Florida

Number of hours, divided into wind direction and velocity categories, during which precipitation occurred during 1957.
M = missing data

Dir.	2-4	5-7	8-12	13+	Total	2-4	5-7	8-12	13+	Total
N					0				1	1
NE					0	1	2	7	6	16
E		5	2		7		1	5		6
SE			1		1		5	7		12
S					0	1				1
SW		4			4	1		1		2
W	1		1		2	1		1		2
NW					0					0
Total	1	9	4	0	14	4	8	21	7	40
January: calm 0, M. 0					14	February: calm 0, M. 0				40

Dir.	2-4	5-7	8-12	13+	Total	2-4	5-7	8-12	13+	Total
N			1	2	3		1	2		3
NE		1	3	4	8		2			2
E	1	5	6	4	16		2	6		8
SE		1		2	3	2	3	5	3	13
S			1	1	2			1		1
SW		1	3	3	7				3	3
W	1		5		6				1	1
NW					0				1	1
Total	2	8	19	16	45	2	8	14	8	32
March: calm 1, M. 0					46	April: calm 1, M. 0				33

Dir.	2-4	5-7	8-12	13+	Total	2-4	5-7	8-12	13+	Total
N			2		2	1		1		2
NE	1			1	2			3		3
E		4	1		5	1	1	3		5
SE	2	5	8	1	16	1	5	3		9
S	1	4			5	5	6	10	8	29
SW	1	2	1		4	1	6	5	6	18
W		3	1		4		2	4		6
NW	1				1		2			2
Total	6	18	13	2	39	9	22	29	14	74
May: calm 1, M. 0					40	June: calm 0, M. 0				74

(continued)

(continued)

Dir.	2-4	5-7	8-12	13+	Total	2-4	5-7	8-12	13+	Total
N			2		2					0
NE			5		5		2	2	2	6
E	1	1			2	1	5	6	1	13
SE	2	11	2	3	18	2	2	2	1	7
S	1	3	2		6	2	1	1		4
SW		7	6		13	1	4	5	1	11
W	1	4	5	1	11		2	2		4
NW		1	3		4	2				2
Total	5	27	25	4	61	8	16	18	5	47
July: calm 0, M. 0					61	August: calm 0, M. 0				47
N			1		1		1			1
NE		2	6		8	1				1
E		2	4		6					0
SE	3	6	10		19	5	6			11
S		3	4	4	11					0
SW	1	3	2	2	8			2		2
W	2		1		3	1		1		2
NW	4	1	1		6	3	2			5
Total	10	17	29	6	62	10	9	3		22
September: calm 3, M. 4					69	October: calm 1, M. 3				26
N					0	1				1
NE		1			1					0
E					0					0
SE	1	1	1		3					0
S		2	4	1	7	1		1		2
SW	1	1		1	3	1	2	2		5
W			2		2	2	3	3		8
NW			2		2		4	1		5
Total	2	5	9	2	18	5	9	7	0	21
November: calm 0, M. 0					18	December: calm 0, M. 0				21
N	2	2	9	3	16					
NE	3	10	26	13	52					
E	4	26	33	5	68					
SE	18	45	39	10	112					
S	11	19	24	14	68					
SW	7	30	27	16	80					
W	9	14	26	2	51					
NW	10	10	7	1	28					
Total	64	156	191	64	475					
Year: calm 7, M. 7					489					

Table 2A-6

ESTIMATED FREQUENCY OF TURBULENCE

Number of occurrences of ratio of wind speed at 125 ft. to wind speed at 30 ft.(R) for three general classifications of turbulence grouped according to wind speed at 30 ft. level (June 1957 - May 1958)

			Wind Speed (MPH) for 30 ft. level						
	n	R	0-1	2-4	5-7	8-12	13+	Total	%
Turbulent	1.000-0.268	0.62-1.25	241	245	193	73	7	759	8.7
Neutral	0.279-0.340	1.26-1.34	0	88	135	75	23	321	3.7
Non-turbulent	0.346-1.000	1.34-4.17	1239	3041	1721	789	88	6858	78.5
						Missing Data			9.1
						Total			100.0

Appendix 2B

LATEST UFTR METEOROLOGICAL DATA
FROM
FLORIDA POWER CORPORATION CRYSTAL RIVER PLANT
AND
GAINESVILLE UTILITIES DEERHAVEN PLANT

Table 2B-1

FLORIDA POWER CORPORATION CRYSTAL RIVER METEOROLOGICAL DATA

FPC - CRYSTAL RIVER 33 FT WINDS DEL T 1/1/75 - 12/31/75

TEMP. LAPSE RATE STABILITY CLASS A
WIND SPEED VERSUS DIRECTION (IN NUMBER OF OBS.)

WIND	WIND SPEED (MPH) AT 10 METER LEVEL						
DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
NNE	1	17	29	1	0	0	48
NE	0	30	83	15	0	0	128
ENE	1	27	84	23	0	0	135
E	3	37	92	8	0	0	140
ESE	0	6	47	3	0	0	56
SE	0	6	19	4	0	0	29
SSE	0	4	24	7	1	0	36
S	0	6	27	24	5	0	62
SSW	1	4	47	34	4	0	90
SW	0	10	18	17	1	0	46
WSW	1	34	58	7	0	0	100
W	3	105	218	7	0	0	333
WNW	2	32	171	16	7	0	228
NW	2	13	50	16	4	2	87
NNW	2	7	12	0	0	0	21
N	1	23	32	5	0	0	51
TOTAL	17	361	1011	187	22	2	1600

PERIODS OF CALM (NO. OF HOURS) - 0

****

TEMP. LAPSE RATE STABILITY CLASS B
WIND SPEED VERSUS DIRECTION (IN NUMBER OF OBS.)

WIND	WIND SPEED (MPH) AT 10 METER LEVEL						
DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
NNE	0	0	2	0	0	0	2
NE	1	3	12	3	0	0	19
ENE	0	5	10	0	0	0	15
E	1	6	9	0	0	0	16
ESE	0	1	5	0	0	0	6
SE	1	0	4	2	0	0	7
SSE	0	2	7	3	0	0	12
S	0	6	4	0	0	0	10
SSW	0	3	13	5	1	0	22
SW	0	3	5	3	1	0	12
WSW	1	4	14	0	0	0	19
W	0	11	8	2	0	0	21
WNW	0	6	7	0	0	0	13
NW	0	4	4	3	1	0	12
NNW	1	0	2	2	0	0	5
N	0	5	2	0	0	0	7
TOTAL	5	59	108	23	3	0	198

PERIODS OF CALM (NO. OF HOURS) - 0

****

TEMP. LAPSE RATE STABILITY CLASS C
WIND SPEED VERSUS DIRECTION (IN NUMBER OF OBS.)

WIND	WIND SPEED (MPH) AT 10 METER LEVEL						
DIRECTION	1-3	4-7	8-12	13-18	19-24	>24	TOTAL
NNE	2	4	4	0	0	0	10
NE	2	8	20	4	[illegible]	0	34
ENE	4	7	22	4	[illegible]	0	37
E	0	12	16	1	0	0	29
ESE	1	13	16	3	0	0	33
SE	1	7	6	1	0	0	15
SSE	0	9	9	4	0	0	22
S	0	7	11	4	0	0	22
SSW	0	4	20	9	0	0	33
SW	3	3	21	5	0	0	37
WSW	1	16	20	1	0	0	38
W	1	23	23	0	0	0	47
WNW	1	10	16	1	[illegible]	0	29
NW	0	5	15	7	[illegible]	0	30
NNW	1	5	4	0	0	0	10
N	1	6	13	0	0	0	20
TOTAL	18	144	236	44	4	0	446

PERIODS OF CALM (NO. OF HOURS) - 0

Table 2B-1 (Continued)

TEMP. LAPSE RATE STABILITY CLASS D
WIND SPEED VERSUS DIRECTION (IN NUMBER OF OBS.)

WIND DIRECTION	WIND SPEED (MPH) AT 10 METER LEVEL 1-3	4-7	8-12	13-18	19-24	>24	TOTAL
NNE	5	17	50	4	0	0	76
NE	3	41	153	12	1	0	210
ENE	10	36	64	13	0	0	123
E	9	61	38	2	0	0	110
ESE	8	42	22	5	0	0	77
SE	5	48	46	10	0	0	109
SSE	5	40	37	9	3	0	94
S	3	29	36	25	7	0	100
SSW	4	43	52	43	9	1	152
SW	10	64	117	60	1	0	252
WSW	6	44	85	9	0	0	144
W	7	50	49	15	0	0	121
WNW	8	30	47	18	2	0	105
NW	7	39	46	26	17	0	135
NNW	4	21	22	10	1	0	58
N	5	56	58	6	0	0	125
TOTAL	99	661	922	267	41	1	1991

PERIOD OF CALM (NO. OF HOURS) - 0

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TEMP. LAPSE RATE STABILITY CLASS E
WIND SPEED VERSUS DIRECTION (IN NUMBER OF OBS.)

WIND DIRECTION	WIND SPEED (MPH) AT 10 METER LEVEL 1-3	4-7	8-12	13-18	19-24	>24	TOTAL
NNE	11	55	47	1	0	0	114
NE	16	81	149	3	0	0	249
ENE	18	168	117	0	0	0	303
E	31	195	39	2	0	0	267
ESE	24	106	45	1	0	0	176
SE	9	142	49	2	0	0	202
SSE	4	42	26	19	0	0	91
S	6	44	53	33	5	1	142
SSW	3	23	39	19	2	1	87
SW	7	20	32	7	0	0	66
WSW	8	36	36	8	0	0	88
W	8	82	51	3	1	0	145
WNW	5	46	37	3	0	0	91
NW	6	33	28	7	2	0	76
NNW	16	61	26	12	0	0	115
N	14	82	65	4	1	0	166
TOTAL	186	1216	839	124	11	2	2378

PERIOD OF CALM (NO. OF HOURS) - 2

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TEMP. LAPSE RATE STABILITY CLASS F
WIND SPEED VERSUS DIRECTION (IN NUMBER OF OBS.)

WIND DIRECTION	WIND SPEED (MPH) AT 10 METER LEVEL 1-3	4-7	8-12	13-18	19-24	>24	TOTAL
NNE	14	52	22	0	0	0	88
NE	32	46	34	2	0	0	114
ENE	28	172	59	0	0	0	259
E	64	169	5	0	0	0	238
ESE	35	78	13	0	0	0	126
SE	11	85	2	0	0	0	98
SSE	3	18	1	0	0	0	22
S	2	9	2	2	0	0	15
SSW	5	1	2	0	0	0	8
SW	1	1	0	0	0	0	2
WSW	0	0	0	0	0	0	0
W	0	0	1	0	0	0	1
WNW	4	0	0	0	0	0	4
NW	5	4	0	0	0	0	9
NNW	9	17	3	2	0	0	31
N	18	61	15	0	0	0	94
TOTAL	231	713	159	6	0	0	1109

PERIODS OF CALM (NO. OF HOURS) - 4

Table 2B-1 (Continued)

TEMP. LAPSE RATE STABILITY CLASS G
WIND SPEED VERSUS DIRECTION (IN NUMBER OF OBS.)

WIND DIRECTION	WIND SPEED (MPH) AT 10 METER LEVEL 1-3	4-7	8-12	13-18	19-24	>24	TOTAL
NNE	16	20	9	0	0	0	45
NE	18	27	6	0	0	0	51
ENE	18	68	31	0	0	0	117
E	35	72	2	0	0	0	109
ESE	18	34	0	0	0	0	52
SE	8	28	1	0	0	0	37
SSE	2	2	0	0	0	0	4
S	0	2	0	0	0	0	2
SSW	0	0	0	0	0	0	0
SW	0	0	0	0	0	0	0
WSW	0	0	0	0	0	0	0
W	0	0	0	0	0	0	0
WNW	0	0	0	0	0	0	0
NW	1	1	0	0	0	0	2
NNW	2	2	1	0	0	0	5
N	8	25	5	0	0	0	38
TOTAL	126	281	55	0	0	0	462

PERIOD OF CALM (NO. OF HOURS) - 2

Table 2B-2

GAINESVILLE UTILITIES - DEERHAVEN PLANT METEOROLOGICAL DATA

METEOROLOGICAL INPUT DATA FOR THE ANNUAL SEASON

MIXING DEPTH = 1450. METERS
AMBIENT TEMPERATURE = 298. DEGREES, KELVIN
AMBIENT PRESSURE = 1000. MILLIBARS

STABILITY CLASS 1

	WINDSPEED CLASS*					
WIND DIRECTION	1	2	3	4	5	6
N	0.0010	0.0021	0.0	0.0	0.0	0.0
NNE	0.0007	0.0034	0.0	0.0	0.0	0.0
NE	0.0007	0.0017	0.0	0.0	0.0	0.0
ENE	0.0003	0.0031	0.0	0.0	0.0	0.0
E	0.0	0.0031	0.0	0.0	0.0	0.0
ESE	0.0	0.0014	0.0	0.0	0.0	0.0
SE	0.0007	0.0017	0.0	0.0	0.0	0.0
SSE	0.0	0.0017	0.0	0.0	0.0	0.0
S	0.0003	0.0010	0.0	0.0	0.0	0.0
SSW	0.0	0.0003	0.0	0.0	0.0	0.0
SW	0.0003	0.0007	0.0	0.0	0.0	0.0
WSW	0.0	0.0021	0.0	0.0	0.0	0.0
W	0.0003	0.0024	0.0	0.0	0.0	0.0
WNW	0.0010	0.0038	0.0	0.0	0.0	0.0
NW	0.0007	0.0014	0.0	0.0	0.0	0.0
NNW	0.0	0.0014	0.0	0.0	0.0	0.0

*Wind Speed Class #1 = 1-3 Knots
#2 = 4-6 Knots
#3 = 7-10 Knots
#4 = 11-16 Knots
#5 = 17-21 Knots
#6 = Greater than 21 Knots

Table 2B-2 (Continued)

STABILITY CLASS 2

WIND DIRECTION	WINDSPEED CLASS 1	2	3	4	5	6
N	0.0014	0.0027	0.0027	0.0	0.0	0.0
NNE	0.0007	0.0007	0.0021	0.0	0.0	0.0
NE	0.0007	0.0041	0.0024	0.0	0.0	0.0
ENE	0.0010	0.0024	0.0034	0.0	0.0	0.0
E	0.0014	0.0045	0.0065	0.0	0.0	0.0
ESE	0.0010	0.0034	0.0051	0.0	0.0	0.0
SE	0.0017	0.0017	0.0041	0.0	0.0	0.0
SSE	0.0014	0.0034	0.0045	0.0	0.0	0.0
S	0.0021	0.0010	0.0021	0.0	0.0	0.0
SSW	0.0003	0.0014	0.0021	0.0	0.0	0.0
SW	0.0017	0.0014	0.0041	0.0	0.0	0.0
WSW	0.0021	0.0017	0.0017	0.0	0.0	0.0
W	0.0017	0.0024	0.0072	0.0	0.0	0.0
WNW	0.0007	0.0024	0.0041	0.0	0.0	0.0
NW	0.0014	0.0021	0.0055	0.0	0.0	0.0
NNW	0.0017	0.0027	0.0051	0.0	0.0	0.0

STABILITY CLASS 3

WIND DIRECTION	WINDSPEED CLASS 1	2	3	4	5	6
N	0.0	0.0003	0.0021	0.0007	0.0	0.0
NNE	0.0	0.0010	0.0034	0.0017	0.0	0.0
NE	0.0	0.0010	0.0048	0.0031	0.0	0.0
ENE	0.0	0.0007	0.0038	0.0031	0.0	0.0
E	0.0	0.0007	0.0055	0.0062	0.0	0.0
ESE	0.0	0.0010	0.0068	0.0062	0.0	0.0
SE	0.0	0.0014	0.0038	0.0031	0.0	0.0
SSE	0.0	0.0014	0.0045	0.0021	0.0	0.0
S	0.0	0.0010	0.0038	0.0027	0.0	0.0
SSW	0.0	0.0014	0.0031	0.0021	0.0	0.0
SW	0.0	0.0	0.0027	0.0017	0.0	0.0
WSW	0.0	0.0007	0.0027	0.0024	0.0003	0.0
W	0.0	0.0003	0.0068	0.0058	0.0003	0.0
WNW	0.0	0.0017	0.0024	0.0065	0.0	0.0
NW	0.0	0.0007	0.0031	0.0017	0.0	0.0
NNW	0.0	0.0010	0.0024	0.0010	0.0	0.0

Table 2B-2 (Continued)

STABILITY CLASS 4

WIND DIRECTION	WINDSPEED CLASS 1	2	3	4	5	6
N	0.0010	0.0092	0.0082	0.0003	0.0	0.0
NNE	0.0034	0.0092	0.0120	0.0017	0.0	0.0
NE	0.0038	0.0110	0.0120	0.0072	0.0	0.0
ENE	0.0024	0.0082	0.0161	0.0031	0.0	0.0
E	0.0027	0.0158	0.0168	0.0068	0.0	0.0
ESE	0.0021	0.0123	0.0164	0.0048	0.0003	0.0
SE	0.0038	0.0086	0.0092	0.0055	0.0	0.0
SSE	0.0031	0.0065	0.0075	0.0045	0.0003	0.0
S	0.0041	0.0079	0.0120	0.0065	0.0	0.0
SSW	0.0038	0.0041	0.0075	0.0027	0.0	0.0
SW	0.0021	0.0045	0.0086	0.0051	0.0003	0.0
WSW	0.0034	0.0034	0.0113	0.0079	0.0007	0.0003
W	0.0027	0.0086	0.0236	0.0130	0.0007	0.0
WNW	0.0017	0.0110	0.0072	0.0096	0.0	0.0
NW	0.0034	0.0086	0.0089	0.0041	0.0	0.0
NNW	0.0031	0.0116	0.0120	0.0021	0.0	0.0

STABILITY CLASS 5

WIND DIRECTION	WINDSPEED CLASS 1	2	3	4	5	6
N	0.0096	0.0089	0.0	0.0	0.0	0.0
NNE	0.0079	0.0062	0.0	0.0	0.0	0.0
NE	0.0079	0.0089	0.0	0.0	0.0	0.0
ENE	0.0092	0.0068	0.0	0.0	0.0	0.0
E	0.0096	0.011[illegible]	0.0	0.0	0.0	0.0
ESE	0.0096	0.0065	0.0	0.0	0.0	0.0
SE	0.0065	0.0058	0.0	0.0	0.0	0.0
SSE	0.0086	0.0031	0.0	0.0	0.0	0.0
S	0.0065	0.0038	0.0	0.0	0.0	0.0
SSW	0.0082	0.0021	0.0	0.0	0.0	0.0
SW	0.0082	0.0024	0.0	0.0	0.0	0.0
WSW	0.0072	0.0021	0.0	0.0	0.0	0.0
W	0.0082	0.0161	0.0	0.0	0.0	0.0
WNW	0.0103	0.0161	0.0	0.0	0.0	0.0
NW	0.0075	0.0082	0.0	0.0	0.0	0.0
NNW	0.0082	0.0089	0.0	0.0	0.0	0.0

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

This chapter identifies, describes, and discusses the principal architectural and engineering design features of the UFTR building and integral structural systems. This presentation is simplified considerably from that required for the typical power reactor due to the characteristics of the UFTR which is an unpressurized, compact reactor that is contained within a single structure. The UFTR reactor building and its integral structural systems are the only features considered in this chapter, while all the systems dealing directly with the reactor are covered in Chapter 4.

3.1 Structural Design

The reactor building, pictured in Figure 3-1, is a "vault-type" building as defined in 10 CRF 73.2(o). The reactor building is divided into two distinct parts based upon the difference in utilization and their structure. The overall reactor building measures approximately 60 ft. by 80 ft. inside as depicted in Figure 3-2. The reactor or cell area is 30 ft. by 60 ft. with 29 ft. of head room, located at the north end of the building. The rest of the building is used for research laboratories, faculty offices and graduate student study areas. The current floor plan for both levels of the building is shown in Figures 3-2 and 3-3 which shows a number of building changes from the original floor plan primarily aimed at increased security and area utilization.

The office laboratory section of the building is constructed of concrete columns and beams with hollow cement block curtain walls and metal sash windows. The exposed cement blocks on the inside are painted with primer and two coats of semigloss enamel while red brick veneer is used on the outside. The floors and roof are poured concrete slabs, covered with vinyl or asphalt tile.

Some relatively minor alterations have been made to the first floor plan including changes in the West entrance door for Laboratory Room 104 plus removal of the loop, rolling mill, and swager. In addition, administrative offices have been installed inside the Laboratory Room 104 on the East side opening into the reactor lobby outside Room 103 only. In addition, a wooden partition with a permanently locked door and intercom alarm has been installed to the South of the stairs in the reactor lobby leading to the second floor. This lock and intercom system limits free access into the area just outside the control room to approved personnel. A metal door unit with intercom at the top of the steps serves the same purpose of limiting free access to the reactor control room door. This pair of permanently locked doors under the control of the reactor operator or his designated representative defines a UFTR "protective zone" outside the reactor control room and reactor cell which makes up the restricted area for this site and is designated a limited access area.

Some alterations have also affected the second floor plan of the UFTR building. First, the viewing windows (glass) in the lobby area looking into the reactor hall and down into the hot cave area have been closed with concrete blocks. The east outside entrance to the lobby was made of glass with a glass door. This arrangement has been replaced with a concrete

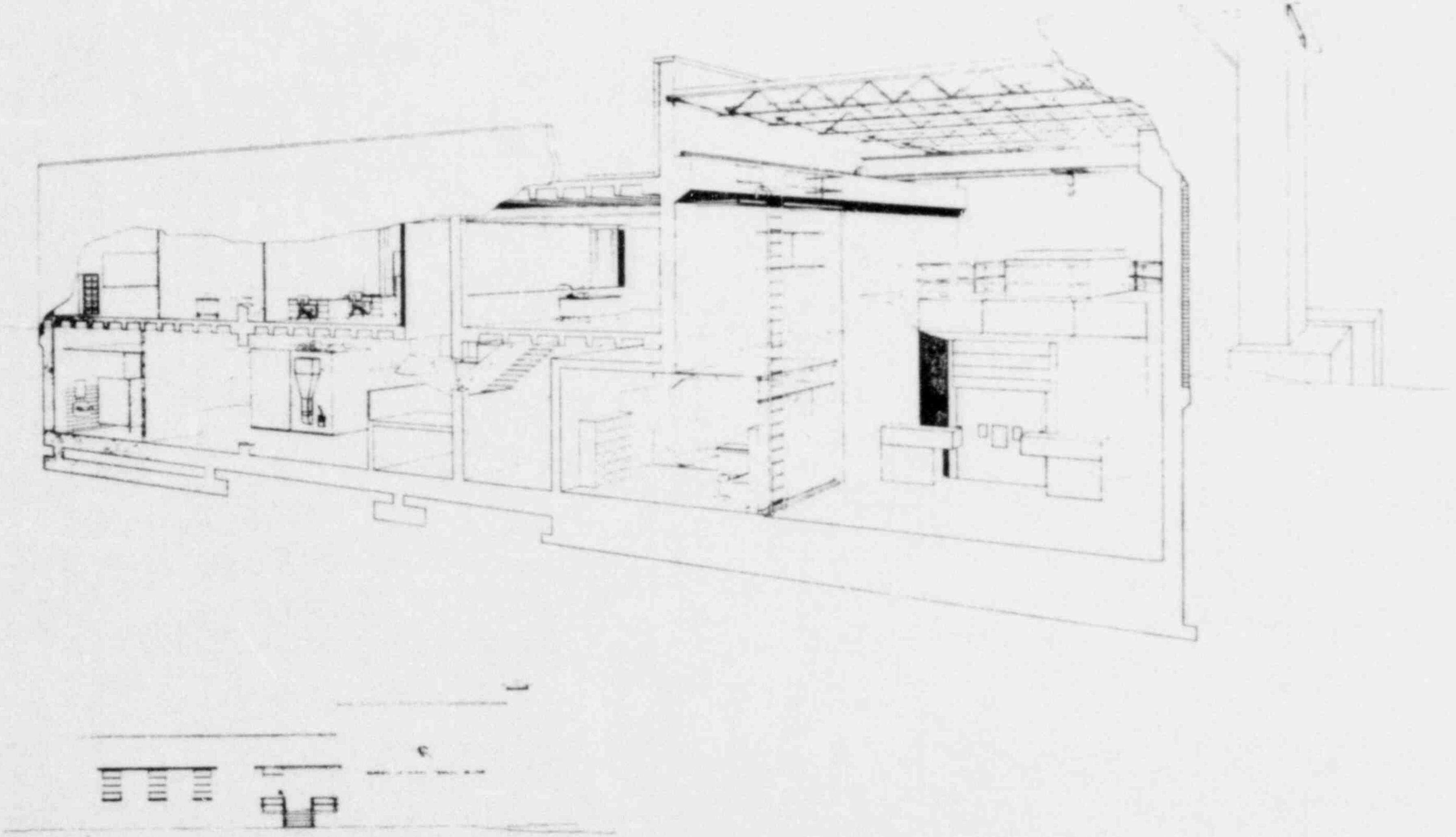


Figure 3-1. University of Florida Training Reactor Facilities - East Face Cutaway View

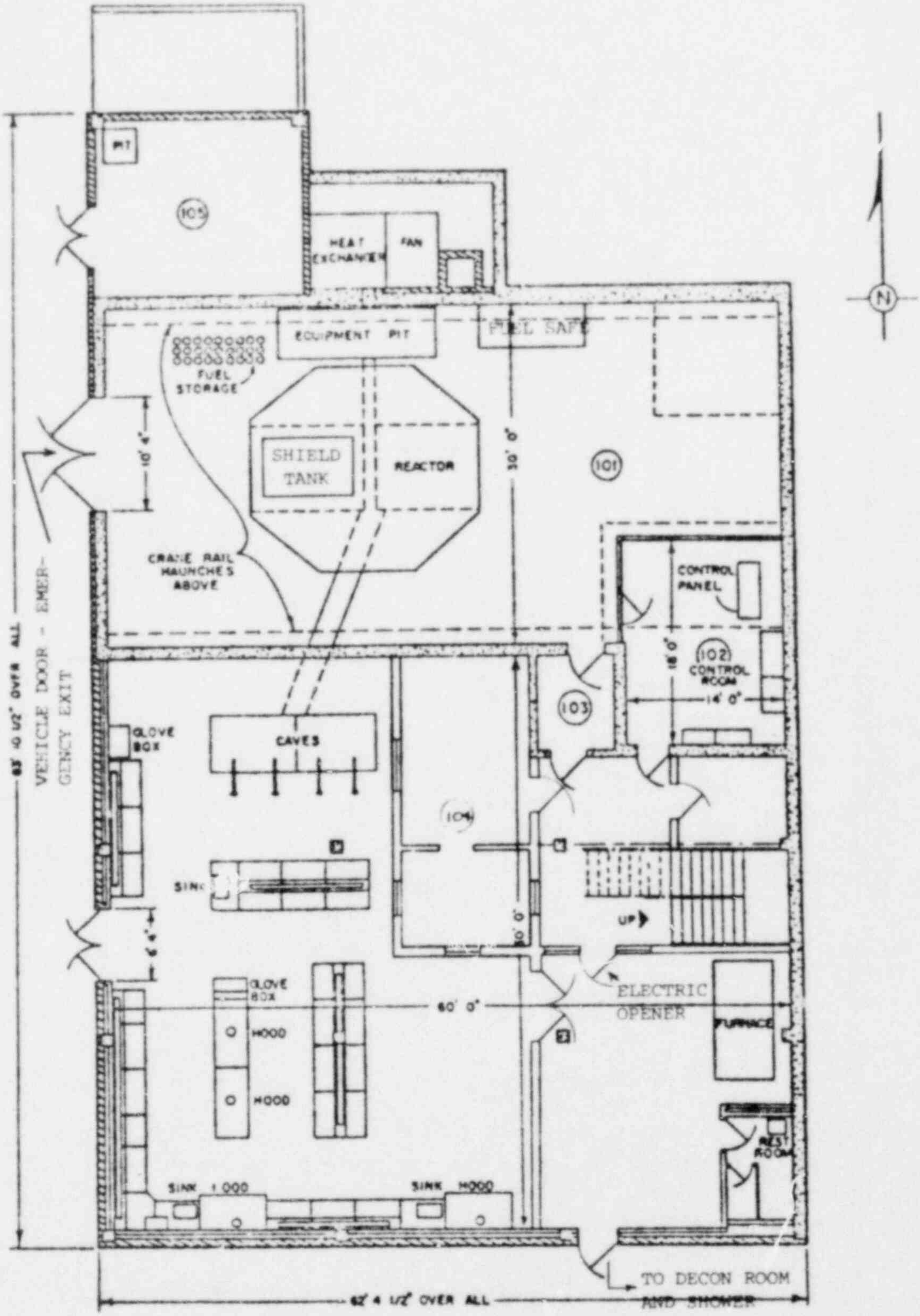


Figure 3-2. First Floor Plan for the University of Florida Training Reactor Building.

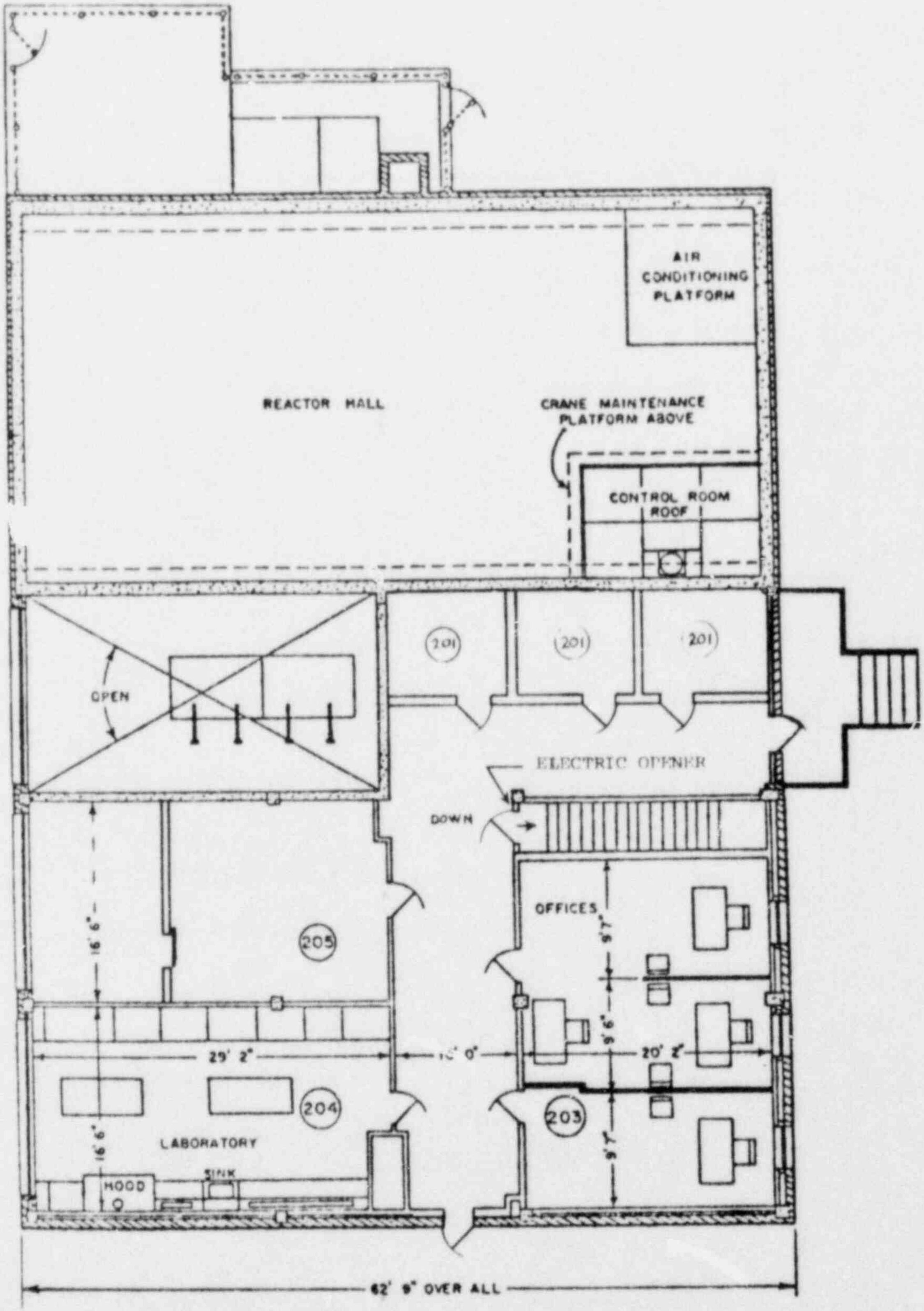


Figure 3-3. Second Floor Plan for the University of Florida Training Reactor Building

block wall faced with bricks similar to the rest of the outside wall construction of the building. The metal access door leading from the outside is always locked and further reduces access to the building to keyed approved personnel only and serves to define the reactor building as an exclusion area for safety analysis purposes.

One other change to the second floor plan is the reduction of lobby space by the addition of three offices labelled 201 as shown in Figure 3.3. All secondary walls are of standard wood/wallboard/paneling con_.ruction for dividing and acoustic insulation purposes only. These walls are cosmetic only and serve no structural purpose. The same type of wall addition was used in Room 205 to make an office and a receptionist area and in Room 203 to divide it into two rooms.

None of the changes outlined here has affected the structural integrity and inherent safety features of the reactor building. These changes have merely been made to facilitate use of the building under chang' g research and training patterns of operation.

The reactor room or cell (area 101), while an integral part of the building, is isolated from the laboratory-office section by a double-door air lock exit/entrance or the single entrance door. The protective zone in the hall outside the cell further serves to isolate the reactor room. The walls of the room are constructed of monolithic reinforced concrete, one foot thick, resting on mat footings. The inside walls of areas 101, 102 and 103 are coated with 7 mils-thick vinyl-epon paint. The floor is a concrete slab resting on undisturbed or compacted earth, as was deemed necessary depending on test boring results. The floor slab has a minimum thickness of one foot, and is increased under the reactor to 18 in. It is designed for a maximum load of 3,000 lbs. per square ft. at the reactor and at least 1,500 lbs. per square ft. over the rest of the area. The floor slab is damp proofed with a barrier of two plies of 15 lbs., felt mopped in place with hot asphalt between the base slab and top slab. All the slab junctions with vertical surfaces are provided with sixteen-ounce copper water stops. These junctions are calked with pre-moulded mastic filler and hot-poured paraplastic seals.

The roof of the reactor is built-up with a 3 in., precast roof tile covered with 2 in. of rigid fiberglass insulation boarded and sealed with 5-ply tarred felt and pitch with a slag covering. The roof of the reactor room is supported on No. 166 steel-bar joists spaced 2 ft. on centers.

The reactor rests on a 16 in. high concrete pedestal in order to raise the beam holes to a convenient 40 in. working level and to support the reactor. A concrete service trench, 5 ft. wide by 2 ft. deep, extends from under the reactor to an equipment pit, measuring 5 ft., 3 in. by 13 ft., 6 in. by 6 ft. deep, located adjacent to the reactor.

The reactor control area (Space 102), housing the reactor console, is located in the southeast corner of the reactor room inside the reactor cell. plate glass wall is provided around the control area to give maximum visibility from the control console to the reactor cell and to isolate this area from the rest of the reactor cell.

The UFTR reactor building has five entrances (exits), but only two--one upstairs and one downstairs--leading into the reactor building from the Nuclear Sciences Center, will be in normal use during regular work hours. The other three entrances (exits) are kept locked at all times. The vehicle/freight doors on the West side of the reactor cell (area 101) are used only in special situations such as refueling the UFTR and now have a personnel door installed for an emergency exit. This door is monitored on the reactor control console by the reactor operator on duty. The door on the West side of the radiochemistry laboratory (area 104) is also only used in special situations such as equipment delivery but is also available for emergency exit from the building. The final exit is on the second floor on the East side outside the offices (area 201) and is also kept locked. This entrance (exit) is in general use for authorized keyed personnel to enter the building at all times. All doors are steel fire-rated doors.

The main reactor room entrance opens close to and in view of the reactor operator in the control room (area 102). The entrance door from the control room to the hall can be easily opened from the inside for use as an emergency exit. This door is weather-stripped with neoprene and is equipped with a door closer. The main reactor room exit (and occasional equipment entrance) (area 103) is equipped with radiation detection/monitoring devices for personnel. This exit has an air lock set-up and is 8 ft., 4 in. long, 7 ft. wide and 8 ft. high. The air lock also opens in view of the reactor operator in the control room and both of its doors are metal fire doors. Both of the doors to the air lock (area 103) are weather stripped with neoprene and have door-closers. These air lock doors are also monitored on the UFTR reactor control console by the reactor operator on duty.

The freight doors will be closed at all times during operation of the reactor and will be opened only during the actual transfer of material or special maintenance activities. The door is 10 ft. wide by 12 ft. high, four-paneled, steel-skinned, honeycombed construction, and hinged door. The sill, jambs, astragals and head have sponge-rubber seals and calking to minimize leakage. The bottom, right-hand panel of the freight door also serves as an emergency personnel exit and can be opened by a panic release. It is also supplied with a door closer.

The reactor is an elongated octagon located in the center of the 30 ft. dimension of the room, 12 ft. from the West end. It has an East-West axis of 20 ft., 4 in. and a North-South axis of 15 ft., 6 in. The clear floor dimensions around the reactor are summarized in Table 3-1.

An observation window was originally provided between the second floor hall and the reactor room and was made up of stationary 1/4 in. thick LEXAN plate, which was a shatter and bullet proof plate, sealed in aluminum frames. An additional observation window was provided between the second floor hall and the hot cave area in the radiochemistry laboratory area in Room 104. For security reasons, these windows were sealed with solid concrete blocks and painted over on the outside with sealer and latex paint. Subsequently, the offices referred to earlier have been added in area 201 on the second floor. These offices are not considered to have any effect on the structural integrity of the reactor building.

Table 3-1

REACTOR DIMENSIONS AND CLEARANCE

1. Room Dimensions (Inside)	
(a) East-West	60 ft.
(b) North-South	30 ft.
(c) Height (Clear)	29 ft.
2. Reactor Dimensions	
(a) East-West	20 ft., 4 in.
(b) North-South	15 ft., 6 in.
(c) Height Above Floor (To Reactor Top)	11 ft., 10½ in.
(d) Height Above Floor (To Top of Water Tank)	14 ft., 10½ in.
3. Clearances	
(a) West End (Water Tank)	12 ft. 0 in.
(b) North Side (Pit)	7 ft., 3 in.
(c) East End (Thermal Column)	27 ft., 8 in.
(d) South Side	7 ft., 3 in.
(e) Corner Beam Tubes	9 ft. to 10 ft.
(f) East End to Control Room	13 ft., 3 in.
(g) Overhead (Crane Hook to Reactor Top)	11 ft., 9 in.
(h) Overhead (Crane Hook to Water Tank)	8 ft., 9 in.
(i) Overhead (Bottom of Bridge Beam to Reactor Top)	15 ft., 1 in.

An air-conditioning equipment platform with 10 ft. by 11 ft. dimensions is located in the northeast corner of the reactor cell. It is built 10 ft. above the floor to provide ample head room for equipment and personnel working under it.

A 3-ton bridge crane is provided for handling shield blocks, lead casks and other heavy equipment; the hoist travel allows coverage of the entire area of the reactor room. Adequate clearance is provided to permit the use of equipment necessary for fuel transfer operations and for the installation of any experimental equipment which might be desired. The clearance over the water tank is sufficient for the lead coffin used to remove irradiated fuel elements from the reactor. A balcony over the control room serves as a shield preventing any damage to the control room from the crane hook or heavy objects being moved with the crane. It also serves as a maintenance area for the crane.

Spent fuel storage is available at the Northwest corner of the reactor, an area which is accessible to the crane. It consists of twenty-seven, 4 in. diameter by 4 ft. deep, steel-lined storage holes embedded in the concrete and equipped with locked plugs.

There are convenience outlets (115-v) and a 208-v., single-phase outlet on the walls of the room. Tap water is available in the vicinity of the equipment pit shown in Figure 3-2. A utility sink is also located in the Northwest corner of the reactor cell. Area 105, located outside the Northwest corner of the reactor cell, is a utility room where service equipment for the building is stored.

The pit outside the reactor room East of Area 105 contains the 10,000-c.f.m. flow rate fan to provide dilution for air coming from the reactor, and a brick flue to carry the exhaust air above the top of the building.

The number of penetrations through the reactor-room walls, floor and ceiling has been kept to a minimum. Table 3-2 identifies the significant penetrations and gives the location of each. All penetrations with the exception of six items (2, 3, 10, 13, 14, 15) are nonmovable installations, either poured-in-place or sealed with neoprene or mastic gaskets.

The main exit door for personnel (Item 2 in Table 3-2) has an air lock 8 ft., 4 in. long, 7 ft. wide and 8 ft. high. The outer door is kept locked to entrance from the Limited Access Area as previously indicated. This air lock opens close to and in plain view of the operator in the control room. Both doors to the air lock are weather-stripped with neoprene and have door-closers. The main entrance door (Item 3 in Table 3-2) from the Limited Access Area to the control room is weather-stripped with neoprene and equipped with a door closer. It is locked to entrance unless under supervision of the reactor operator but can be opened easily from the inside for use as an emergency exit only. It opens directly into the control room and in plain view of the operators present.

Table 3-2

SIGNIFICANT PENETRATION IN UFTR REACTOR CELL

PENETRATION	INSIDE LOCATION	ELEVATION FROM FLOOR	OUTSIDE ACCESS
1. Pipe Chase	South wall, east end	10 ft. (above control room)	Northeast corner of lab (104) above control room
2. Entrance door	South wall, control room	Floor level	Limited Access Area outside control room
3. Exit door	South wall, air lock	Floor level	Limited Access Area outside control room
4. Two conduits, 3/4 in.	South wall	29 ft.	Roof of lab
5. Three conduits, 3 1/2 in.	South wall, west end	29 ft.	Roof of lab
6. Pipe chase	South wall, west end	15 ft.	N.W. corner of lab (104)
7. Pipe chase	South wall, west end	10 ft.	N.W. corner of lab (104)
8. Conduit, 1/2 in.	South wall, center	14 ft.	Lab roof, north end behind brick veneer
9. Freight door, 10 ft. x 12 ft. containing inset personnel exit	West wall	Floor level	West side, ground level
10. Pipe chase	North wall, west end	5 ft.	Utility room (105), north end of building
11. Three conduits, 3 1/2 in.	North wall, west end	29 ft.	Roof of utility room (105)
12. Air conditioning	North wall, east end	11 ft. (above platform)	N.E. corner, ground level
13. Reactor coolant system	North wall, equipment pit	1 ft. 4 in. below floor	North side of building in heat exchange pit
14. Sandfilled trench to laboratory (will soon be plugged)	Under south wall, 18 ft. from west end	1 ft. below	North wall of lab (104)
15. Drain (plugged)	Southwest corner of pit floor	6 ft. below	Storm sewer
16. Reactor vent	Equipment pit, north wall, east end	3 ft. below	Top of stack
17. Drain (sanitary)	Floor, northwest corner	Floor level	Sanitary sewer
18. Drain vent	Roof, northwest corner	30 ft.	Roof, reactor room (101)
19. Two conduit openings to Annex building (1 capped, 1 plugged)	North wall, center	20 ft. above	Annex Building
20. Wellpump discharge 3 in., and conduit, 1 in.	Northeast corner	9 ft. above	Ground level
21. Telephone cable	North wall, east end	11 ft. above	Annex building
22. Well pump conduit piping, 1 in.	South wall, west end	10 ft. above	Radiochemistry lab, north wall
23. Personnel Emergency Exit	West wall, center	Floor level	Ground level
24. Fire alarm conduit, 3/4 in.	South wall, east side	14 ft. above	Limited Access Area outside cell entrance

3.2 Waste Water Holdup System

The waste water holdup system for the UFTR is designed for operation whenever there is a need for liquid waste holdup originating from the UFTR facility or the laboratories located in the reactor building and the adjacent Nuclear Sciences Center Building.

3.2.1 General Description

The waste holdup system consists of two holdup tanks with a capacity of approximately 26,000 gallons each. These are in ground tanks located outside of the reactor building on the North side of the building. Any liquid waste from the UFTR is pumped or drained into the reactor sink and subsequently stored in the waste holdup tanks. Periodic samples of the liquid waste are taken by the Radiation Control Office and assays are performed to determine the type and quantity of isotopes present. If the activity levels are below acceptable levels for release, then the contents of the tank are released into the University of Florida sanitary sewage system where it is diluted by an average flow of approximately one million gallons per day of sewage. If the level of activity of the sample is found to be above acceptable limits, the remaining part of the original sample will be returned to the holdup tank via the "hot drains" leading to the holdup tank.

If at any time the activity in the storage tank is long-lived and above the acceptable levels for discharge, these wastes will be placed in appropriate containers properly labelled and suitable for permanent storage. The containers will be stored in the NRC-approved storage area for low level waste located on campus until the activity has decayed sufficiently to permit safe shipment and until sufficient quantity is accumulated to warrant pickup and ultimate disposal by an NRC-approved agency. The procedures for this operation are found in Radiation Control Technique #3 and the UFTR SOP's.

3.3 Utilities and Services

3.3.1 Ventilation (Reference 2)

The reactor room is completely air-conditioned, the air-conditioning unit has a design capacity of 1500 c.f.m., providing approximately 2 air changes per hour with a total air delivery of 6050 c.f.m., at 75°F, dry bulb, and 50 percent relative humidity, during both summer and winter.

All inlet and circulated air is filtered through a roughing filter. The inlet air duct is provided with a motor-operated damper to close the duct whenever the unit fan is not operating.

The room exhaust air, used to ventilate the reactor structure, is passed through a roughing and an absolute filter to an outside stack where it is diluted and released to the atmosphere. Monitoring and maintenance of these filters is covered in the UFTR technical specifications and SOP's.

3.3.2 Fire Protection

Conventional smoke and fire detection equipment is available throughout the reactor building. Three CO_2 extinguishers are found in the reactor cell and the control room. A fire hose and five (5) extinguishers are found outside the control room in the ground foyer. Since the construction materials of the reactor are predominately nonflammable, such as concrete blocks, bricks and floor tile, a serious fire is considered to be very unlikely. An automatic fire alarm system, connected through a computerized system to the Campus Police, provides adequate fire protection capabilities for the entire facility.

3.3.3 Flood Protection

From accumulated experience at the UFTR site, it has been established that no flooding conditions (water intrusion into the cell) will exist in the UFTR site from an accumulated precipitation of 8" of rainfall in a 24-hour period. (3) The most recent heavy rainfall recorded for a 24-hour period occurred in September, 1964, under the effects of Hurricane Dora which caused approximately 11 inches within a 4-day period. Flooding did not occur at the UFTR site or any other area of the University of Florida campus, while flooding was reported in the Southwest area of greater Gainesville. The drainage system has been improved since that incident; therefore, it is estimated that no major flood will occur in the city of Gainesville or anywhere near the UFTR site. In the unlikely event that the U.S. Weather Bureau gives a significant probability of a hurricane or other severe storm to produce an accumulated rainfall of more than 8 inches of rain in a 24-hour period, the UFTR personnel shall proceed according to the UFTR SOP-B.4, "Emergency Flood Procedure." (3)

4. REACTOR

4.1 Summary Description

4.1.1 General Reactor System Description

The UFTR is a research and training reactor of the general type known as the Argonaut with modifications made by the General Nuclear Engineering Corporation of Dunedin, Florida, to adapt it to a university program by improving shielding and minimizing the possibility of an accident. The UFTR has been operational since May, 1959. Originally licensed for operation up to 10 Kw, the UFTR is currently licensed for operation at 100 Kw (thermal) under License Number R-56, Amendment 8, effective January 28, 1964. Similar operating reactors are located at the Universities of Washington and California (UCLA), at Virginia Polytechnic Institute and in the United Kingdom. Other similar facilities include the MTR, BSTR, Borax I, II, and III. (5)

The UFTR is heterogeneous in design, using 93 percent enriched uranium-aluminum fuel elements. Cutaway longitudinal and transverse sectional views of the UFTR including shielding are shown in Figures 4-1 and 4-2. A horizontal section of the UFTR at the beam tube level is shown in Figure 4-3. An isometric of the UFTR with shielding removed is shown in Figure 4-4. These four figures serve to indicate how the reactor is generally set up but especially the diverse experimental applications available with the UFTR. An isometric diagram of UFTR components including control rod drive system and control rod shrouds, overall fuel box arrangement with covers, deflectors and shield plugs, coolant lines, graphite stringers, and shield test tank is presented in Figure 4-5. Figure 4-5 provides an excellent description of the interconnection of the various basic components that constitute the UFTR.

As indicated, the thermal power level of the UFTR reactor is currently limited to 100 Kw (thermal) with water used as a coolant and also as part of the moderator; the remainder of the moderator consists of graphite blocks which surround the boxes containing the fuel plates and the water moderator as indicated in Figures 4-1 through 4-5. The fuel is contained in MTR-type plates assembled in bundles. Each bundle is composed of 11 fuel plates, each of which is a sandwich of aluminum clad over a uranium-aluminum alloy "meat."

The reactor core has a two-slab geometry and it is presently composed of 21 fuel bundles plus three (3) dummy bundles arranged in six water filled aluminum boxes which are surrounded by reactor grade graphite.

The primary coolant (demineralized water) is pumped upward over the fuel plates and then fed by gravity through the side orifices to the heat exchanger where the primary coolant transfers the heat from the reactor. The heat is removed by the secondary coolant through the heat exchanger to the storm sewer.

The reactor is equipped with four control blades (3-safety and 1-regulating) of the swing-arm type consisting of four cadmium vanes protected by magnesium shrouds as shown in Figure 4-5. The control blades operate by moving in a vertical arc within the spaces provided between the fuel boxes. These blades are moved in and out by mechanical drives or they may also be

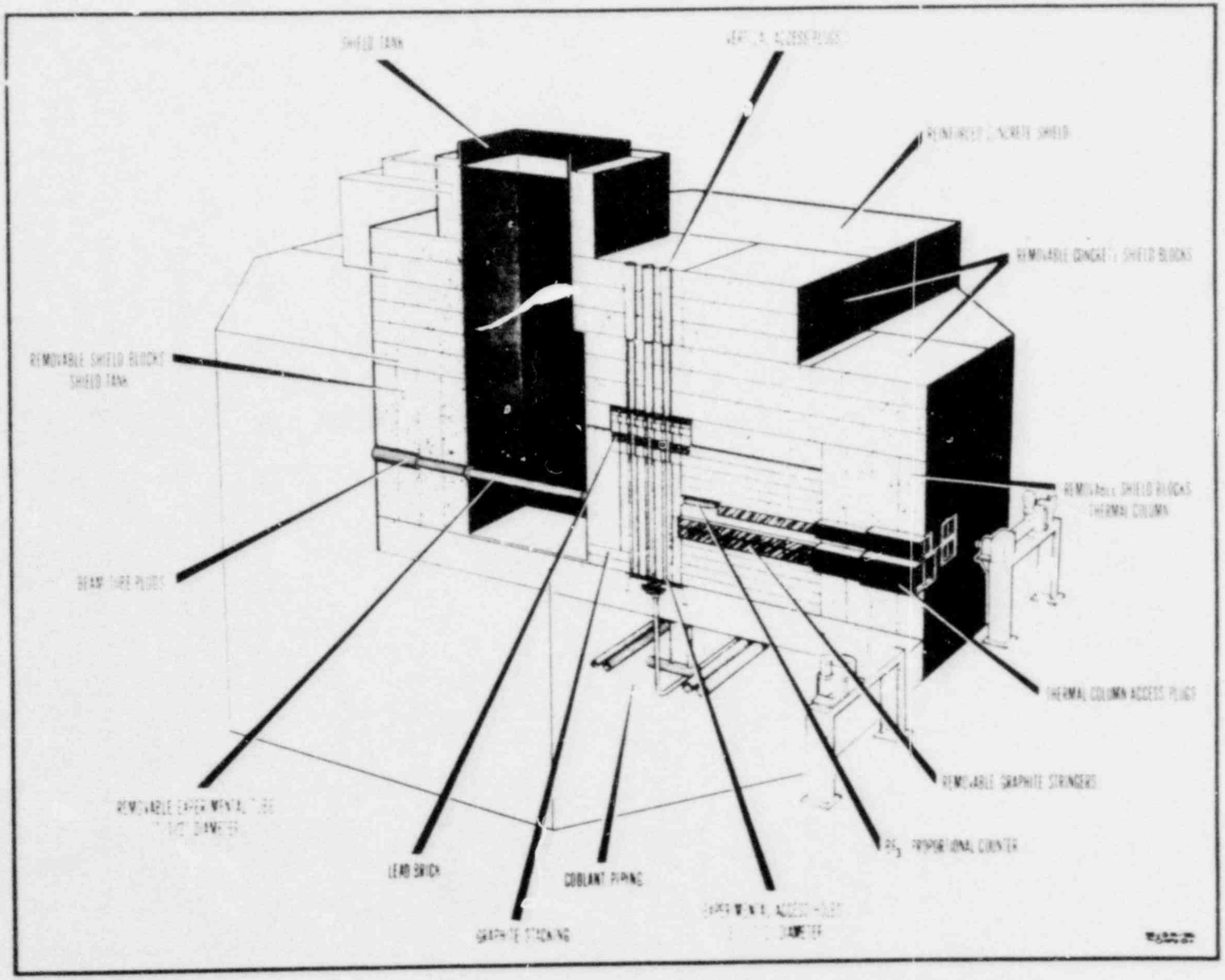


Figure 4-1. Longitudinal Section Diagram of UFTR.

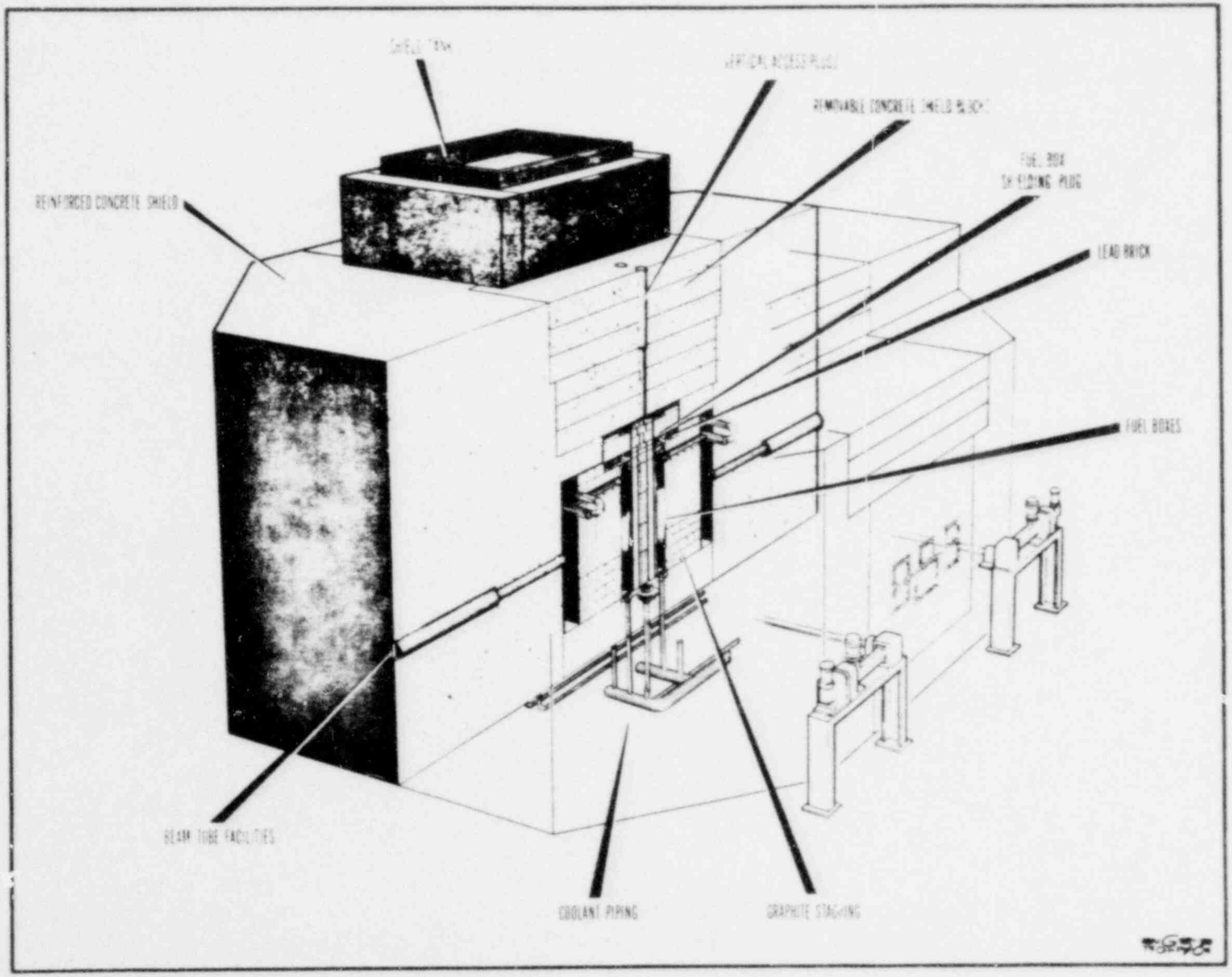


Figure 4-2. Transverse Section Through the UFTR Core Center.

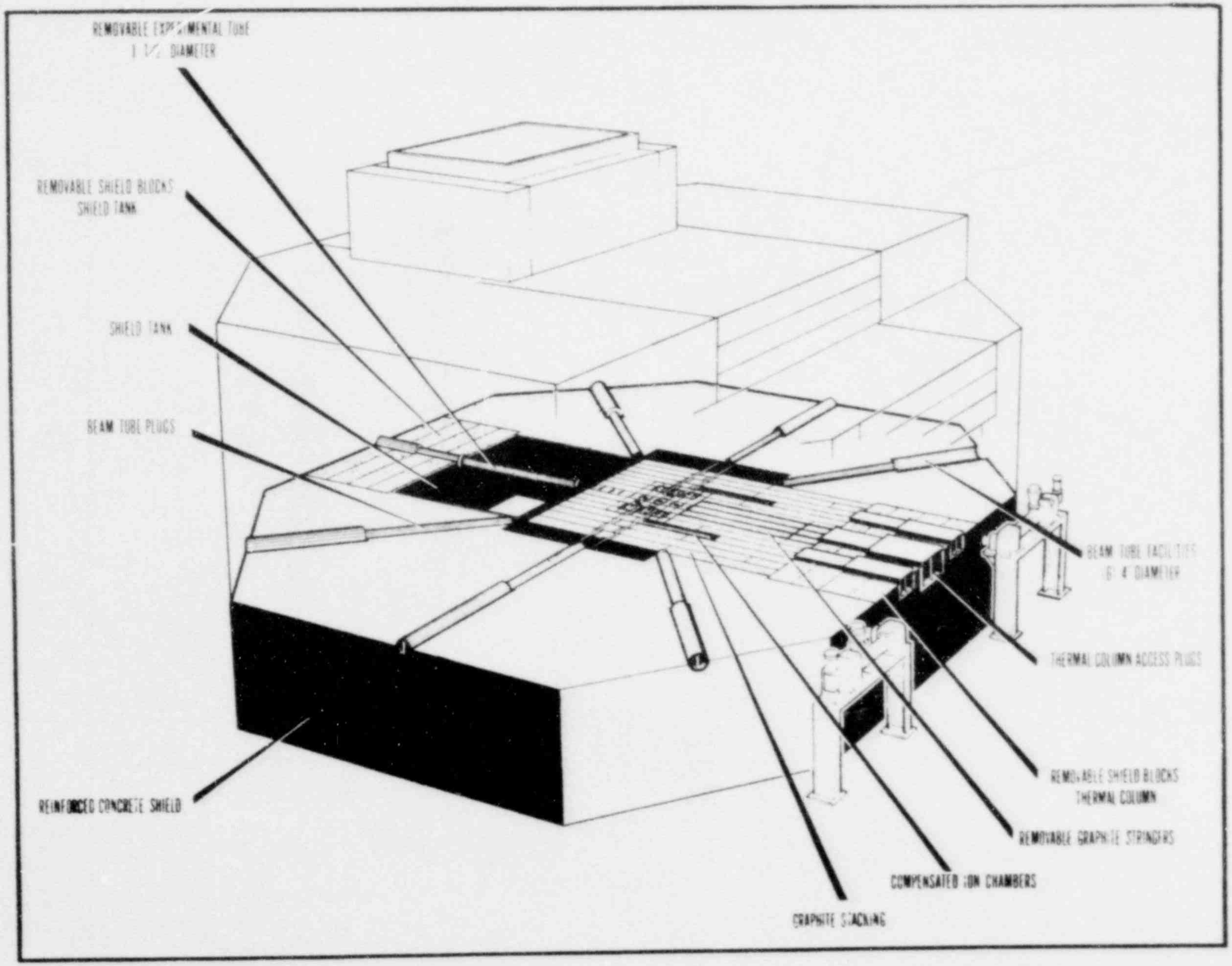


Figure 4-3. Horizontal Section Diagram of UFTR at Beam Tube Level.

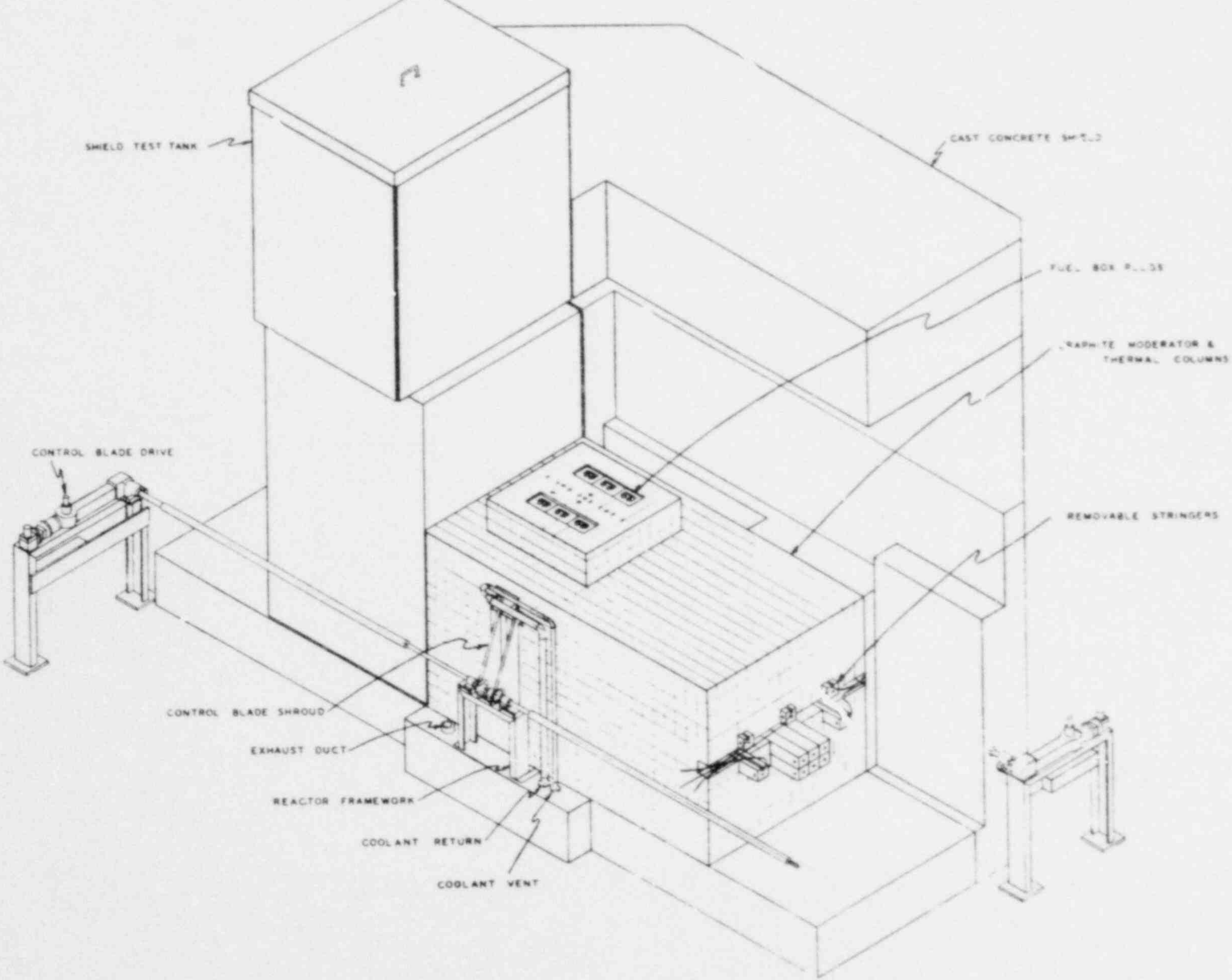


Figure 4-4. Isometric Sketch of the UFTR With Shielding Removed.

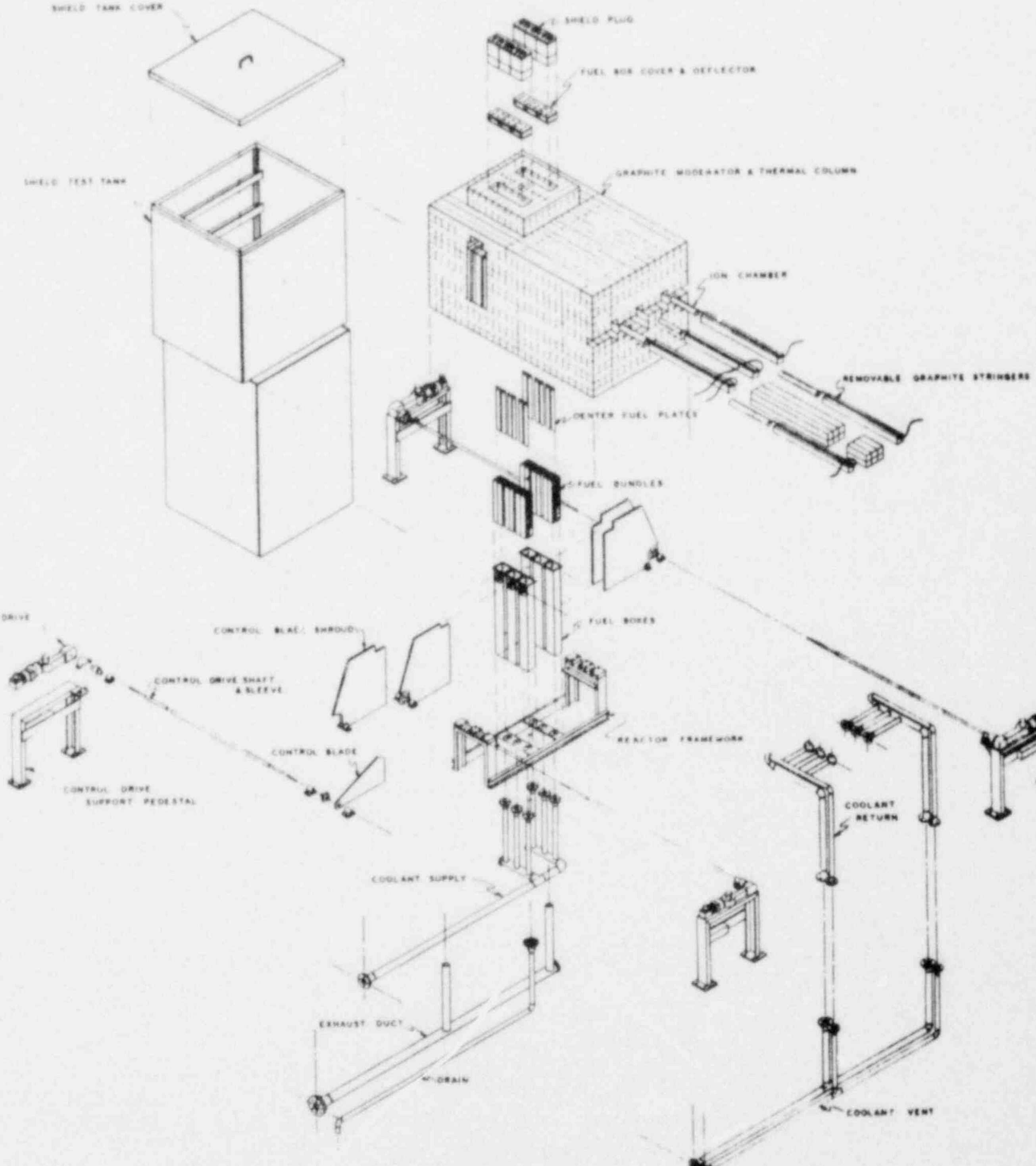


Figure 4-5. Isometric Diagram of UFTR Components.

disconnected by means of electromagnetic clutches and allowed to fall by gravity into the reactor. The drives which are connected to the blades by means of long shafts are located outside the reactor shield for accessibility as shown in Figure 4-4.

The maximum reactivity addition rate of the safety and control blades is limited to 0.06% $\Delta k/k$/sec by system design to prevent sudden large reactivity increases. Such a limitation insures the integrity of the fuel and other systems; essentially this limit assures that there can be no chance of prompt critical operation.

The nuclear design of the core will insure that the combined response of all reactivity coefficients and an increase in reactor power yields a net decrease in reactivity, as discussed in the safety analysis of Chapter 15.

The operation of the reactor is monitored and controlled from a desk-type console. The console displays all the pertinent data such as control blade positions, reactor period, reactor power level, coolant temperature and other information necessary for safe operation and control of the UFTR.

Reactor instrumentation consists of three neutron flux channels, control blade position indicators, the electrical interlock system, control blade selector and drive switches, and the reactor scram circuitry. The reactor instrumentation is discussed in Section 7.1, Instrumentation and Controls.

The experimental facilities in the UFTR include:

1. Sixteen (16) vertical foil slots placed at intervals in the graphite between the fuel compartments, each are 3/8 in. by 1 in.;

2. Three (3) vertical experimental holes of 1-1/2 in. in diameter located centrally with respect to the six fuel compartments;

3. Five (5) vertical holes 4 in. by 4 in.;

4. A thermal column having 4 in. by 4 in. removable stringers;

5. A shield tank placed against the west face of the reactor opposite the thermal column;

6. Six (6) horizontal openings 4 in. in diameter are found on the center plane of the reactor;

7. A horizontal throughport which is an approximately 1.88 in., ID pipe with 20 ft. length running east-west across the reactor.

Shield plugs are normally inserted in these facilities except where an experiment or test requires otherwise.

The core mechanical design is presented in Section 4.2; the core nuclear design is summarized in Section 4.3; key thermal and hydraulic design considerations are presented in Section 4.4.

4.1.2 Design and Performance Characteristics

The principal design and performance characteristics for the UFTR reactor are summarized in Table 4-1. The UFTR self-limits the maximum power and energy release in an accidental nuclear excursion or loss of coolant accident by means of either the negative moderator void coefficient or the negative temperature coefficient. These inherent nuclear control features are effective even if the control rods or the instrumentation which is part of the reactor protection system fail, or if the operator mistakenly or deliberately violates established operating procedures and rules. The worst situation occurs if a large amount of reactivity is added suddenly. The maximum excess reactivity for the UFTR is limited with the present fuel loading to approximately 2.3% $\Delta k/k$. Calculations made by Listing (22) have shown that the necessary reactivity required to raise the temperature of the fuel plates to the melting point is about 2.4% $\Delta k/k$; therefore, there is no danger of fission product release or damage to the structural integrity of the reactor due to a large addition of reactivity into the system. Reactivity accidents are discussed further in Chapter 15.

Reactivity control is provided by the three control blades and one regulating blade described in Table 4-1. Table 4-1 also shows the corresponding reactivity worths for each blade, along with the maximum allowed reactivity addition rate for the UFTR. The shutdown margin available with the most reactive blade out is approximately 2.7% $\Delta k/k$. The control blades are "fail-safe" in the sense that they will drop into the core by gravity in the event of a loss of electrical power. The reactor protection system provides a series of control blade interlocks and reactor scrams preventing the occurrence of situations which may endanger the integrity of the reactor system and assuring its safe operation as discussed in Chapter 7 - Instrumentation and Control.

Temperature limits are not considered to present any problems during reactor operation at 100 Kw (thermal). At 100 Kw (thermal), the equilibrium inlet temperature is found to be 86 $\pm$ 2°F and the equilibrium outlet temperature is 103 $\pm$ 2°F when using the main secondary cooling system and increased by ~40°F when the back-up secondary cooling system is used.

4.1.3 Shielding

Biological shielding is provided around the UFTR to minimize the exposure to any individual working with the reactor to levels as low as reasonably achievable (ALARA) and as specified by 10 CFR 20. The biological shielding is made of cast-in-place concrete with sections of barytes concrete carefully located to reduce the overall shield thickness while assuring its affectiveness. As specified in Table 4-1, the shielding consists of the following:

- 6 ft. cast-in-place barytes concrete found at the center sides;
- 6 ft. 9 in. cast-in-place barytes concrete at the end sides; in the middle are barytes concrete blocks;
- 5 ft. 10 in. barytes concrete blocks at the top; and
- 3 ft. 4 in. barytes concrete blocks at the end.

Access to the ends and top of the reactor is provided by removal of ordinary concrete blocks cast to fit the openings. These blocks, weighing up to 4,500 lbs. each, have pick-up plugs so that they may be handled by means of the overhead bridge crane. The arrangement of these movable blocks is illustrated in the section views of the UFTR shown in Figures 4-1 through 4-3.

TABLE 4-1

PRESENT UFTR CHARACTERISTICS

General Features

Reactor Type	Heterogeneous, Thermal
Licensed Rated Power Level	100 Kw
Maximum thermal flux level in center vertical port at 100 Kw	1.8×10^{12} n/cm^{2}sec
Excess reactivity (at 72°F)	~1.17% Δk/k
Clean, cold critical mass	3.07 kg U-235
Effective prompt neutron lifetime	2.8×10^{-4} sec
Uniform water void coefficient	-0.2% Δk/k/% voids
Temperature coefficient	-0.3×10^{-4}% Δk/k per °F
U-235 mass coefficient	0.4% Δk/% U-235
Startup source	Sb-Be ≤ 25 curies or PuBe ≤ 10 curies
Reflector	graphite (1.6 gm/cm^3)
Moderator	H_2O and graphite

Fuel Plates

Fuel	93% enriched, U-Al
Fuel loading	3354.61 gm U-235
Plate thickness	0.070 in.
Plate width	2.845 in.
Plate length	25.625 in.
Water channel width	0.137 in.
Aluminum to water ratio (volume)	0.49
"Meat" composition	14.05 w/o U

Coolant

Type	H_2O
Flow (at 100 Kw)	40.5 gpm
Equilibrium Inlet Temperature (100 Kw)	86 ' 2" F
Equilibrium Outlet Temperature (100 Kw)	103 ' 2"F

Control Blades

Type	Cd, swinging vane, gravity fall
Number	3 safety; 1 regulating
Insertion time	≤ 1 sec
Removal time	100 sec (minimum)
Blade worth, safeties	Safety #1 ~ 1.5% k
	Safety #2 ~ 1.3% k
	Safety #3 ~ 2.1% k
Blade worth, regulating	Reg. Rod ~ 0.91% k
Reactivity addition rate, maximum allowed	0.06% Δk/k/sec

Shield (concrete)

Sides, center	6 ft., cast, barytes
Sides, ends	6 ft. 9 in., cast, barytes
Middle	Barytes concrete blocks
Top	5 ft. 10 in.
End	3 ft. 4 in.

Experimental Facilities

Thermal column, horizontal	60 in. x 60 in. x 56 in. high
Thermal column, vertical	2 ft. diam. x 5 ft.; H_2O or D_2O
Shield test tank	5 ft. x 5 ft. x 14 ft. high
Experimental holes	5 vertical, 4 in. x 4 in.
	3 vertical, 1 1-1/2 in.
Foil slots	16 vertical, 3/8 in. x 10 in.

4.1.4 Experimental Facilities and Conduits

The experimental exposure facilities and instrumentation ports are described in Table 4-1. The overall physical arrangement of these exposure facilities is shown in Figure 4-3. More detailed sketches of the size and orientation of these exposure facilities are presented in Figure 4-6 for the center vertical port and horizontal throughport and in Figure 4-7 for the other major experimental exposure facilities.

System vertical foil slots, 3/8 in. by 1 in. are placed at intervals in the graphite between the fuel compartments and are used for flux mapping. The foils can be installed by lifting off the top shield, placing the foil holders, replacing part of the shield as deemed necessary for irradiation, and removing it to recover the foils. Shield removal can be accomplished by the use of the bridge crane.

There are three (3) vertical experimental holes, 2", 1-3/4", 1-1/2" in diameter, which are centrally located with respect to the six fuel compartments. The maximum neutron flux is available in the vicinity of these ports; therefore, they may be used for irradiating samples or for installing an oscillator. Mated openings are provided in the upper shield for convenience in the use of these holes.

A thermal column is provided in the east face of the reactor having four 4 in. by 4 in. removable stringers. The horizontal thermal column is 60 in. x 60 in. x 56 in. high; the vertical thermal column comprises an area 2 ft. in diameter by 6 ft. long, filled with H_2O or D_2O as necessary for experimental purposes.

Six other horizontal openings, 4 in. in diameter are located in the center plane of the reactor as shown in Figure 4-7. These horizontal holes (or ports) may be fitted with collimators to allow neutron beams to escape or with other equipment for the irradiation of special samples.

A water tank is placed against the west face of the reactor opposite the thermal column and is shielded on the outer three sides by concrete. This 5 ft. by 5 ft. x 14 ft. high shield tank can be used to perform shielding experiments or for the irradiation of large objects. A horizontal aluminum pipe passes through the shield tank outer wall and is welded to the inner wall; it is provided to allow the extraction of a neutron beam to the reactor west face. The tube allows the insertion of the east-west throughport (EWTP). The EWTP, or horizontal throughport, is a horizontal tube ~1.88 in ID x 20 ft. in length. If the shield tank is not needed for experiments, it can be removed after draining by lifting it out with the crane and other equipment installed in that area. (5)

4.2 Fuel System Design

The reactor core has a two slab geometry; it is presently composed of 21 fuel bundles and three (3) dummy bundles (labeled "D") arranged in six (6) water filled aluminum boxes, surrounded by reactor grade graphite as shown in Figure 4-8. Room is provided so that up to 24 fuel bundles can be inserted into the UFTR reactor core illustrated in Figure 4-8.

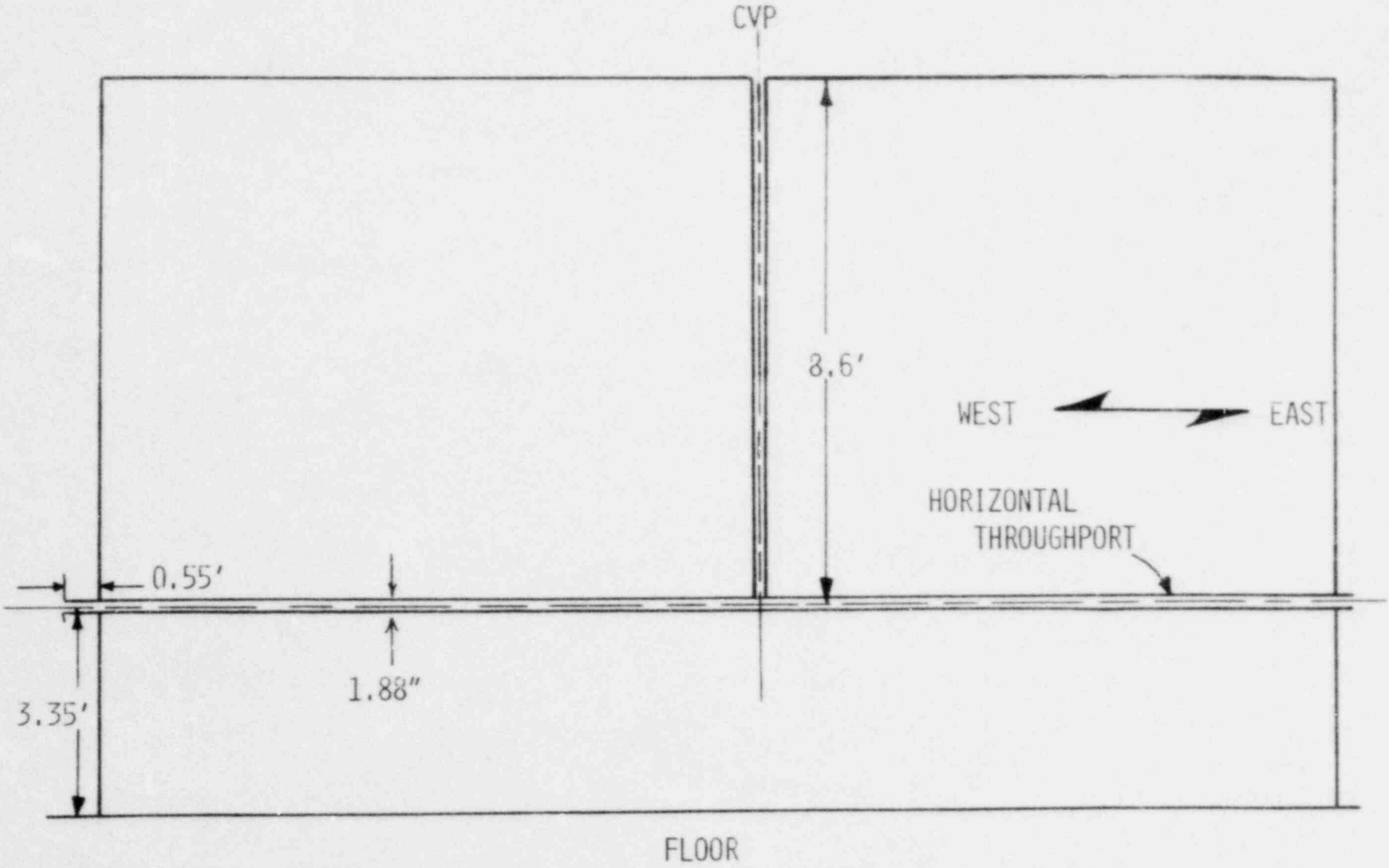


Figure 4-6. UFTR Cross Section Showing Center Vertical Port (CVP) and East-West Through Port (EWTP) Arrangement with Dimensions. (Not to scale)

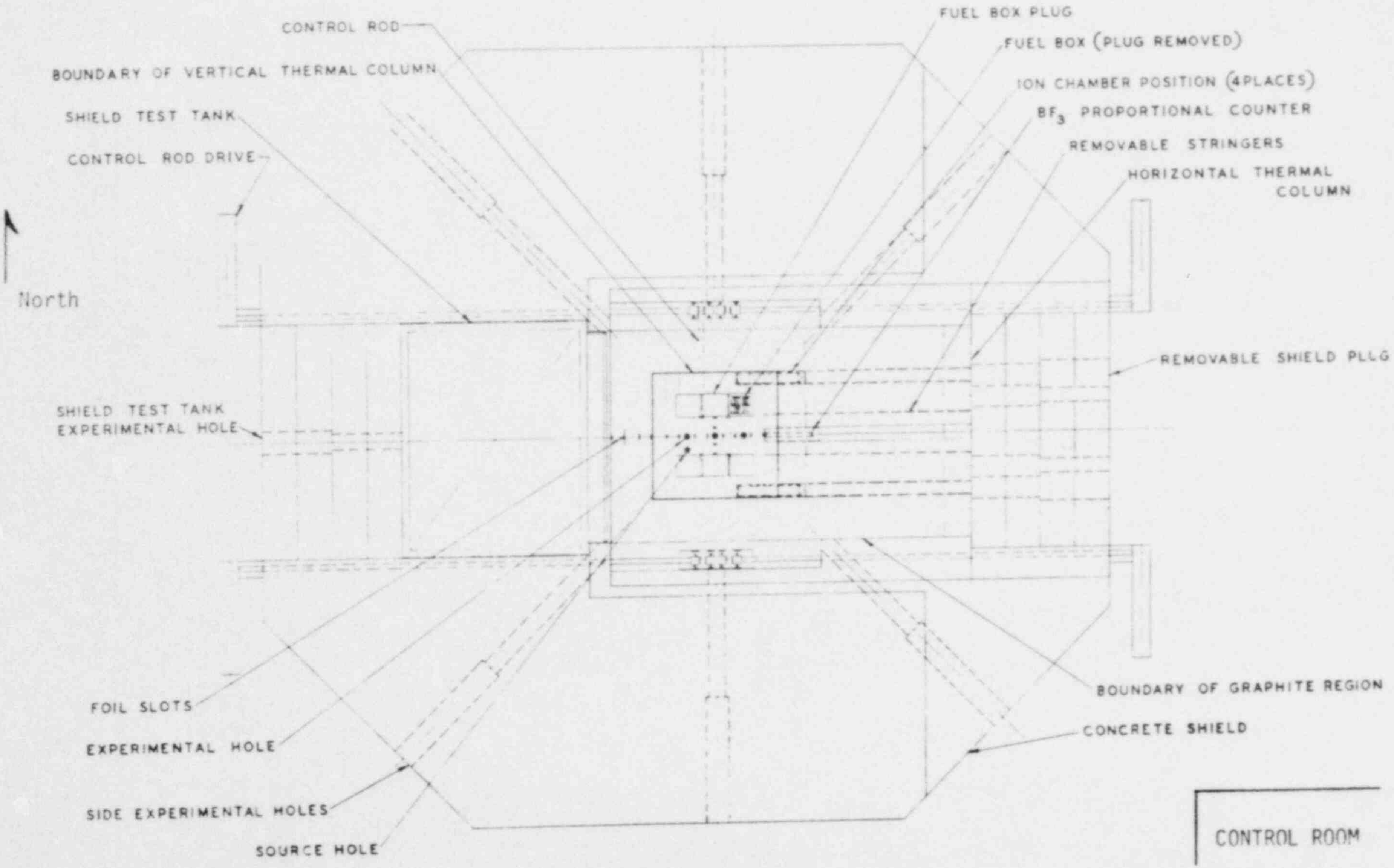
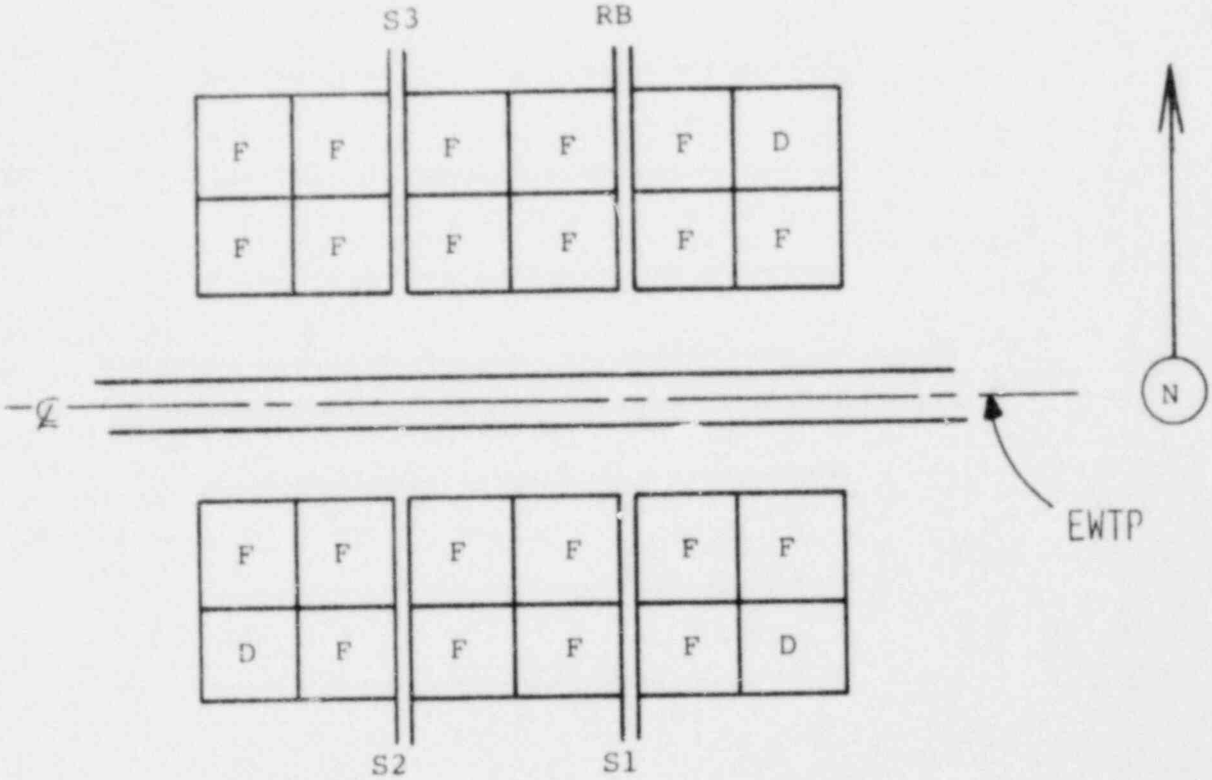


Figure 4-7. Geometric Arrangement of Major UFTR Experimental Facilities.



NOMENCLATURE:

F = ACTIVE FUEL BUNDLE
D = DUMMY FUEL BUNDLE
RB = REGULATING BLADE

S1 = SAFETY BLADE 1
S2 = SAFETY BLADE 2
S3 = SAFETY BLADE 3

EWTP = EAST-WEST THROUGHPORT

Figure 4-8. Vertical Section View of UFTR Core Illustrating Fuel and Fuel Box Arrangement.

The arrangement of the fuel bundles in the fuel boxes is illustrated in the isometric of the fuel boxes shown in Figure 4-9. The coolant inlet and outlet positions are also shown in Figure 4-9 along with the positions of the shield plug and fuel support components for each fuel box.

The fuel elements are fabricated from 93 percent enriched uranium-aluminum alloy, and each bundle is composed of 11 fuel plates. Each plate is a sandwich of aluminum cladding over a uranium-aluminum alloy "meat" as illustrated in Figures 4-10 and 4-11. The UFTR fuel is of the MTR type with the following characteristics as indicated in the fuel plate description in Figure 4-10.

A sheet of 0.04 in. thick uranium-aluminum alloy sandwich is placed between the 0.015 in. thick aluminum clad plates. Each plate is 25.625 in. long, 2.875 in width, having a total thickness of 0.07 with approximately 14.5 grams of U-235. The detailed fuel cell geometry is presented in Figure 4-12.

Each fuel bundle is made up of 11 fuel plates as shown in Figure 4-11. A 0.137 in. spacing is provided between fuel plates for coolant flow as indicated in the detailed cell geometry shown in Figure 4-12. In order to achieve a particular desired excess reactivity capability in the core, dummy aluminum bundles are placed in the configuration as illustrated in Figure 4-8. The UFTR is presently licensed to have up to 2.3% $\Delta k/k$ excess reactivity, which would require approximately a full fuel load with no dummy elements; the actual available excess reactivity in the present configuration is about 1% $\Delta k/k$.

4.3 Nuclear Designs

4.3.1 Flux Distributions

The principal nuclear parameters for the UFTR are listed in Table 4-1 in Section 4.1.2. The experimentally obtained neutron flux distribution for the UFTR during 100 Kw (thermal) operation is shown in Figure 4-13 which includes both thermal and epithermal fluxes. (5) These flux distributions are currently being updated with new experimental determinations. This work is not yet complete but will be included as an addendum to this report when available. It is not expected that the flux distributions will have changed significantly.

Several studies have been carried out concerning the operation of the UFTR at 500 Kw (thermal) in anticipation of eventually applying for a license to increase the licensed power level of the UFTR to 500 Kw. Neutronic analyses of the UFTR were carried out by Wagner in one of these studies. As stated by Wagner in calculating the neutron flux distribution in the UFTR, the fuel, water, and graphite are assumed to be at an average coolant temperature of 133°F, which approximates reactor conditions at prolonged 100 Kw (thermal) power operation. The thermal expansion of the fuel can be neglected, and the total assumed fuel loading was 3480 grams U-235. For simplicity, all materials other than fuel plates and boxes, water, and graphite were neglected. The geometry and the critical dimensions for the fuel plate unit cell and the quarter fuel box assembly are indicated in the core section sketches--Figures 4-14A and 4-14B.

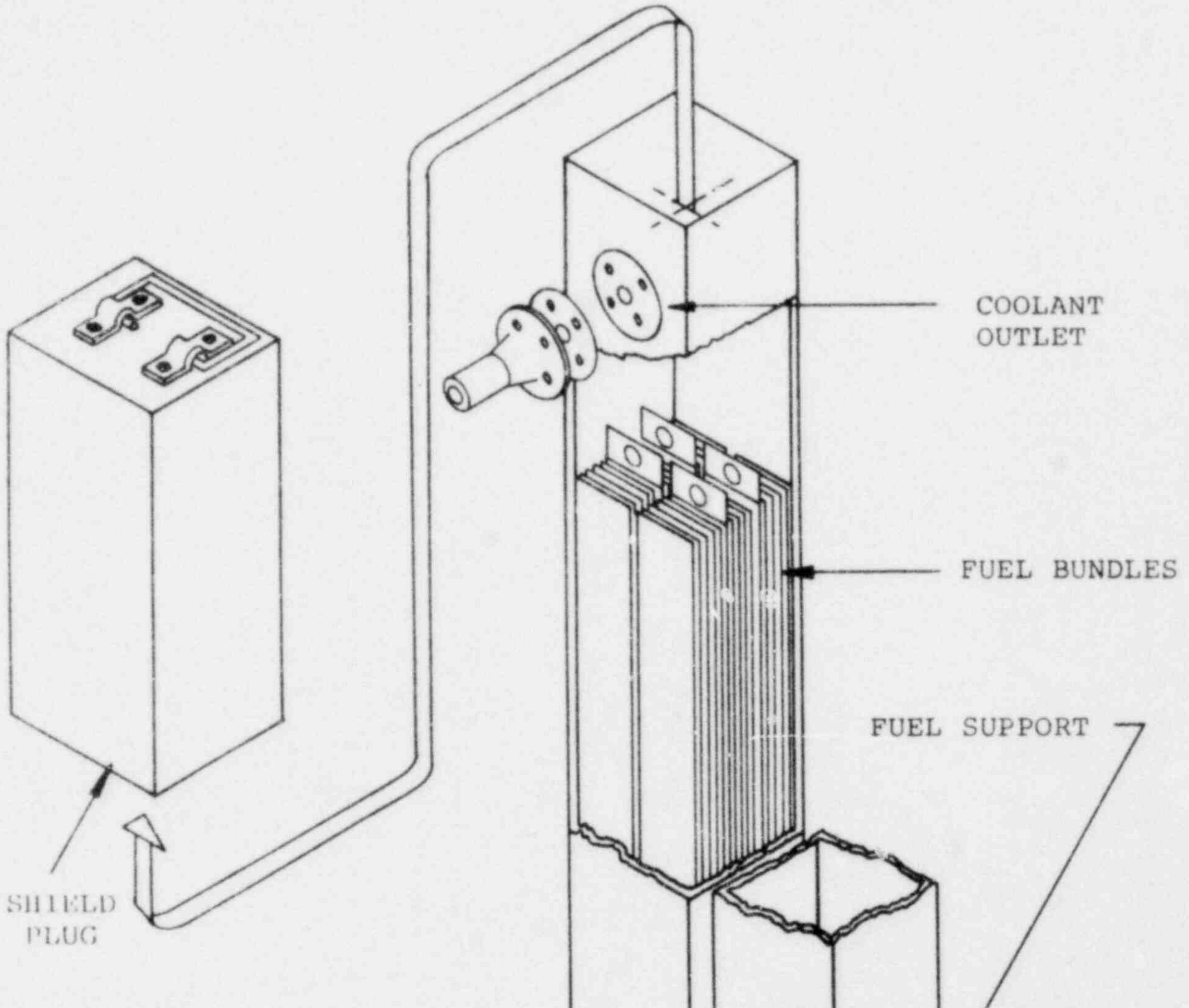


Figure 4-9. ISOMETRIC OF UFTR FUEL BOXES.

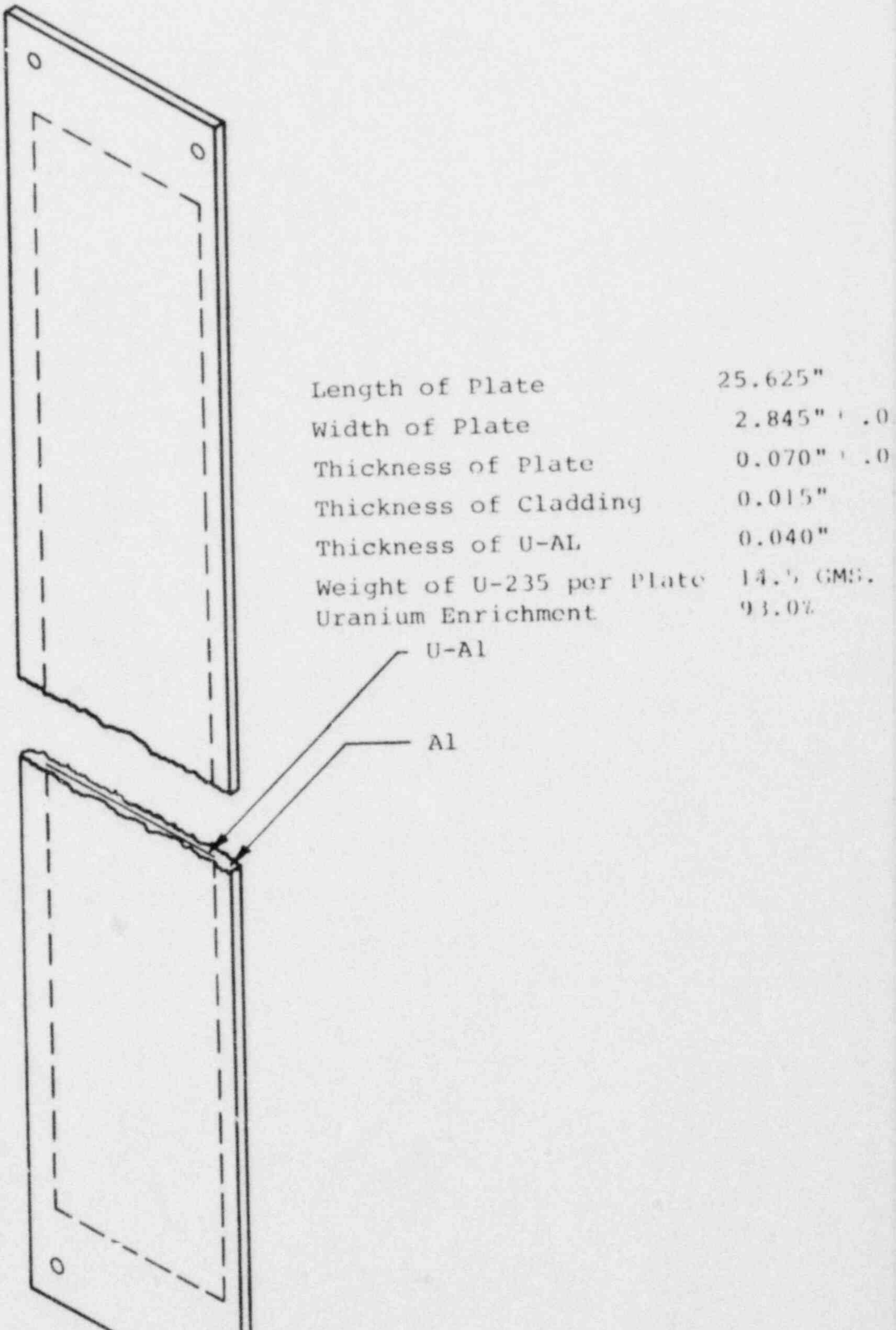


Figure 4-10. Schematic Showing UFTR Fuel Plate Geometry.

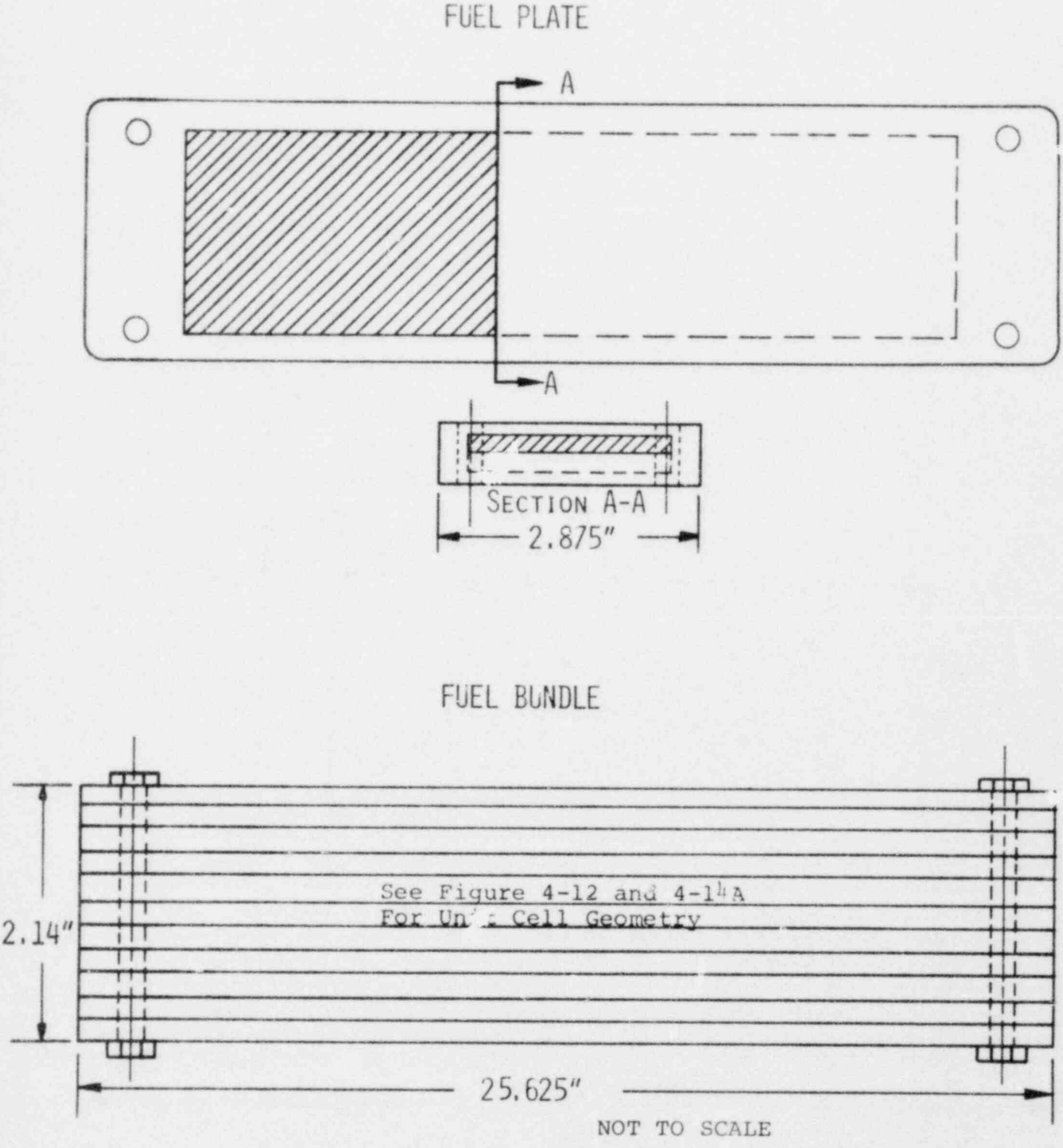


Figure 4-11. UFTR Fuel Plate and Fuel Bundle Geometric Arrangement.

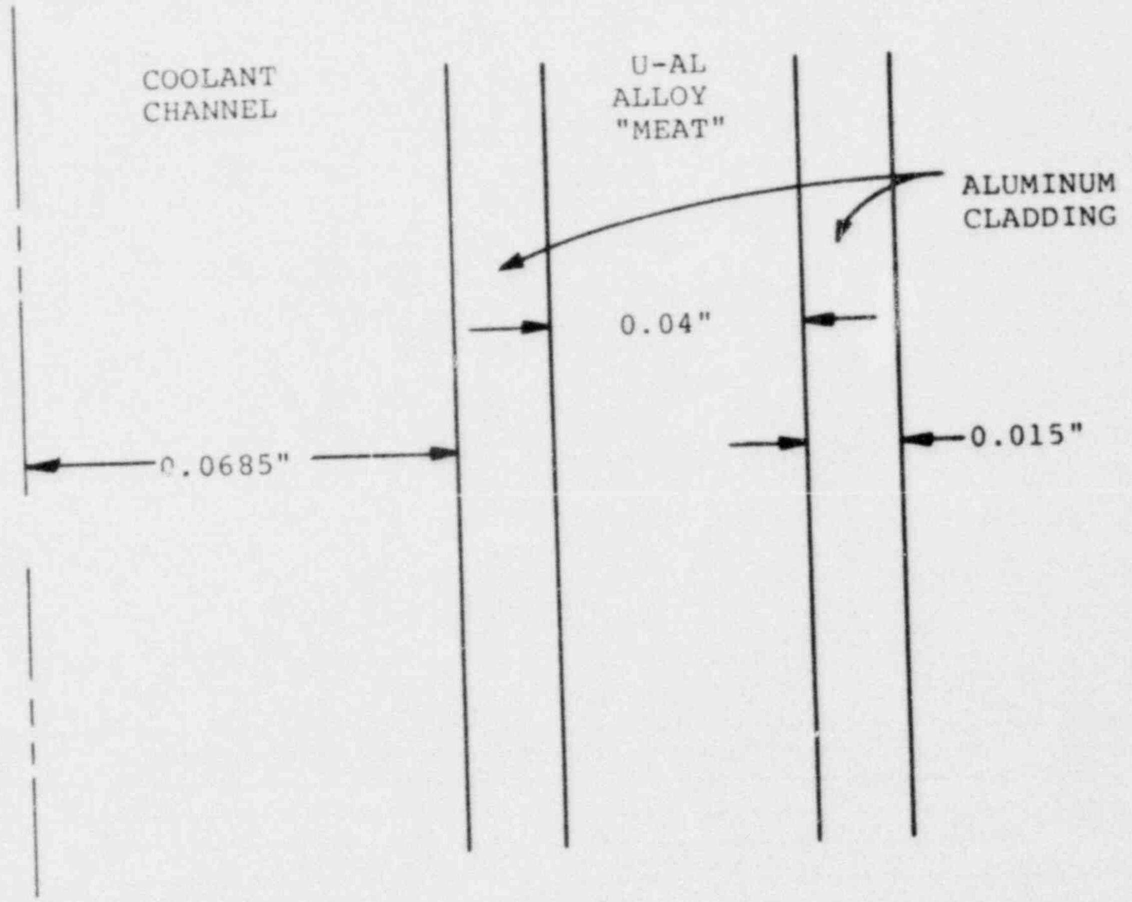


Figure 4-12. Fuel Plate/Coolant Channel Enlargement Showing UFTR Fuel Cell Arrangement Detail.

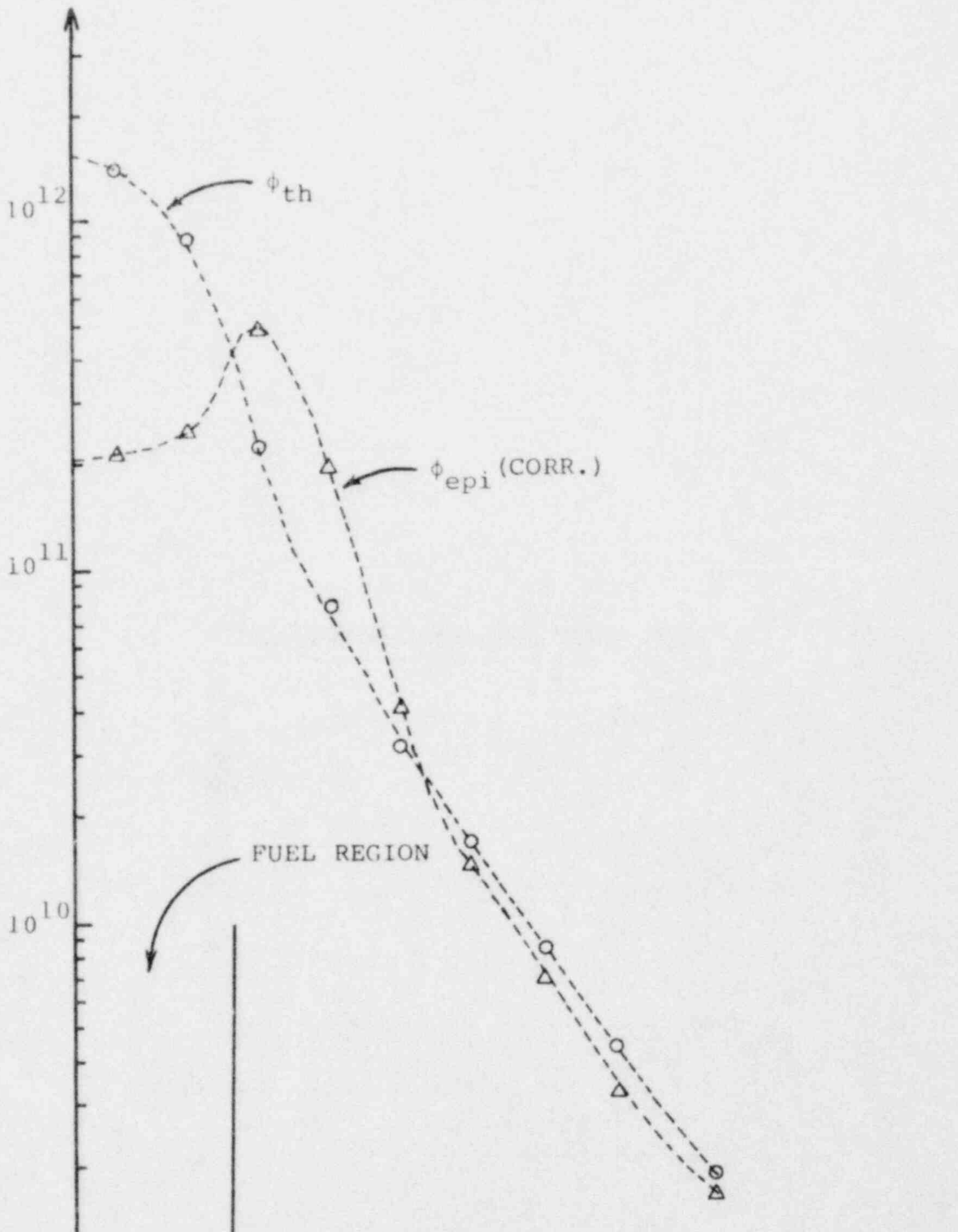


Figure 4-13. UFTR Absolute Flux Measurements Results in CVP (Gold Foil). (5)

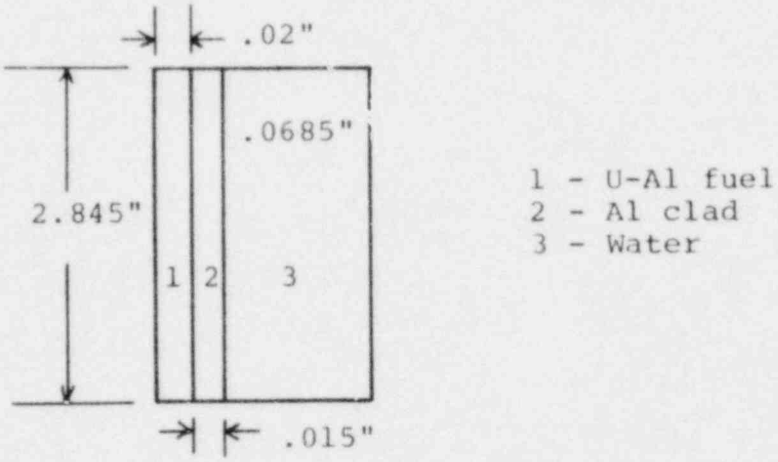


Figure 4-14A. UFTR Fuel Plate Unit Cell

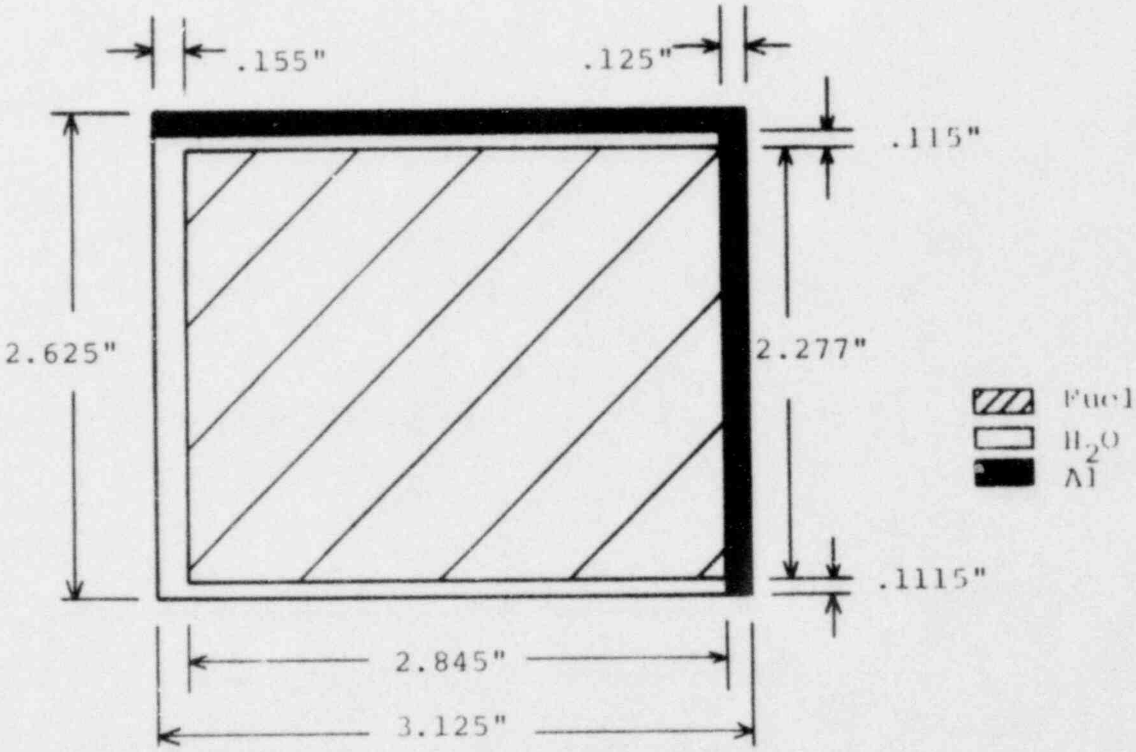


Figure 4-14B. UFTR Quarter Fuel Box Unit Assembly.

Thermal and fast group constants for the fuel and graphite regions were calculated by the use of computer codes. BRT-1, Battelle-Revised-Thermos (24) was used to generate thermal neutron spectra and correspondingly weighted thermal group constants; PHROG (25) was used to generate fast neutron spectra and the corresponding average fast multigroup constants. These calculations began with the determination of cross section data for the fuel plate unit cell model comprised of uranium-aluminum fuel, aluminum clad, and water presented in Figure 4-14A. The results of these calculations were then used as the microscopic input data needed to calculate the parameters for a quarter fuel box model (see Figure 4-14B) containing 11 fuel plate unit cells and other structural material. After determining all the necessary parameters, the UFTR core region was represented and modeled as shown in the schematic drawing in Figure 4-15.

This UFTR model shown in Figure 4-15 was used in a four-group diffusion theory calculation performed using CORA, a multigroup diffusion theory code for one dimensional reactor analysis. (26) The flux distributions presented in Figures 4-16 and 4-17 were obtained by Wagner (23) from these CORA calculations. Figure 4-16 shows the normalized "flux per watt versus distance from core centerline" distribution along the North-South UFTR direction; Figure 4-17 shows the corresponding normalized distribution along the East-West UFTR direction. Average flux data for the UFTR was obtained from these distributions and is summarized in Table 4-2 for the fueled regions as well as for the total reactor. Table 4-2 also contains the peak-to-average flux ratios for the fueled regions and for the total reactor. The peak-to-average flux ratio (peaking factor) is less than 1.12 for all groups in the fueled power-producing regions. It is only over the total reactor that the peak-to-average flux ratios exceed 2 or more.

4.3.2 Fission Product Poisoning Considerations

The fission product poisoning effects during hypothetical operation of the UFTR at 500 Kw (thermal) have also been studied by Mr. Otaduy. (27) Although this is five times the currently licensed power level, some points in this analysis are worthy of inclusion in this section. To perform this study, several parameters had to be determined using a two-dimensional, diffusion theory calculation of the four energy group parameters. (22) The EXTERMINATOR-2 computer code (28) was used to model the two-dimensional UFTR core shown in Figure 4-18. The required fast and thermal four-group neutronics constants were taken from Wagner's work. (23) Results obtained from the EXTERMINATOR calculations were comparable with those obtained by Wagner, with differences attributed to the total mass of U-235 considered in each model.

The two isotopes Xe-135 and Sm-149 are usually considered the most important poisons in thermal reactors since they have very large cross sections and also characteristically reach a saturation level with reactor operation, while the bulk of the fission products are non-saturating and build up with burnup; therefore, these two isotopes were treated separately in the study by Otaduy. (27) The detailed study of the Xe-135 and Sm-149, as well as the gross fission product behavior performed by Otaduy, models the UFTR as a one-group, one-region homogeneous reactor. The homogenized core parameters and the necessary constants used for Otaduy's analysis are presented in Table 4-3 and Table 4-4 respectively to augment the basic information about the UFTR nuclear data contained in this report.

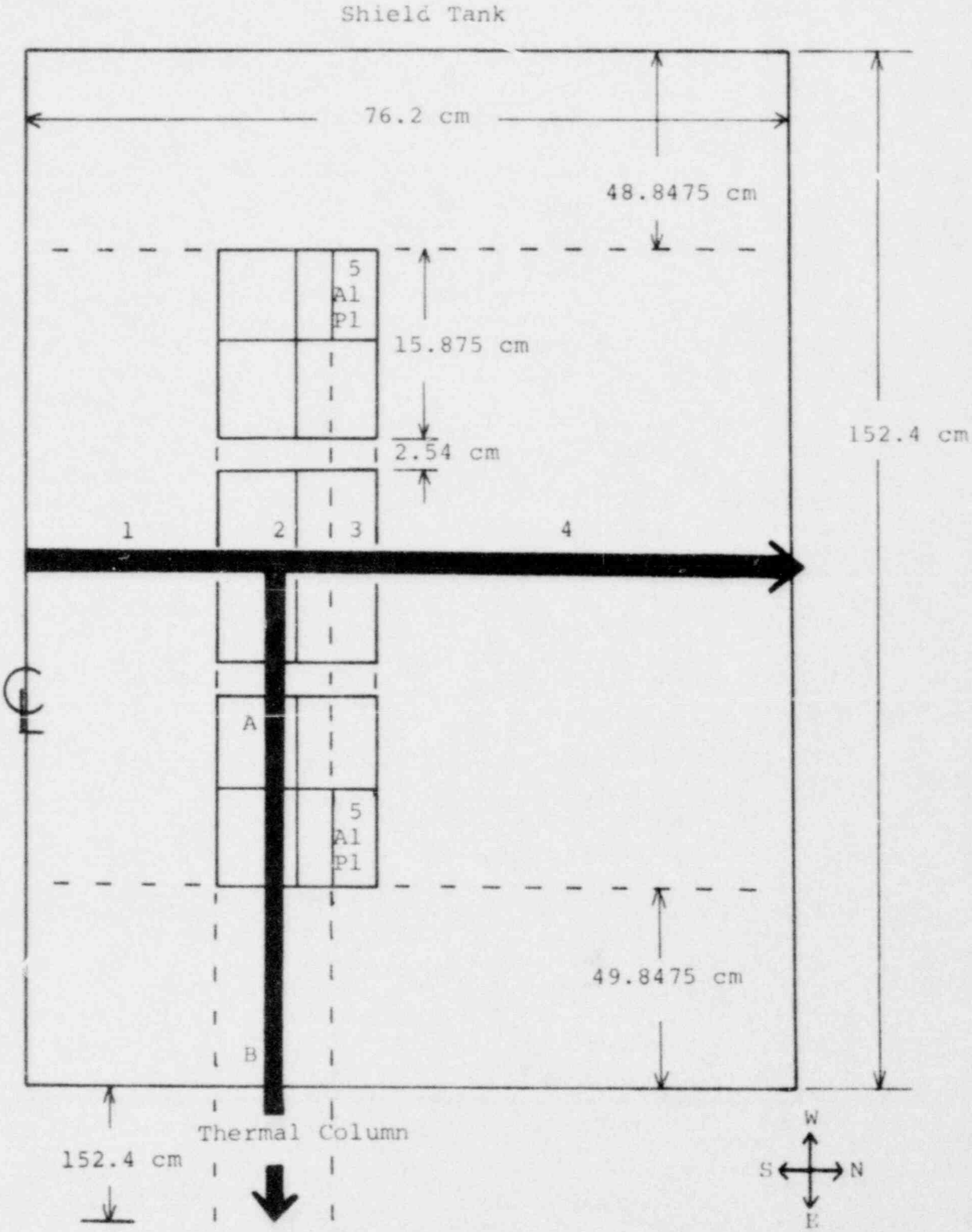


Figure 4-15. Model of UFTR Core Region (Top View) Used for CORA Calculation. (23)

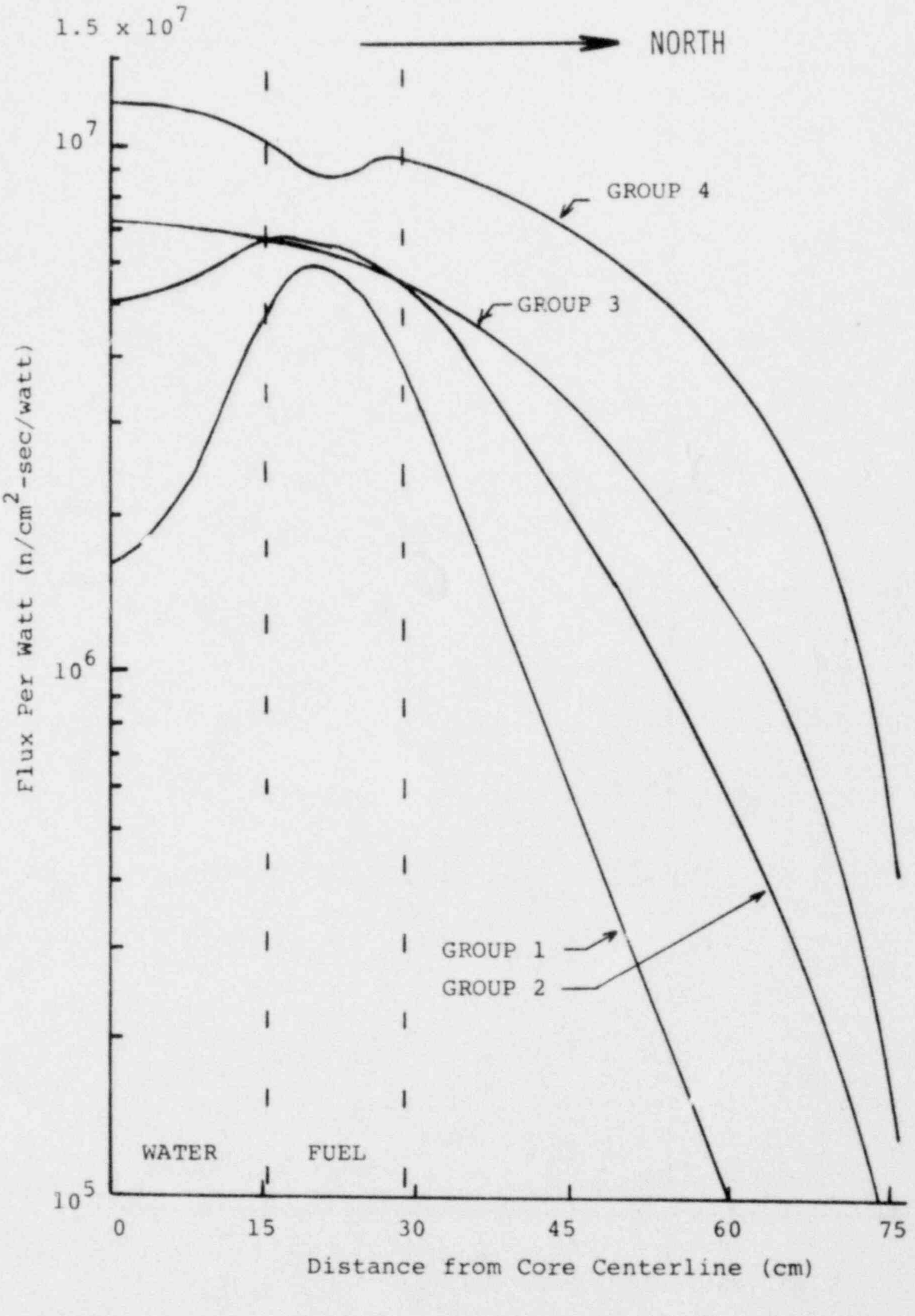


Figure 4-16. Group-Dependent Fluxes Along the North-South Direction--CORA Code Calculations.(23)

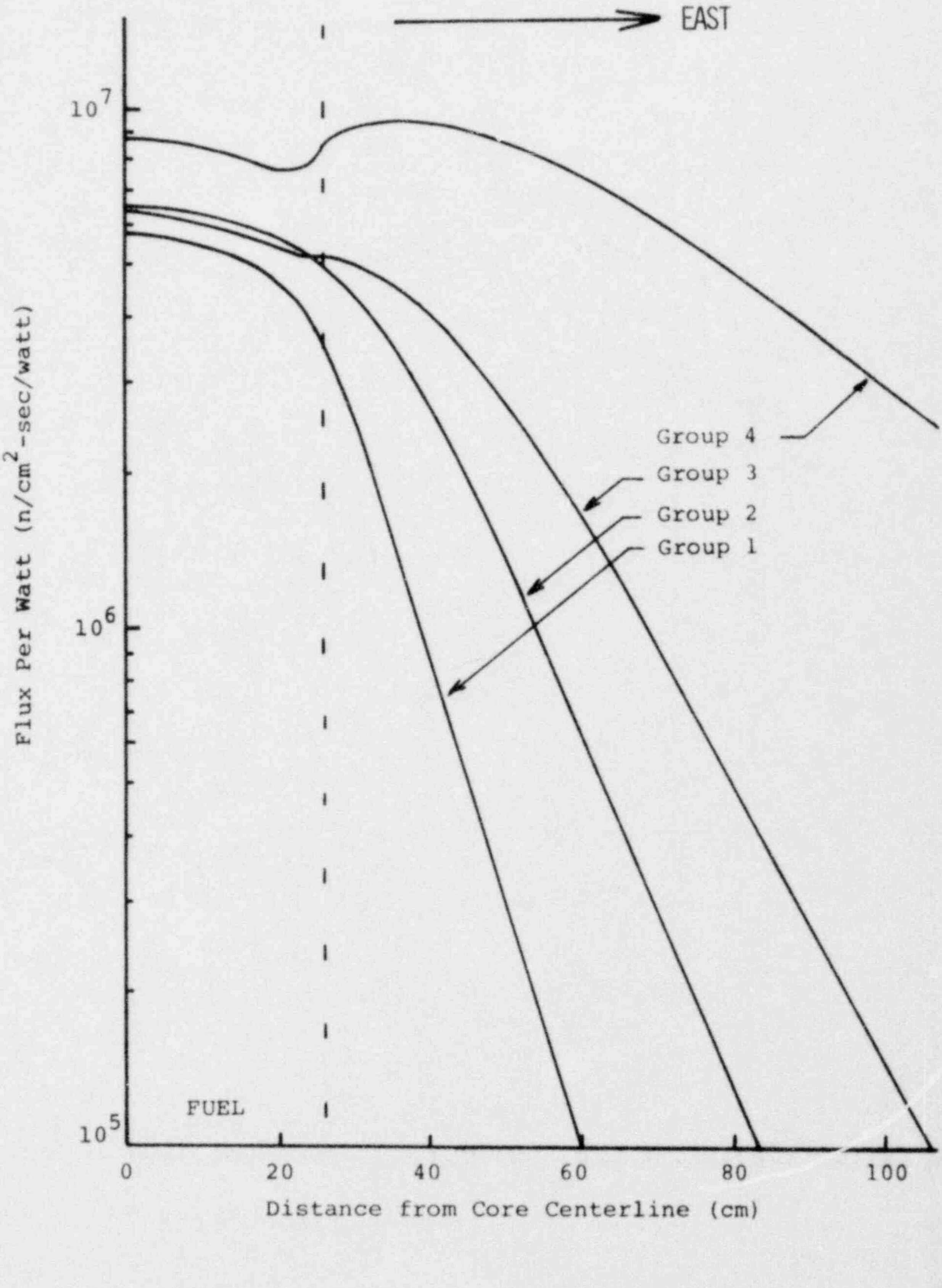


Figure 4-17. Group-Dependent Fluxes Along the East-West Direction--CORA Code Calculation (23)

TABLE 4-2

AVERAGE FLUX DATA FOR THE UFTR (23)

FUELED REGIONS

Direction	Group	Average Flux Per Watt	Peak-to-Average Flux Ratio
NORTH-SOUTH	1	5.45097 E-06	1.10878
	2	6.39734 E-06	1.06056
	3	6.33132 E-06	1.08064
	4	9.32447 E-06	1.13057
EAST-WEST	1	5.19175 E-06	1.11071
	2	5.99736 E-06	1.08844
	3	5.87437 E-06	1.07811
	4	8.33717 E-06	1.06152

TOTAL REACTOR

Direction	Group	Average Flux Per Watt	Peak-to-Average Flux Ratio
NORTH-SOUTH	1	1.92642 E-06	3.13738
	2	3.37201 E-06	2.01209
	3	4.15711 E-06	1.75352
	4	7.30648 E-06	1.65883
EAST-WEST	1	7.50292 E-05	7.68570
	2	1.07666 E-06	6.06296
	3	1.32403 E-06	4.78329
	4	3.65015 E-06	2.60856

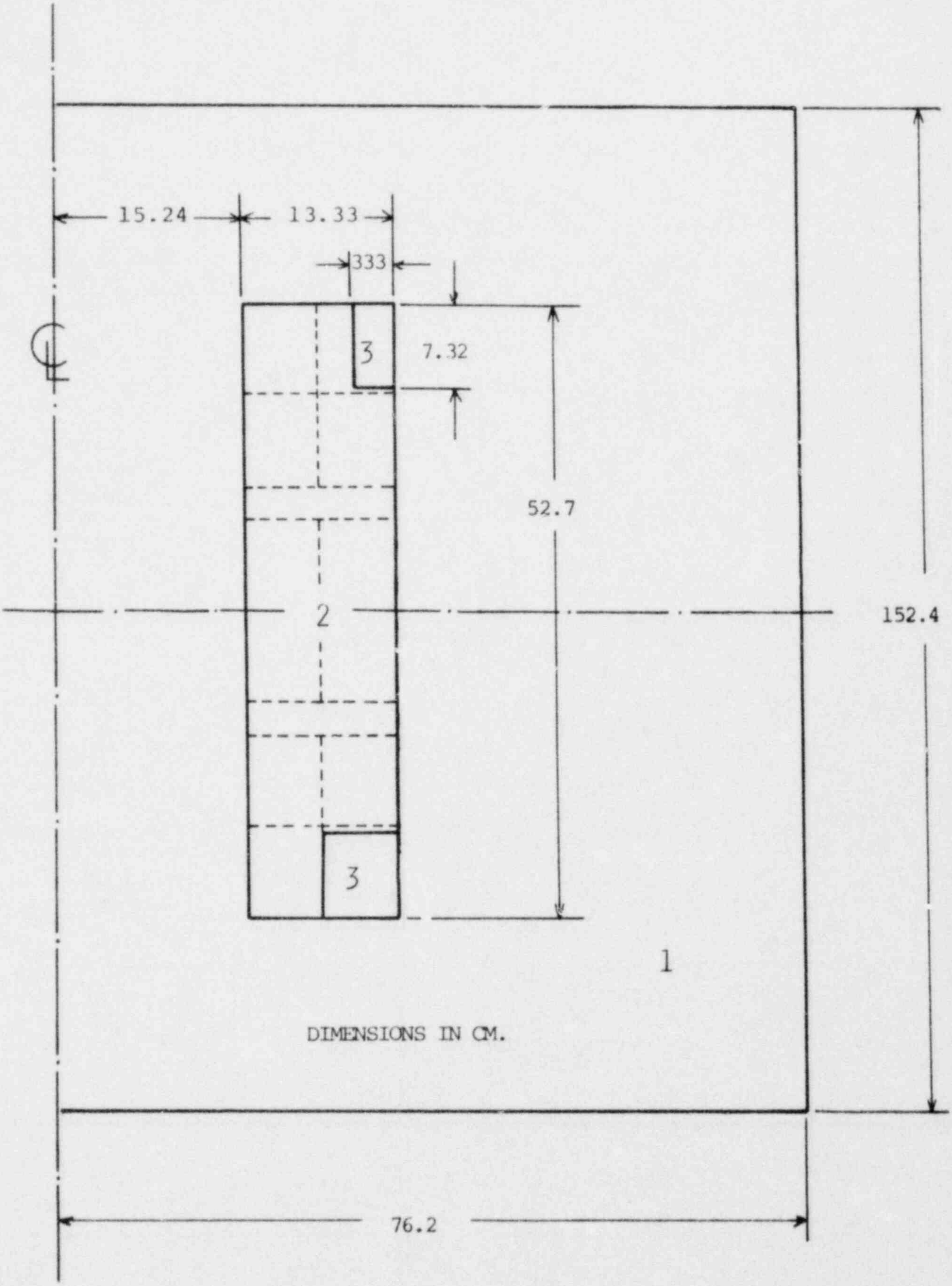


Figure 4-18. Reactor Model Used for Exterminator Code Calculation.

TABLE 4-3

SUMMARY OF UFTR HOMOGENIZED AVERAGE PARAMETERS (500 Kwth)*

$V^{Total} = 2.768\ m^3$	$N^{U-235} = 3.0923 \times 10^{18}$ atoms/cm^3
$\Sigma_a = 18.89 \times 10^{-4}\ cm^{-1}$	$\overline{\Phi} = 4.79 \times 10^{12}\ n\ cm^{-2} s^{-1}$ @ 500 KW
$\Sigma_a^{U-235} = 14.01 \times 10^{-4}\ cm^{-1}$	$\Sigma_f = 11.7426 \times 10^{-4}\ cm^{-1}$
$\sigma_a^{U-235} = 453b$	$\sigma_f^{U-235} = 379.7$ barns

NOTE: The only parameter greatly dependent upon the difference in power level considered (500 Kwth versus 100 Kwth) is the average flux. All other parameters are relatively independent of power level in this range and are considered applicable estimates for the UFTR at its current 100 Kwth rating.

TABLE 4-4

POISON PARAMETERS FOR Xe-135 AND Sm-149 ANALYSIS

Isotope	Fractional Fission Yield γ	Decay Constant $\lambda(sec^{-1})$	Absorption Cross Section (barns) σ_a
I-135	0.061	2.89×10^{-5}	negligible
Xe-135	0.003	2.09×10^{-5}	2.72×10^{6}
Pm-149	0.0113	3.56×10^{-6}	negligible
Sm-149	zero	stable	5.0×10^{6}

At 500 KWth, the Xenon equilibrium concentration, reached as usual in about 40 hours, is given by:

$$X_{eq} = \frac{\phi\Sigma_f\ (\gamma^I + \gamma^X)}{\lambda^X + \phi\sigma_a{}^X} = 1.062 \times 10^{13}\ cm^3 \qquad (4\text{-}1)$$

with a corresponding equilibrium absorption cross section given by:

$$\Sigma^{Xe}_{a_{eq}} = X_{eq}\ \sigma_a{}^X = 2.88 \times 10^{-5}\ cm^{-1}$$

This value is in good agreement with that read from Figure 4-19A for equilibrium Xe-135 in a highly enriched reactor. For significant (40-50%) "step" reductions, the buildup of Xe-135 after a step reduction in flux level from equilibrium conditions reaches a maximum concentration after ~3-4 hours with maximum Xe-135 concentrations about 7% larger than equilibrium.

The isotope samarium-149 is found to reach its equilibrium concentration in about eight or nine months due to the small fluxes found in the UFTR. This behavior is also considered approximately applicable whether the power level is 100 KWth or 500 KWth. The equilibrium absorption cross section in this case is:

$$\Sigma^{Sm}_{a_{eq}} \simeq \Sigma_f \gamma^{Pm} = 1.327 \times 10^{-5}\ cm^{-1}, \qquad (4\text{-}2)$$

in good agreement with results found on Figure 4-19B which shows approximately the behavior of Sm-149 at 500 KWth. The equilibrium absorption cross section will be approximately the same for 100 KWth operation; however, the original time to reach the equilibrium level is considered to have taken somewhat longer since the UFTR 100 KWth curve would fit just over Curve 6 shown in Figure 4-19B. A step reduction in flux levels from equilibrium causes the Sm-149 concentration to increase to a level given by:

$$N_{Sm} = N^{eq}_{Sm} + N^{eq}_{Pm} \qquad (4\text{-}3)$$

At 500 KWth the absorption cross section was calculated to be 1.423×10^{-5} cm^{-1}, a value approximately 7.2% larger than equilibrium. At the UFTR 100 KWth power level, the Sm-149 absorption cross section then increases considerably less than 7% above the equilibrium level since the Pm-149 level available for decay depends directly upon the equilibrium power level prior to a step reduction in power level or flux. A plot of the calculated samarium concentration increase with burnup is presented in Figure 4-20. It should be noted that the Sm-149 buildup is relatively flux independent when related to burnup at the low flux levels ($\lesssim 10^{13}$) present in the UFTR.

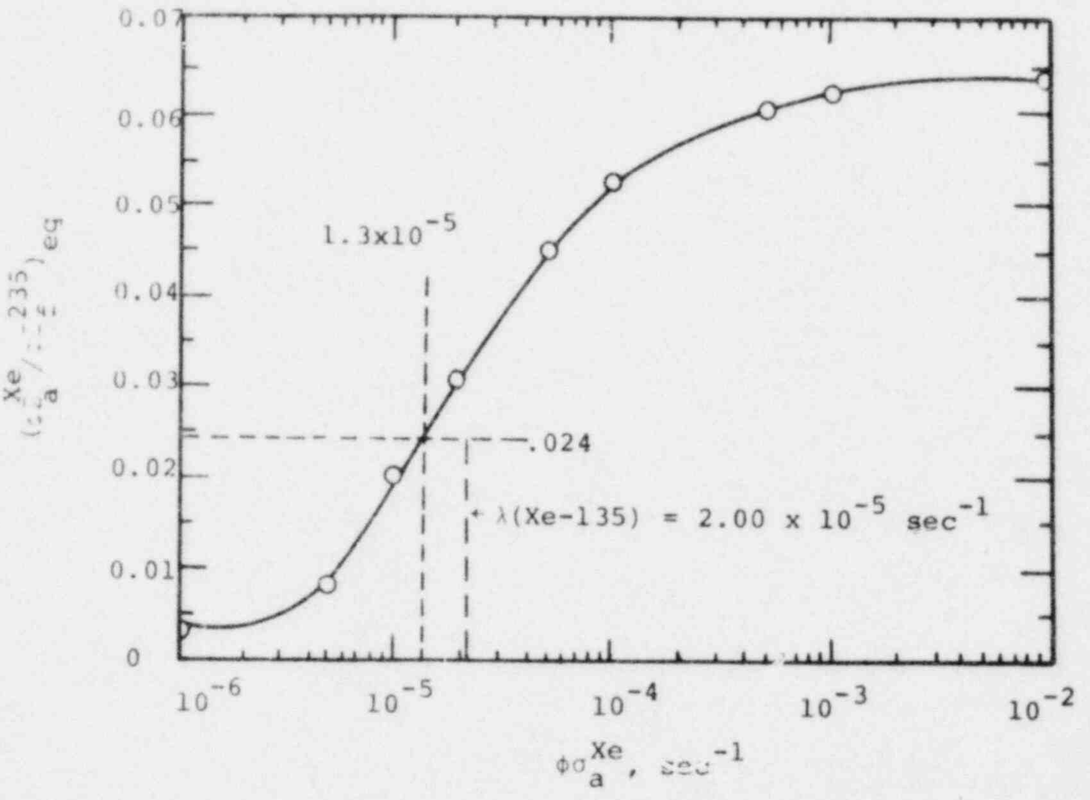


Figure 4-19A. Equilibrium Xenon-135 in a Highly Enriched Reactor.

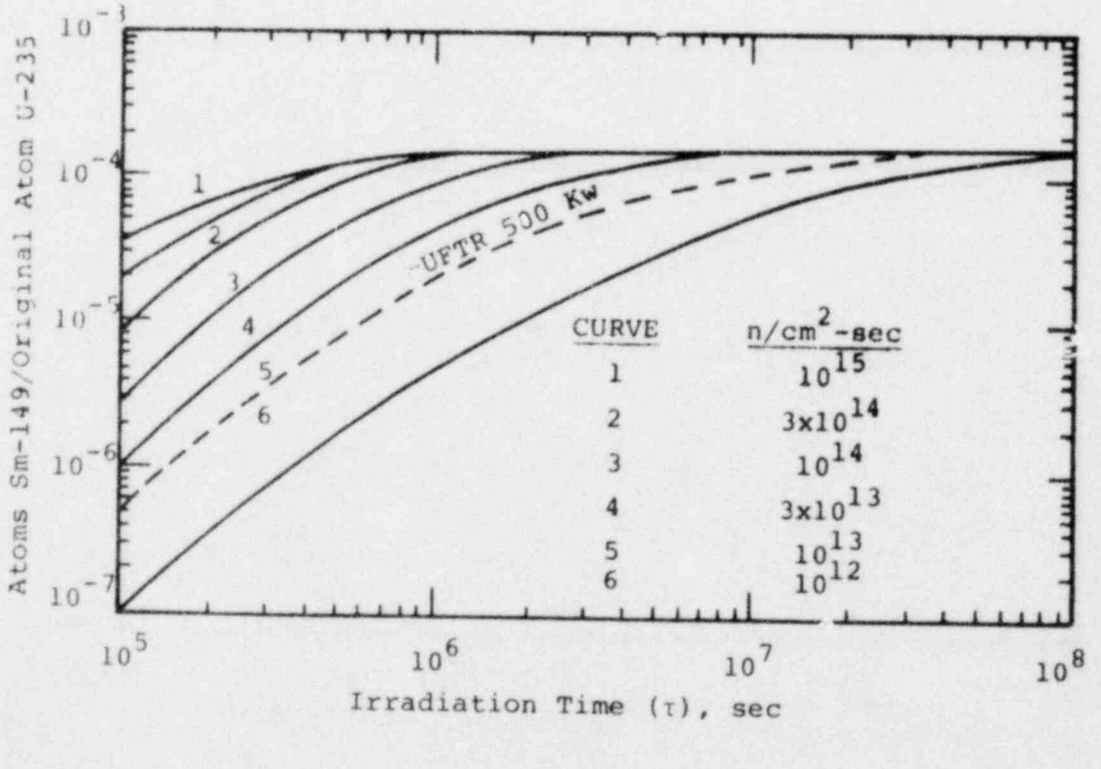


Figure 4-19B. Samarium-149 Buildup with Irradiation Time.

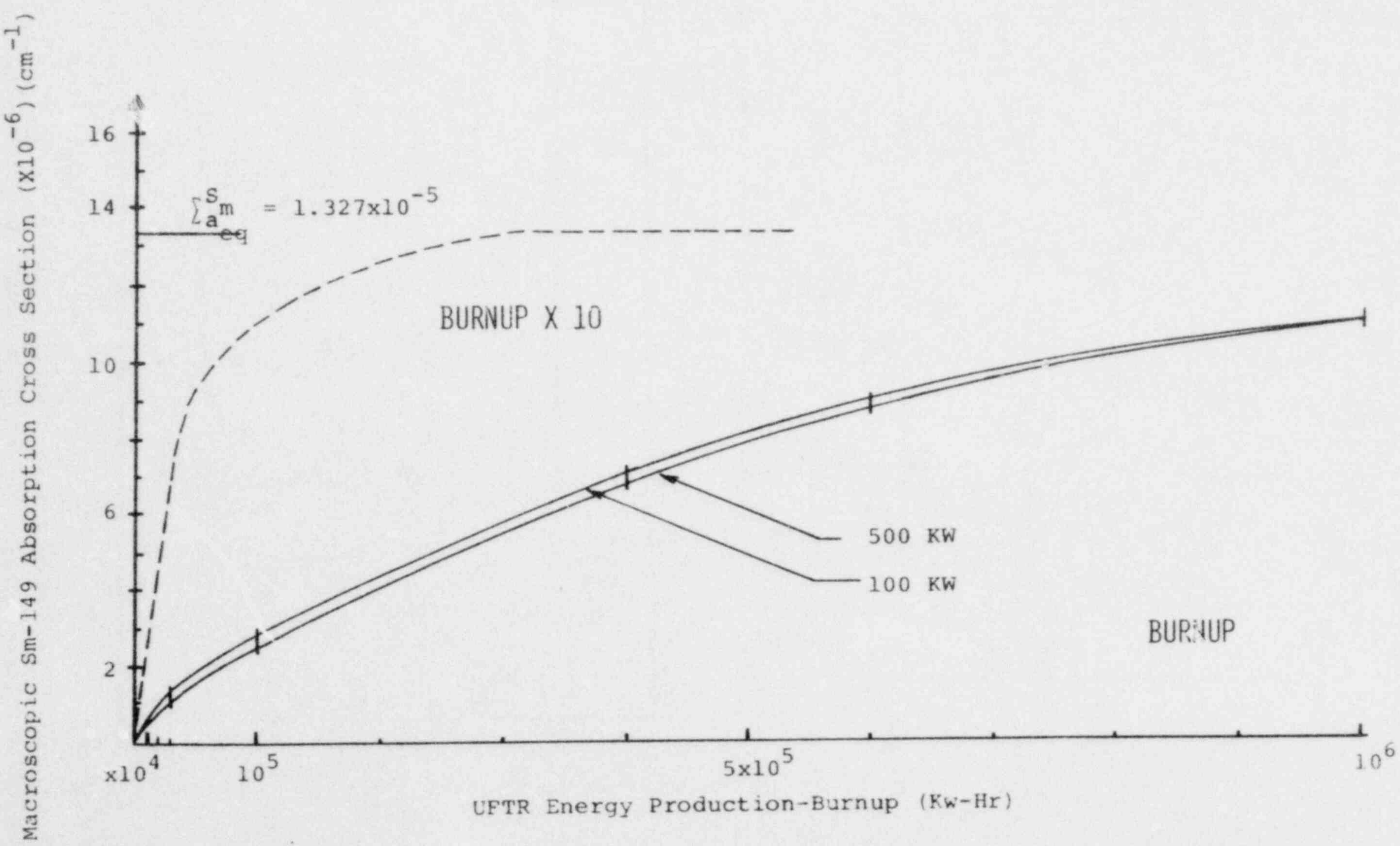


Figure 4-20. Samarium-149 Buildup for Operation at 100 and at 500 KWth.

For the gross fission product behavior in the presence of the low UFTR flux levels, the overall microscopic absorption cross section does not change appreciably with irradiation. Therefore, a constant rate of poison production is a recommended simplifying assumption. (27) A constant value of 51.2b is considered reasonable.

4.3.3 Reactivity Time Dependence

Since the fuel is depleted only very slightly in this reactor and since the fission product poisons are present in very small concentrations, Otaduy explains that it is reasonable to consider the effective multiplication factor to be proportional to the thermal utilization factor during the entire lifetime of the reactor. The beginning thermal utilization factor is equal to 0.7416. With time, the combined effects of burnup of the fuel and poisoning due to the fission products other than Xe-135 and Sm-149 start to become noticeable. The altered thermal utilization factor is given by the following equation:

$$f'' = \frac{\Sigma_a^{U-235} - \delta\Sigma_a^{U-235}}{\Sigma_a - \delta\Sigma_a^{U-235} + \Sigma_a^{Xe} + \Sigma_a^{Sm} + \Sigma_a^{fp}} \quad , \qquad (4\text{-}4)$$

where Σ_a^{U-235}, Σ_a and Σ_a^{Xe} are considered to remain roughly constant while the rest vary with burnup. The thermal utilization factor at end of life (EOL) will then be given by the following relationship:

$$f'' \approx 1-\Delta\rho \qquad (4\text{-}5)$$

where $\Delta\rho$ is the excess reactivity available for fuel burnup in the reactor.

Otaduy performed calculations concerning the change in reactivity with burnup for a number of UFTR thermal power levels. Table 4-5 includes these results for the 100 KWth case of interest here, as well as the 500 KWth case for comparison. Reactor behavior is clearly demonstrated in Figure 4-21. It should be noted that after the initial reactivity drop due to the rapid buildup of Xe-135, the change in reactivity present a linear tendency with an approximately constant slope of 10^{-8} $\Delta k/k$ per kwhr due to the combined effect of Sm-149 buildup and fuel burnup. This slope decreases to a value of 6.0×10^{-9} after the Sm-149 reaches equilibrium. The shape of the curves is also relatively independent of the power level of operation; therefore, accurate predictions of long-term reactivity changes are possible based upon knowledge of the total power produced over the time of operation--regardless of the power level at which the reactor is operated during that period.

Otaduy also found that since the maximum allowed excess reactivity for the UFTR is 2.3%, it is the initial linear response that governs the reactor life at 500 KWth operation. The same dependence should apply for the current 100 KWth rated system. The End of Life (EOL) is reached before Sm-149 reaches its equilibrium concentration. Therefore, EOL in the UFTR is primarily determined by Sm-149 buildup and not fuel depletion as it is in power reactors. Due to the 2.3% excess reactivity limit on the UFTR, a decrease

TABLE 4-5

SM-149 AND REACTIVITY VS. BURNUP FOR 100 AND 500 KW OPERATION (27)

POWER	100 Kw(1)				500 Kw(2)			
Burnup KwHr	Time hrs	Σ_a^{Sm} cm^{-1}	f"	Reactivity %Δk/k	Time hrs	Σ_a^{Sm} cm^{-1}	f"	Reactivity %Δk/k
2 E04	200	6.11 E-7	.7381	- .482	40	8.02 E-7	.73016	-1.554
4 E04	400	1.40 E-6	.7377	- .534	80	1.43 E-6	.72984	-1.596
1 E05	1000	2.26 E-6	.7371	- .610	200	2.83 E-6	.72908	-1.699
4 E05	4000	6.71 E-6	.7343	- .998	800	7.1 E-6	.72629	-2.076
6 E05	6000	8.63 E-6	.7327	-1.204	1200	8.9 E-6	.72482	-2.274
1 E05	10000	1.1 E-5	.7302	-1.538	2000	1.1 E-5	.72244	-2.595
5 E06	50000	1.327 E-5	.7129	-3.875	10000	1.321 E-5	.70497	-4.951

(1) $\overline{\phi}_{th}^{F}$ = 9.5872 E+11 cm^{-2}-sec^{-1}, Σ_a^{Xe} = 8.336 E-6 cm^{-1}

(2) $\overline{\phi}_{th}^{F}$ = 4.7936 E+12 $cm^{-2}sec^{-1}$, Σ_a^{Xe} = 2.887 E-5 cm^{-1}

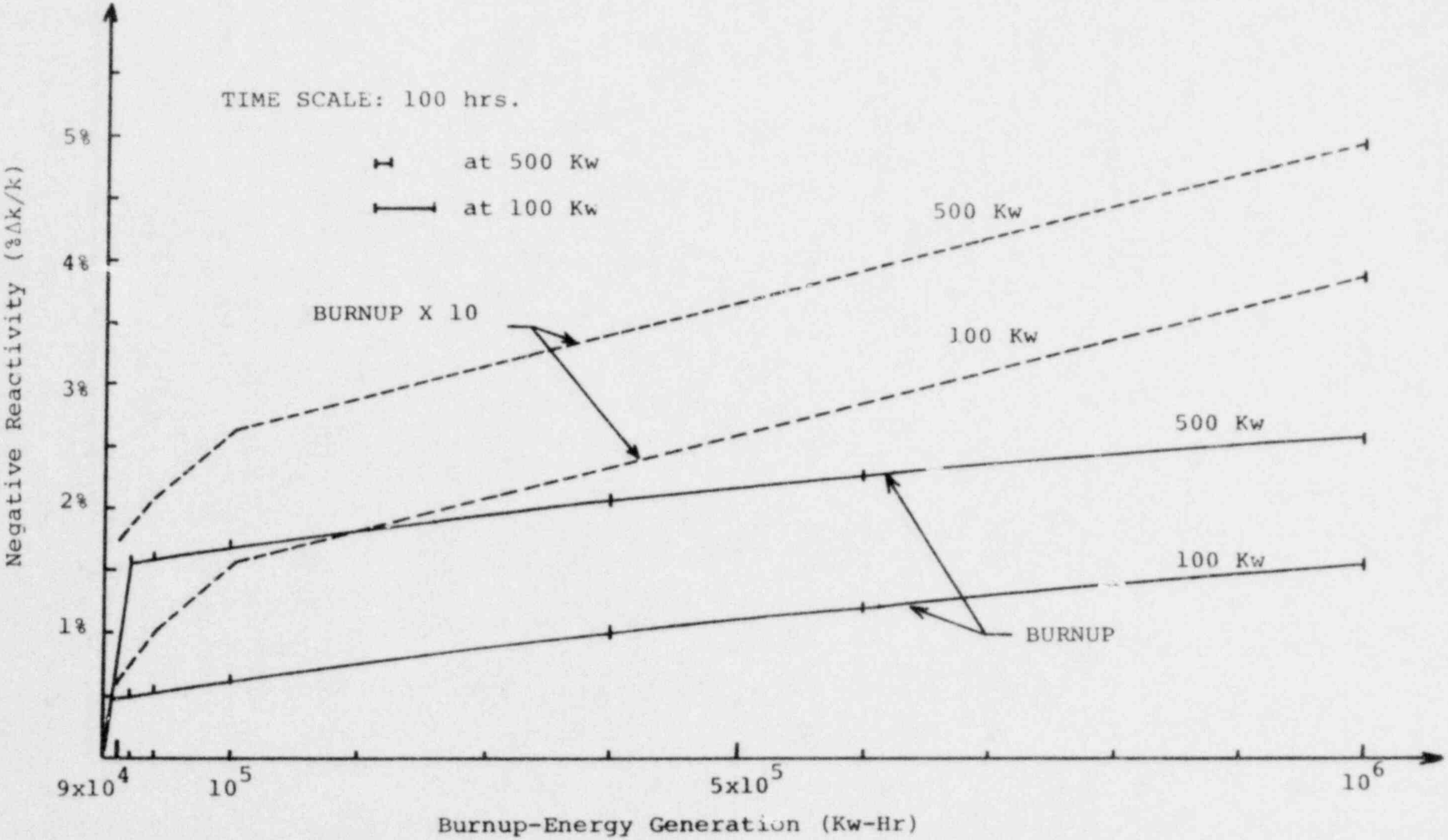


Figure 4-21. Reactivity Drop with Burn-up for Operation at 100 and 500 KW.

of 0.9% in reactivity in the linear part of the curves from Figure 4-21 will determine the EOL of the core. Since the slope of this curve is approximately 10^{-6}%/KWth, the EOL is defined by a burnup of 3.75 MWD or approximately 1800 hours at 500 KWth or 9000 hours at 100 KWth. The change in thermal utilization factor or the reactivity change at EOL is calculated to be -2.515%Δk/k.

Two other methods of analysis were used to check the validity of this simplistic model. One model uses first order perturbation theory with a homogenized fuel region without giving consideration to the space dependence of burnup, and the last method treats the space dependence of the change of parameters due to the non-uniformity of the power distribution using perturbation theory also. Table 4-6 includes the results of the reactivity change with burnup at 500 Kwth as calculated by the first order perturbation theory analysis and the thermal utilization analysis for comparison purposes. In analyzing these results obtained by these different methods calculated at EOL for the UFTR, there is little difference in the results. Therefore, since all the results are comparable for this low burnup, highly-enriched research reactor, it is considered to be unnecessary and not useful to perform a detailed space-dependent calculation to analyze the reactivity time dependence.

Fuel management studies performed by Otaduy have led to several conclusions. (27) First, rearranging the fuel in the core at the time of the selected EOL for this study (9.0×10^{5} kwhr), will not produce a significant gain in reactivity. The gain associated with the rotation of the fuel elements is found to be of the order of 3.66×10^{-5} Δk/k equivalent to a power production of 6840 kwhr or only 14 hours of operation at 500 KWth or 70 hours at 100 KWth. Second, shuffling the fuel produces a reactivity gain equivalent to 21 hours of operation at 500 KWth. The combination of shuffling and fuel rotation yields a predicted gain equivalent to 30 hours of operation at 500 KWth or 150 hours at 100 KWth. Therefore, shuffling and rotation operations are of little interest. The introduction of fresh fuel in place of the four (4) most highly burned bundles yields a predicted reactivity gain of 2.858×10^{-3} Δk/k, equivalent to 44,000 kwhr, which represents 880 hours of operation at 500 KWth or 4400 hours at 100 KWth. This gain is approximately equivalent to 50% of the selected EOL of the reactor core and does represent a significant gain.

4.4 Thermal and Hydraulic Design

Average inlet and outlet coolant temperatures and the coolant flow rate for the UFTR at 100 KWth operation are included in Table 4-1.

Studies have been carried out to evaluate the heat transfer properties and the corresponding fuel plate temperature distribution of the associated water channel for the UFTR core.

4.4.1 Fuel Plate Heat Transfer Computational Model

The temperature distribution of the "hottest" fuel plate and associated water channel was calculated by use of a steady-state heat conduction, digital computer program (8). This program is designed to describe fuel plate heat transfer and temperatures by setting up heat balance

TABLE 4-6

REACTIVITY CHANGE WITH BURNUP (4)

OPERATION TIME, HRS	Sm(old) $\Sigma_a cm^{-1}$	BURNUP KWHR	REACTIVITY CHANGE, %Δk/k Perturbation Theory	Thermal Utilization Analysis
40	8.02 E-7	2.0 E+4	-1.570%	-1.557%
80	1.43 E-6	4.0 E+4	-1.614%	-1.593%
200	2.83 E-6	1.0 E+5	-1.719%	-1.696%
800	7.10 E-6	4.0 E+5	-2.099%	-2.073%
1200	8.90 E-6	6.0 E+5	-2.298%	-2.270%
2000	1.10 E-5	1.0 E+6	-2.616%	-2.590%
10000	1.33 E-5	5.0 E+6	-4.809%	-4.947%

NOTE: UFTR Operation at 500 Kwth.

relationships at nodal points distributed throughout the modeled fuel plate and water channel as shown in Figure 4-22. The model for which the temperature distribution is determined includes the following ten assumptions which are used in setting up heat balances at the nodal points:

1. The plate is 25 inches long.

2. Constant properties are used based on average temperatures for all materials involved.

3. The heat generation rate and the temperature distribution across the 2.845 inch side of the plate are constant, thus reducing the analysis to two dimensions.

4. The heat generation rate in the meat of the fuel (0.04 inches) is constant--does not vary with thickness position in the plate.

5. The heat generation rate along the length of the plate follows a sinusoidal distribution.

6. The maximum heat produced per unit volume is assumed to be a multiple of the average heat production density. This factor corresponds to the hot-channel factor and is chosen as 1.5 to best represent the "hottest" fuel plate in a conservative manner.

7. The heat generation rate and the temperature distribution is symmetrical about the midplane of the fuel plate.

8. The top and bottom boundaries or fuel plate endpoints are assumed to be insulated.

9. The coolant temperature distribution is calculated on the basis of defined values (i.e., power level, heat flux, coolant flow rate, and coolant entrance temperature).

10. The coolant flow is laminar and, for the purpose of finding the appropriate Nusselt number, a constant fuel plate surface temperature is assumed.

As indicated, these assumptions are used to set up heat balances at nodal points distributed throughout the plate and water channel. The length between nodal points is 1.0 inch and the width between horizontally spaced nodal points is 0.005 inches as indicated in the nodal point grid distribution presented in Figure 4-23.

The grid is, therefore, 9 by 26 (234 nodal points). The nodal equations are then solved iteratively by the Gauss-Seidel method.

At the 100 KWth power level, a primary coolant flow rate is assumed to be 31.2 gpm; a conservative hot channel factor of 1.5 is also assumed. Additional calculations were performed at an assumed power level of 500 KWth. At a power level of 500 KWth the primary coolant flow will be approximately 65 gpm. The corresponding Reynolds Number is 721. Thus the flow is definitely laminar and the above model directly applies. Similarly at 100 KWth the primary coolant flow is nominally only 31.2 gpm so that, at

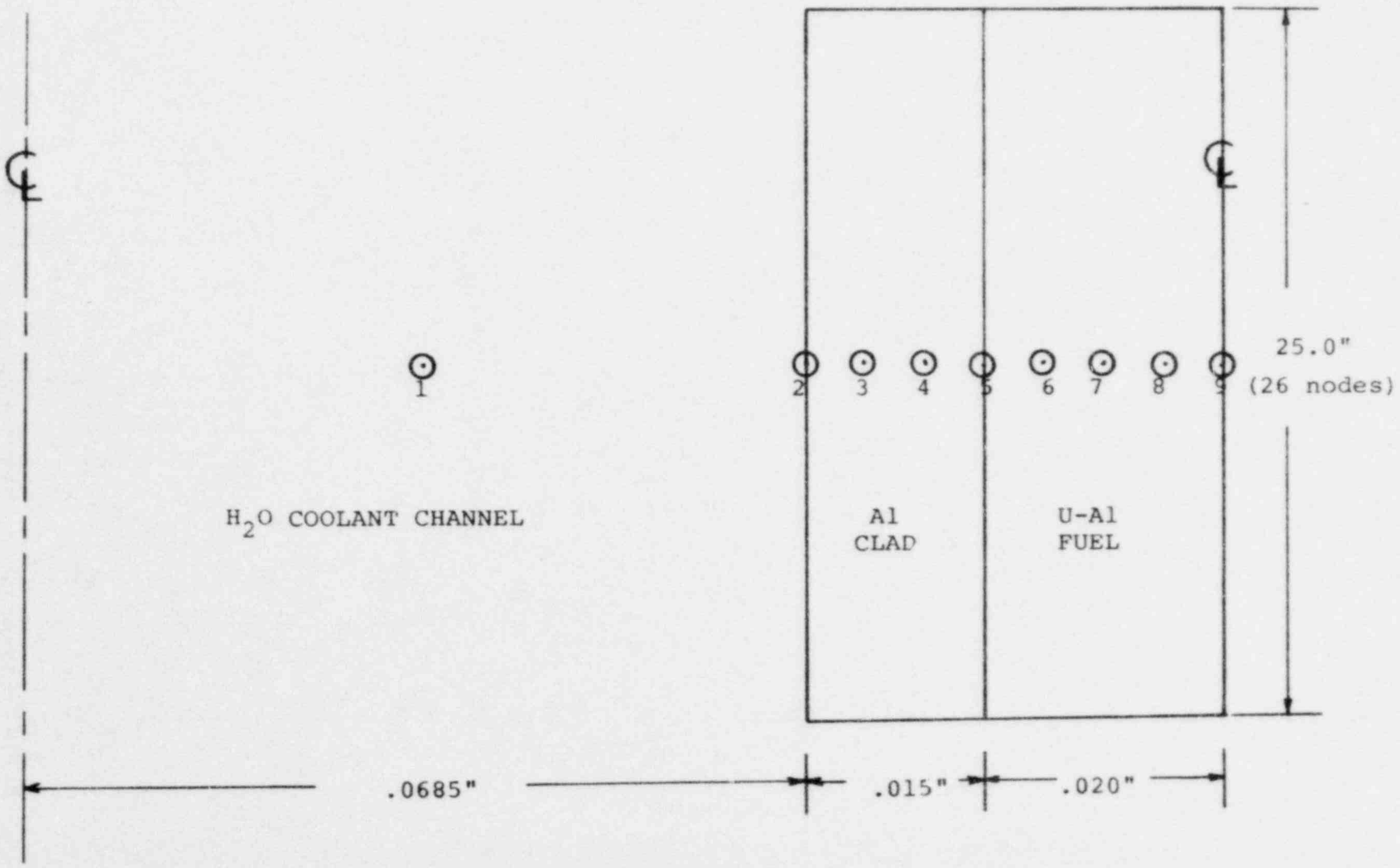


Figure 4-22. Grid for Nodal Point Distribution Used For UFTR Heat Transfer Calculation. (23)

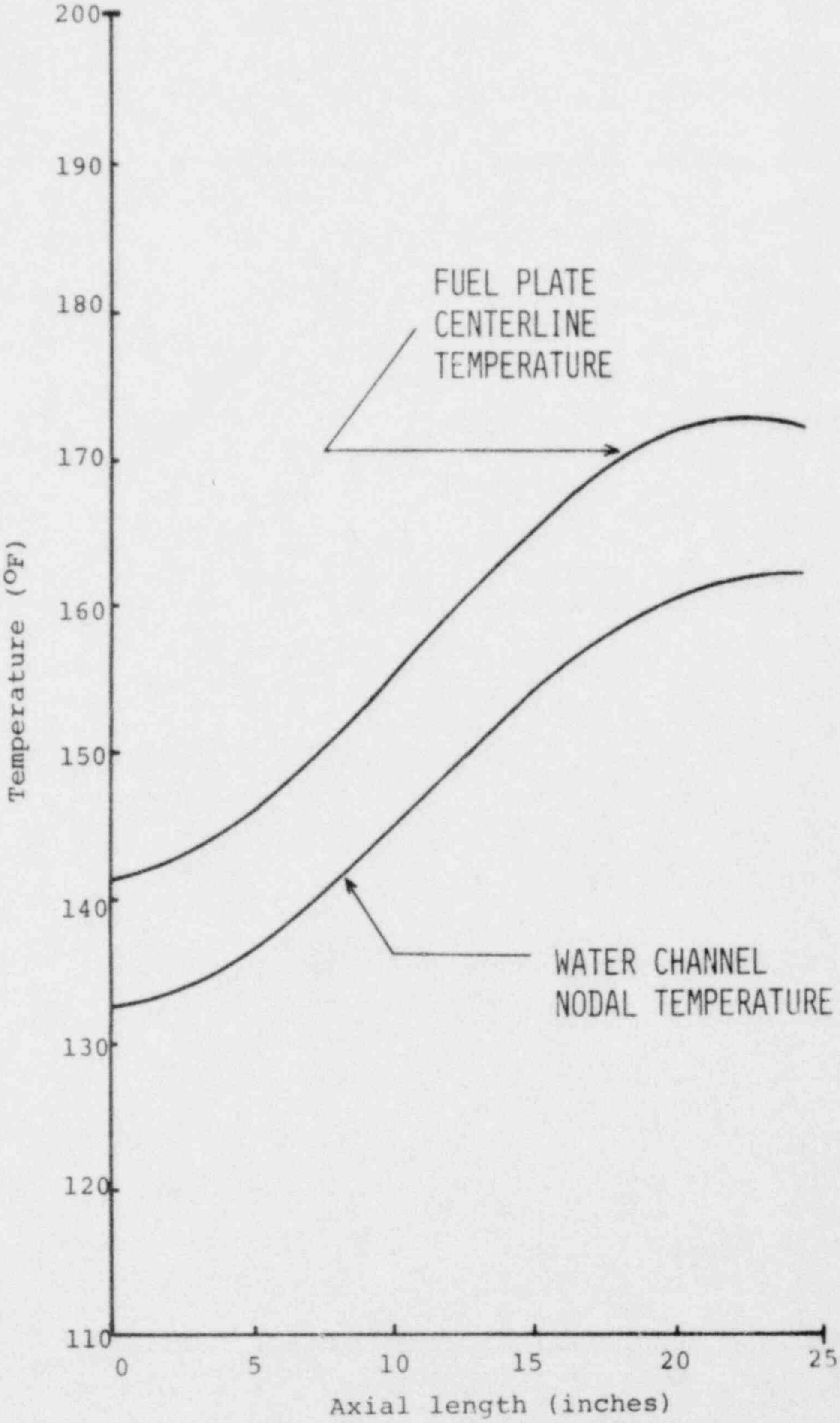


Figure 4-23. Temperature Distribution of the "Hottest" Fuel Plate and Water Channel at 100 KWth (Coolant Flow Rate = 31.2 gpm).

current rated power conditions the UFTR flow through the core has a Reynolds Number below 400 and is definitely laminar in nature. To better approximate the "hottest" fuel plate, hot-channel factors were also calculated by Wagner. (23) As presented in Table 4-2, the peak-to-average thermal flux ratio in the North-South direction for the fueled regions is 1.13 and in the East-West direction 1.06. The conservative assumption that both effects apply simultaneously is made so that an overall hot-channel factor of 1.20 is obtained. On this basis, a conservative value of 1.5 was selected for the calculations. For comparison, hot-channel factors of 1 and 3 were also used.

4.4.2 Results of Heat Transfer Calculations

A brief summary of the computed heat transfer and temperature results is presented in Table 4-7 for various coolant flow rates, power levels, hot-channel factors, and coolant inlet temperatures for both comparison with the 100 KWth results which are also included. The results in Table 4-7 indicate that, even assuming a conservative hot-channel factor of 1.5, 197.8°F is the maximum fuel plate temperature for operation at a power level of 500 KWth (which is five times the currently rated power for which the report is supporting relicensing and a primary coolant rate of 65 gpm). Similarly, at 100 KWth with an assumed hot-channel factor of 1.5, the maximum fuel plate temperature is calculated to be 173.8°F. Both maximum temperatures are well within the operating temperatures of the fuel plate. To check the validity of the model, actual operational data at 100 KWth is compared to computed results using a hot-channel factor of unity. The computer primary coolant temperature change (ΔT) is found to underestimate the actual operational temperature rise by 8 percent. Assuming this correlation to hold true for 500 KWth power operation, the primary coolant ΔT as predicted by this model will be approximately 53°F. In the same manner, the computed results using a hot-channel factor of 1.5 are used to predict the coolant outlet temperature of the hottest fuel box. Actual operational data shows that the model overestimates, as expected, the hottest fuel box coolant temperature change (ΔT) by 13 percent. Therefore, the highest fuel box coolant outlet temperature to be expected for the hypothetical 500 KWth operation and for a coolant inlet temperature of 111.7°F is considered to be approximately 176°F.

From these results, it is concluded that a hot-channel factor of 1.5 can be expected to yield a relatively good representation of the temperature distribution of the "hottest" fuel plate and water channel in the UFTR. Centerline fuel plate (nodal point 9 in Figure 4-22) and bulk water channel (nodal point 1) axial temperature distributions are shown in Figures 4-23 and 4-24 respectively. Figures 4-23 and 4-24 include the results of calculations made assuming reactor operation at the current rated UFTR power of 100 KWth and at a hypothetical upgraded power level of 500 KWth as presented by Wagner. (23) The temperature distribution of the fuel plate cladding surface (nodal point 2) is considered to be of the same shape as the centerline distribution. (23) The ΔT across the fuel plate varies from about 0.13°F to 0.21°F for operation at the current UFTR rated power level of 100 KWth; at 500 KWth, the fuel plate ΔT variation is calculated to be about 0.13°F to 0.37°F. In either case, the metallic nature of the fuel and the bonded cladding in conjunction with low power densities prevent excessive fuel temperatures from being reached in the UFTR when coolant is present.

TABLE 4-7

FUEL PLATE HEAT TRANSFER DATA AND RESULTS OF CALCULATIONS

Coolant Flow Rate (gpm)	Power Level (Kwth)	Coolant Inlet* Temperature (°F)	Coolant Outlet Temperature (°F)	Hot Channel Factor	Maximum Predicted Fuel Temperature (°F)
31.2	100	132.7	153.1	1.0	162.9
65.0	500	111.7	160.5	1.0	171.5
31.2	100	132.7	163.3	1.5	173.8
65.0	500	111.7	185.0	1.5	197.8
31.2	100	132.7	193.8	3.0	206.4
65.0	500	111.7	212.0	3.0	238.3

*These values were computed by primary coolant heat balances assuming equilibrium coolant outlet temperatures of 155°F for 100 KW operation and of 165°F for the hypothetical KW operation.

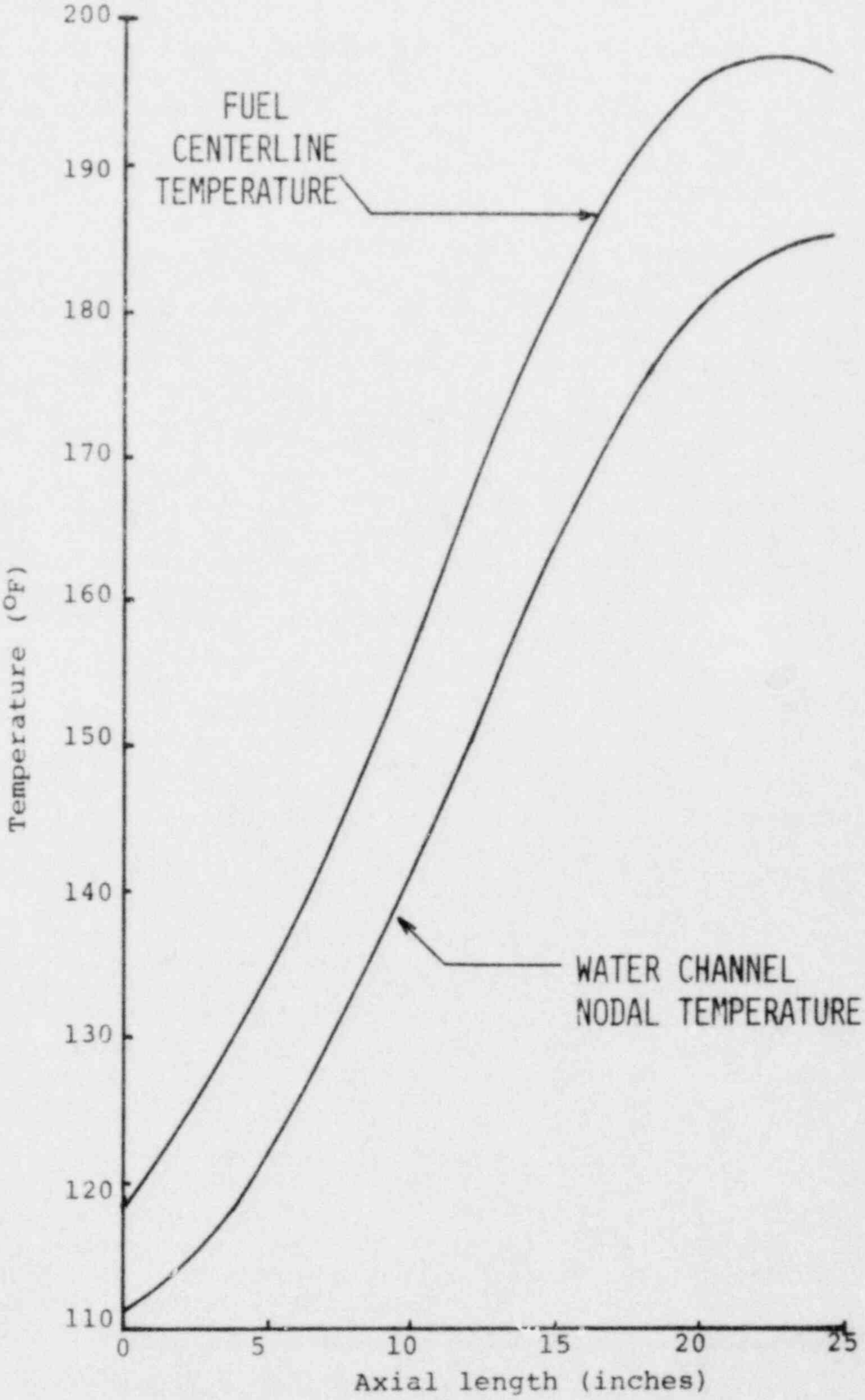


Figure 4-24. Temperature Distribution of the "Hottest" Fuel Plate and Water Channel at 500 KWth (Coolant Flow Rate = 65 gpm).

Investigations of the fuel temperature behavior after a loss of coolant accident and shutdown of the reactor was also investigated by Wagner. (23) Using a conservative heat transfer model, it was concluded that the fuel plate temperatures will increase only about 30°F under these circumstances. Further explanation and discussion on this topic are contained in Chapter 15, Accident Analysis.

As a result of the above studies and the operational experience of the UFTR since May 1959, first rated at 10 KWth and then at 100 KWth power levels, it is concluded that the thermal and hydraulic design of the UFTR facility is safe and considered more than adequate for continued operation at the 100 KWth power level. The large safety margin in effect even for operation at 500 KWth further substantiates the safety of the UFTR from a thermal hydraulic point of view and supports continued licensing of the UFTR for operation at the 100 KWth rated power level.

Descriptions and drawings of the UFTR's present primary and secondary cooling systems are found in Chapter 5, "Reactor Coolant System and Connected System" The instrumentation necessary for measuring the temperatures and flows of the reactor fuel and coolant is also shown in the schematic diagrams of Figure 5-1 showing Primary Cooling System and Figure 5-5 showing the Secondary Cooling System.

4.5 Reactor Materials

The usual detailed information on control rod system structural materials is not included here since the control rod systems are previously operated systems. They have been designed and installed to meet licensing requirements previously. The basic construction and materials that make up the control blades has been presented and discussed in previous sections such as Section 4.1.

Information on reactor internals is not considered necessary for the UFTR since there is no structural integrity problem with the core itself. Again, the basic arrangement and design of the core internals is described in Section 4.1. Further detailed information on reactor system materials is available from drawings and records maintained at the UFTR Facility in accordance with NRC requirements and UFTR Standard Operating Procedures.

4.6 Functional Design of Reactivity Control System

Reactivity control of the UFTR is provided by four control blades, (3 safety and 1 regulating), as previously illustrated with their drive mechanisms in Figures 4-4 and 4-5. Reactor shutdown can also be accomplished by voiding the moderator/coolant from the core. Two independent means of voiding the moderator/coolant from the core are provided:

(a) water dump via the Primary Coolant System Dump Valve opening under Full Trip conditions,

(b) water dump via the rupture disk breaking under pressure conditions above design value.

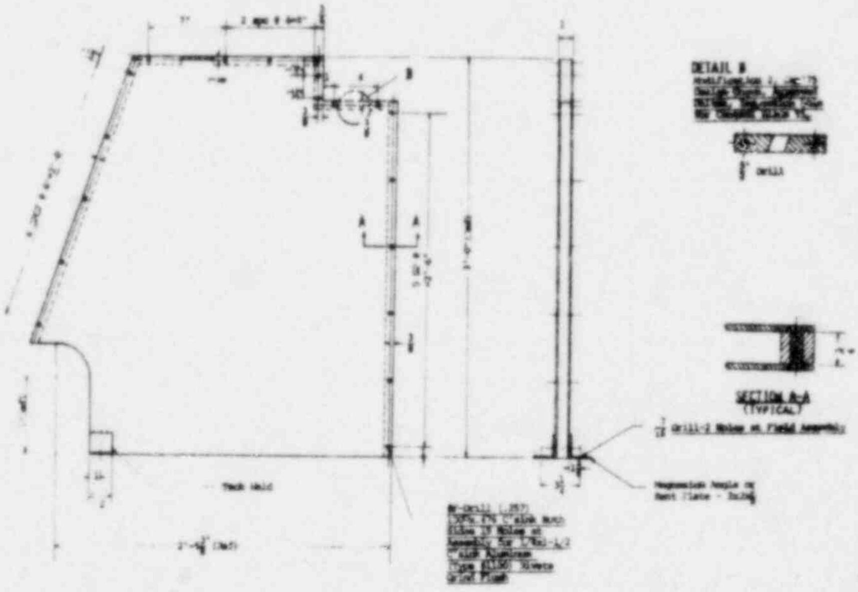


Figure 4-25A. UFTR Control Blade Shroud Assembly.

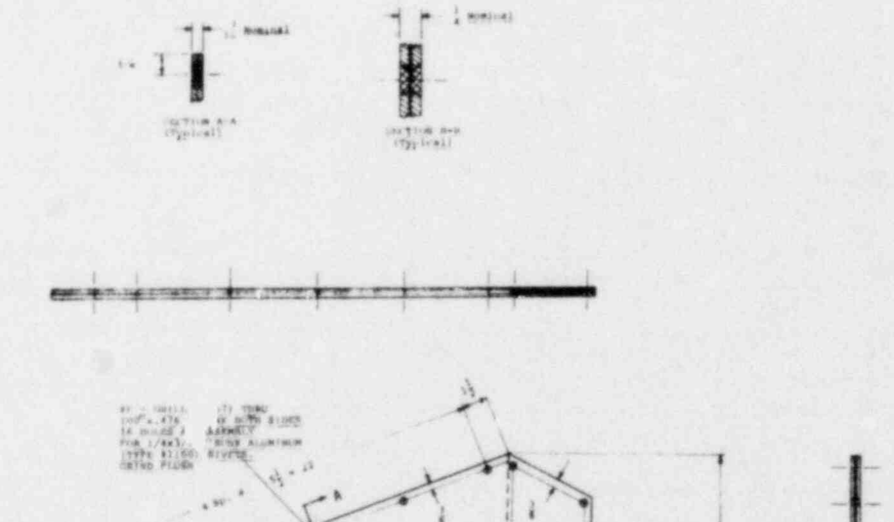


Figure 4-25B. UFTR Control Blade Assembly.

POOR ORIGINAL

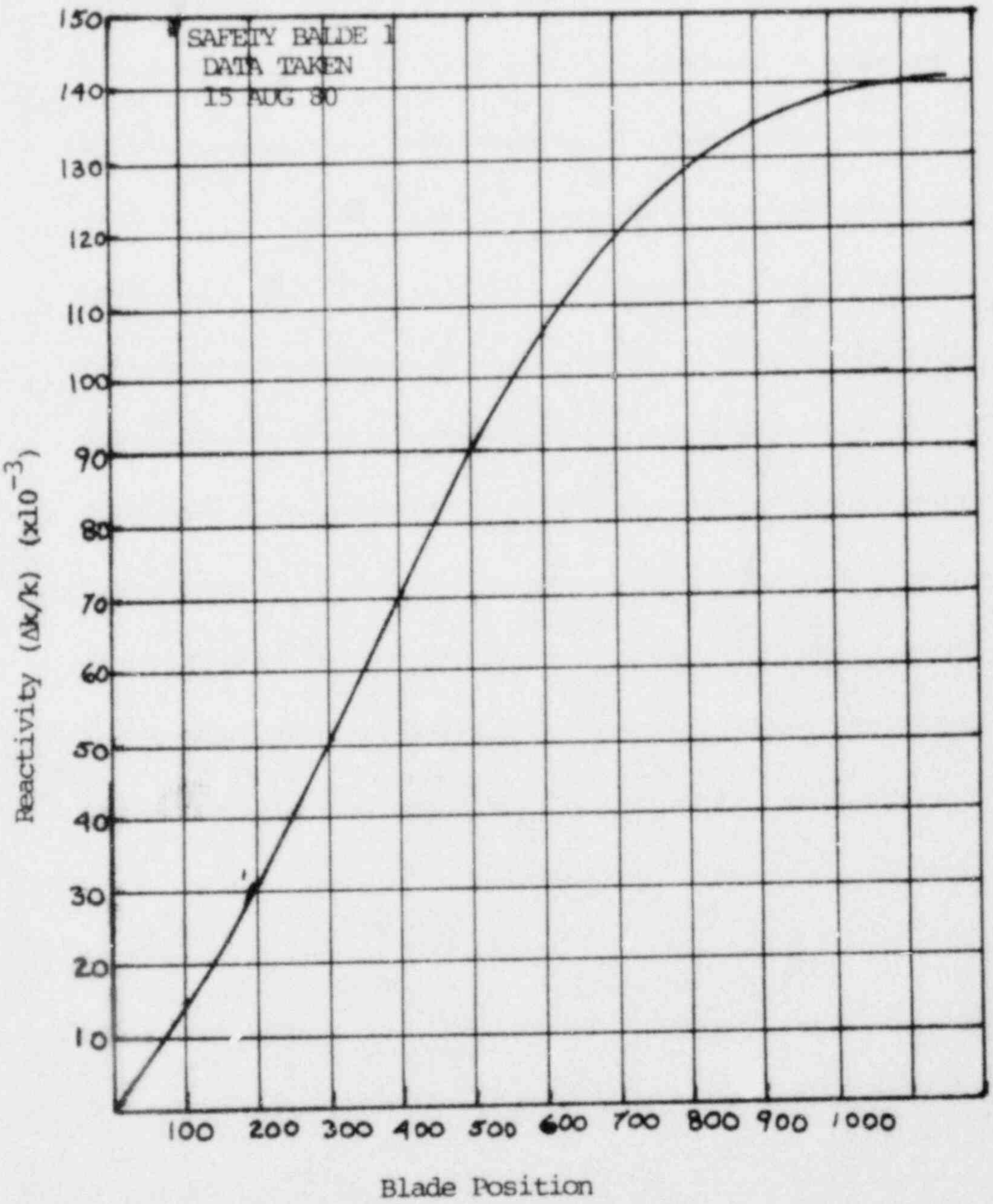


Figure 4-26. Reactivity Integral Rod Worth Curve for UFTR Safety Blade #1.

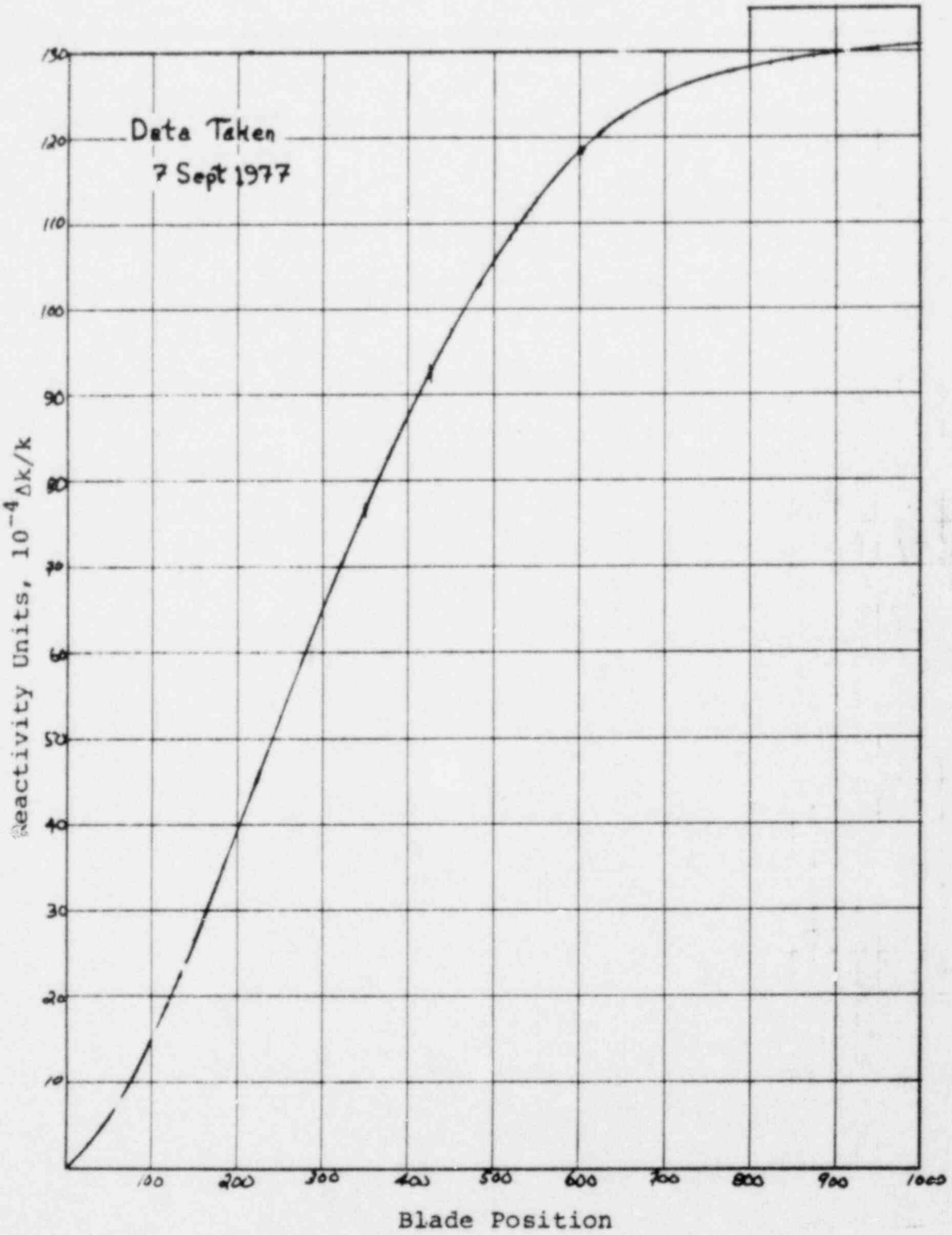


Figure 4-27. Integral Rod Worth Curve for UFTR Safety Blade #2.

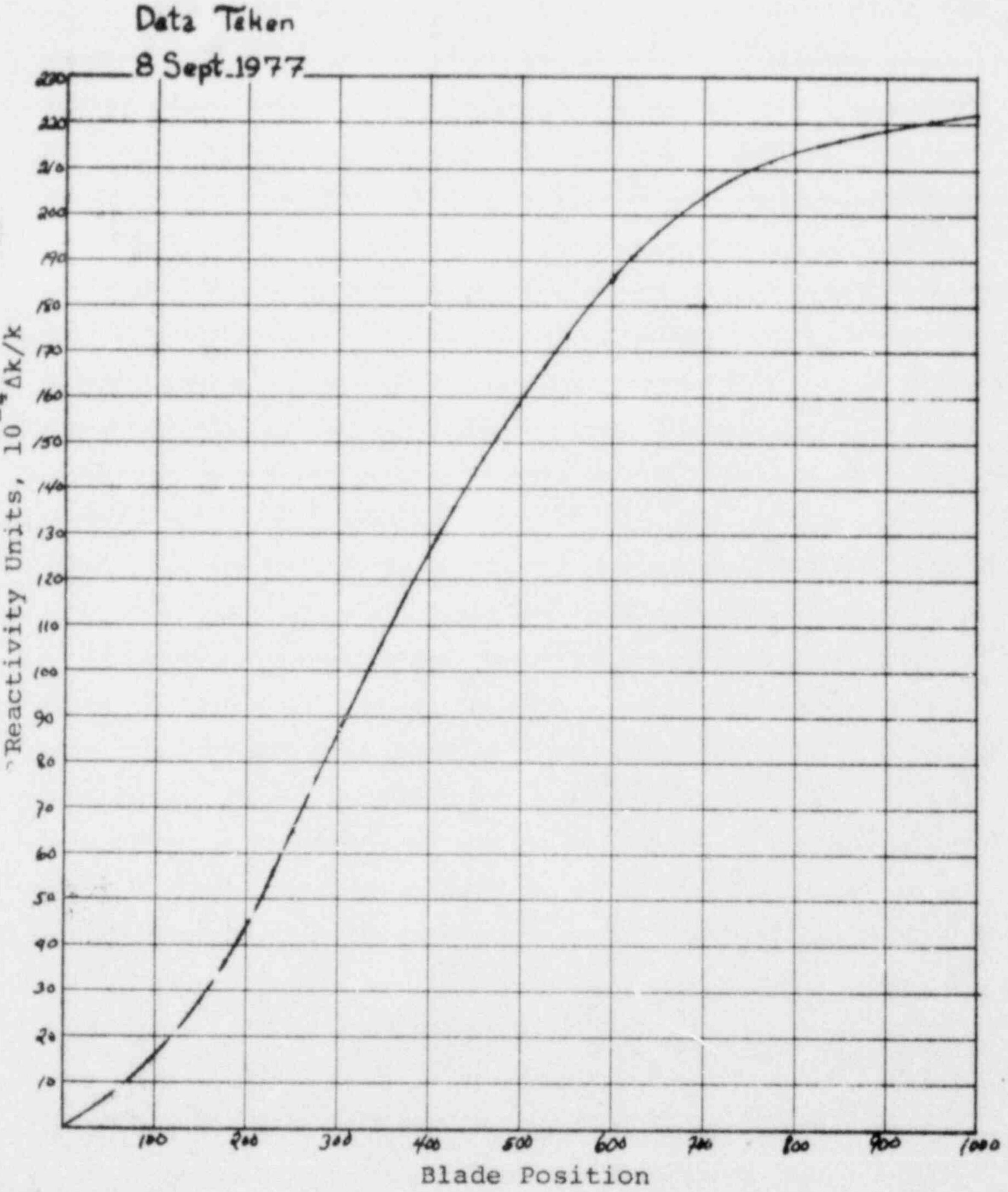


Figure 4-28. Integral Rod Worth Curve for UFTR Safety Blade #3.

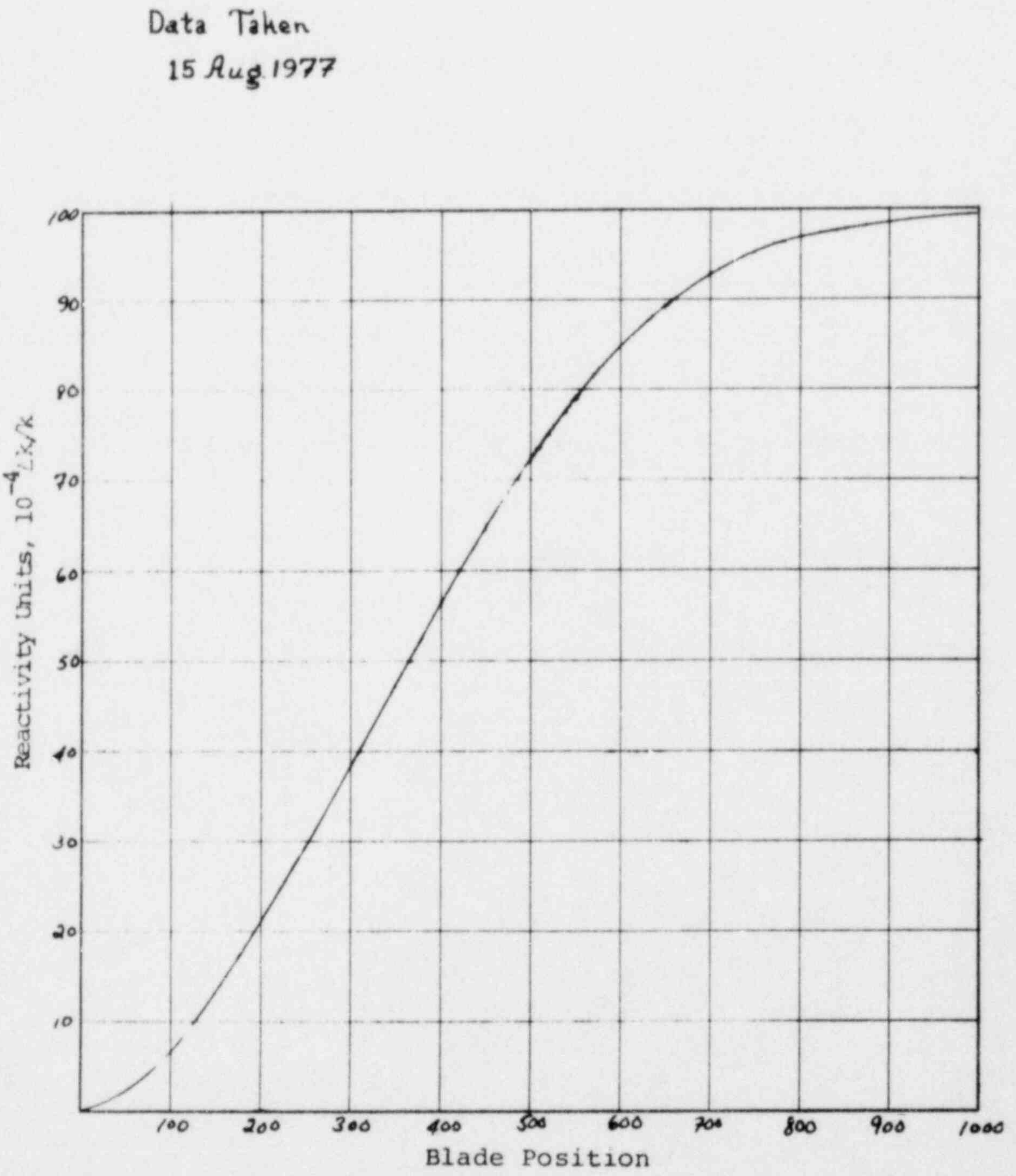


Figure 4-29. Integral Rod Worth Curve for UFTR Regulating Blade

The control blades are of the swing-arm type consisting of four cadmium vanes protected by magnesium shrouds; they operate by moving in a vertical arc within the spaces between the fuel boxes as illustrated in Figure 4-5. The control blade, regulating blade and shroud configuration are shown in the isometric diagram of Figure 4-5 while actual dimensions are presented on the drawings in Figure 4-25. Current rod calibration (integral rod worth versus position) curves for the UFTR system are presented in Figures 4-26 through 4-29.

Blade motion is limited to a removal time of at least 100 sec and the insertion time trip conditions is measured to be less than 0.4 sec. The reactor blade withdrawal interlock system prevents blade motion which will exceed the reactivity addition rate of 0.06%Δk/k per second, as specified in the UFTR Technical Specifications. The control blade drive system consists of a two phase fractional horsepower motor that operates through a reduction gear train, and an electrically energized magnetic clutch that transmits a motor torque through the control blade shaft, allowing motion of the control blades. The blades are sustained in a raised position by means of this motor, acting through the electromagnetic clutch. Interruption of the magnet current results in a decoupling of the motor drive from the blade drive shaft, causing the blades to fall back into the core. Position indicators, mechanically geared to the rod drives transmit rod position information to the console. Circuitry associated with control blade movement is presented and discussed in Chapter 7. Additional data for the UFTR core fluxes is presented in Table 4-2.

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter describes the UFTR cooling systems and its various components. The UFTR is cooled by a primary and secondary coolant system. Due to the simplicity of design and low power operation of the UFTR argonaut type reactor, this chapter is greatly simplified from what is required for a typical power reactor.

In general, the primary coolant system transfers the heat from the reactor to the heat exchanger. This heat is removed by the secondary coolant systems to the storm sewer with no mixing of water between the two systems.

5.1 Primary Coolant System

The primary coolant loop and purification system of the UFTR are shown schematically in Figure 5-1. The UFTR has a reactor core capacity of 33 gallons and a primary coolant flow rate of approximately 40 gpm, with a capability of 65 gpm flow. (5) The primary coolant is demineralized water with a low permissible value of resistivity of 400,000 ohm-cm. The primary coolant is stored in the coolant storage tank which has a capacity of 200 gallons of water, approximately six (6) times the capacity of the reactor. Water is made up to the primary system by demineralizing city water and using a temporary connection to the primary coolant tank (see Section 5.1.3). The primary pump (rated at 65 gpm), which draws its suction from the primary storage tank, circulates the water through the heat exchanger before delivering it to the fuel boxes. The water flows up and around the fuel bundles, rises to the top of the fuel boxes and it is discharged, gravity driven, through the side orifices. Flow from the coolant storage tank is controlled by a ball valve in the pump discharge line which presently limits the flow rate to 40 gpm. A flow measuring instrument which is located on the exit line from the heat exchanger, transmits a flow indication and a scram signal to the control console. This scram signal is part of the reactor safety system, preventing operation when the primary flow is insufficient for heat removal. The normal flow is 40 gpm with a reactor trip set at 30 gpm. A reactor trip will also occur in the event of loss of power to the primary coolant pump.

Each of the six fuel box (2" schedule 40) discharge lines has a type "T" thermocouple (copper constantan) which sends temperature information to the 12 point recorder in the control room. The six fuel boxes flow together into a single 3" schedule 40 pipe which discharges into the primary storage tank. Located in this primary coolant return line is a type "T" thermocouple (No. 8 in Figure 5-1) which monitors the combined coolant bulk temperature, and a primary coolant flow switch which monitors the flow from the core. The information from the thermocouple No. 8 is supplied to the reactor protection system with an alarm setpoint at 150^{o}F and a reactor trip at 155^{o}F. This safety measure prevents reactor operation under conditions such as restriction or reduction of primary coolant flow, reduction or restriction of secondary coolant flow, a malfunction of the heat exchanger, excessive reactor power or the malfunction of a thermocouple.

The flow switch in the coolant return line will also actuate a reactor trip signal in the event of loss of primary coolant flow; this serves as a backup to the low flow reactor trip in the fill line previously discussed and also monitors the integrity of the piping.

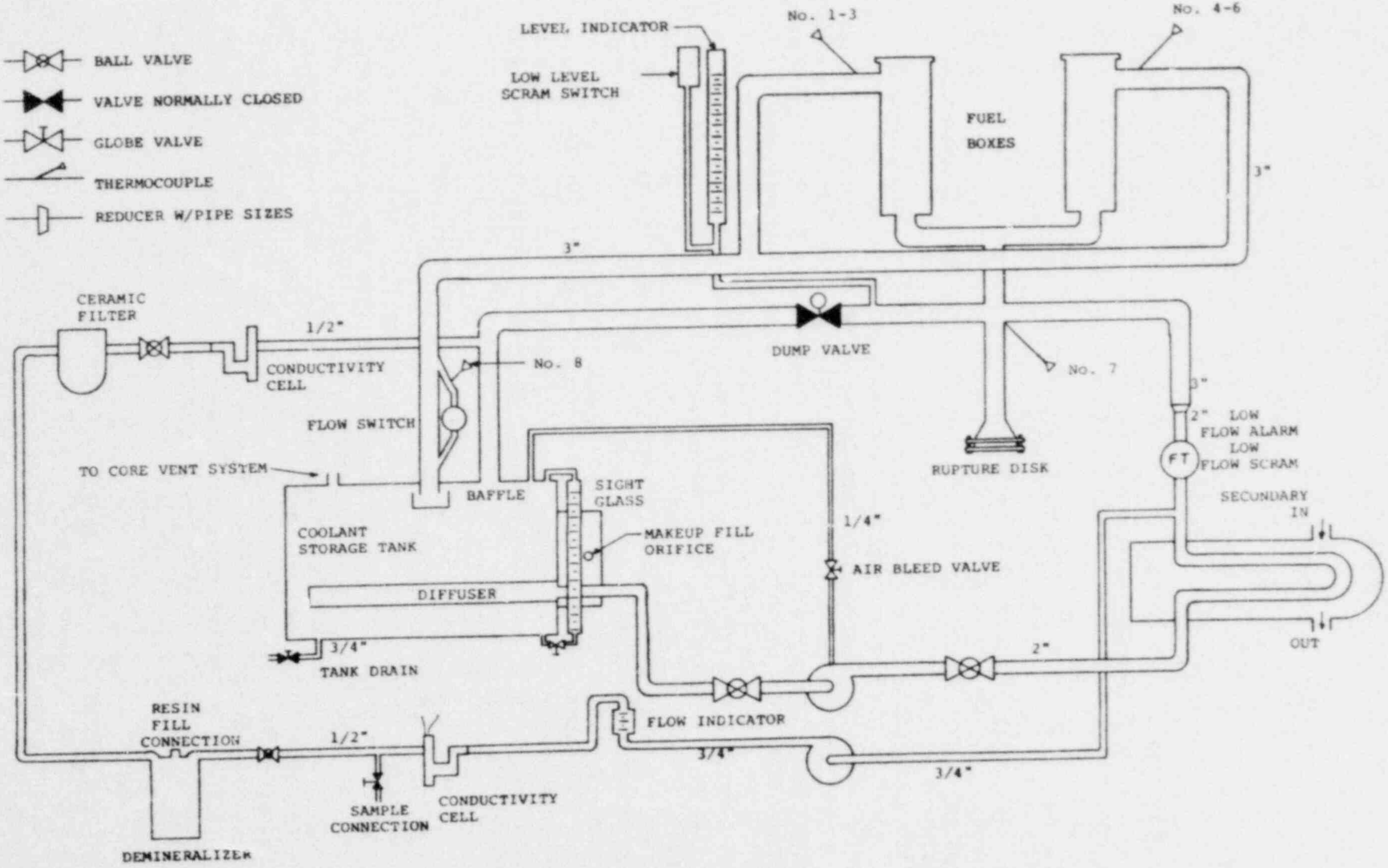
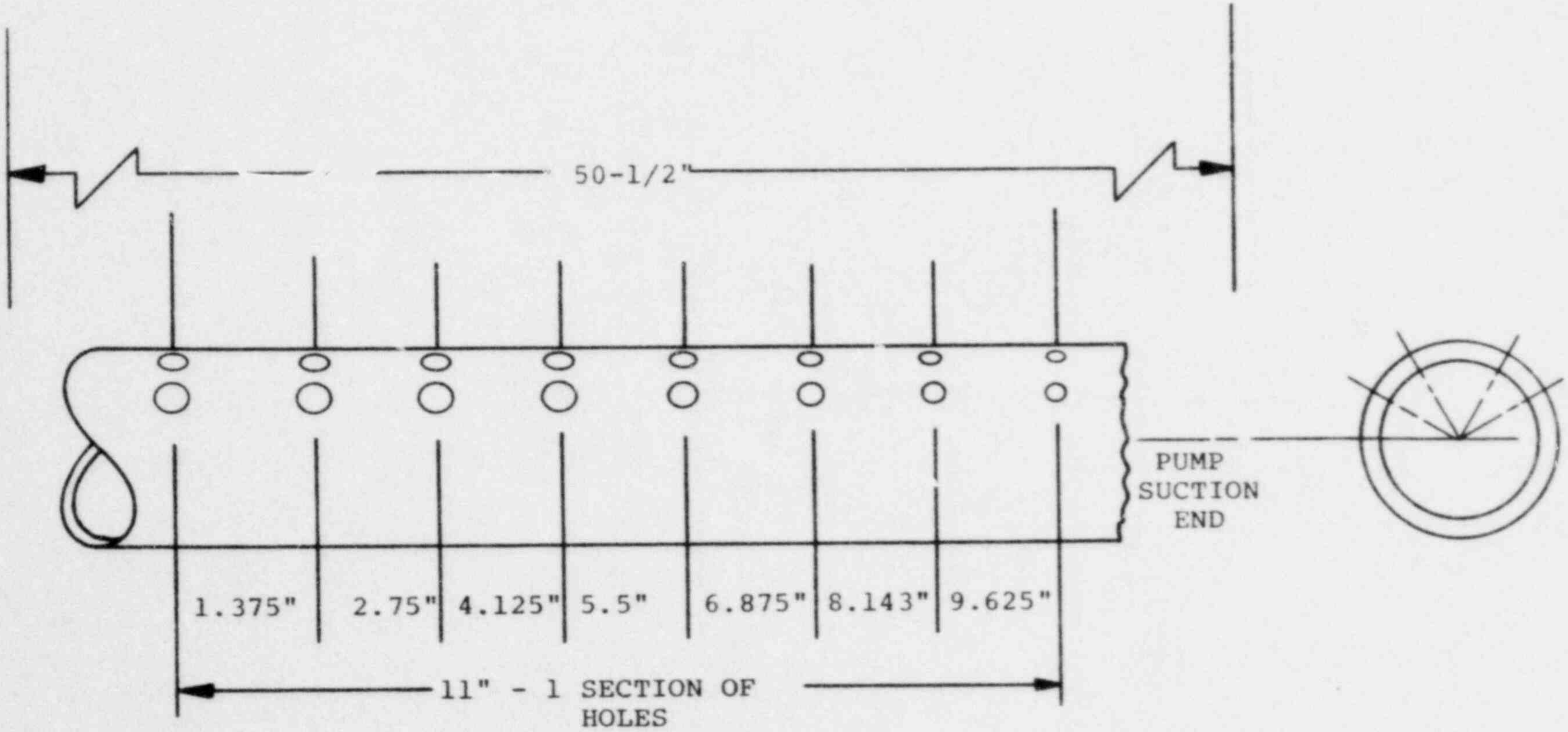


Figure 5-1. Schematic of UFTR Primary Coolant Loop and Purification System.



SECTION A: 8/32" diameter holes
SECTION B: 9/32" diameter holes
SECTION C: 10/32" diameter holes
SECTION D: 11/32" diameter holes

Figure 5-2. UFTR Storage Tank Diffuser Arrangement (Vortex Eliminator)

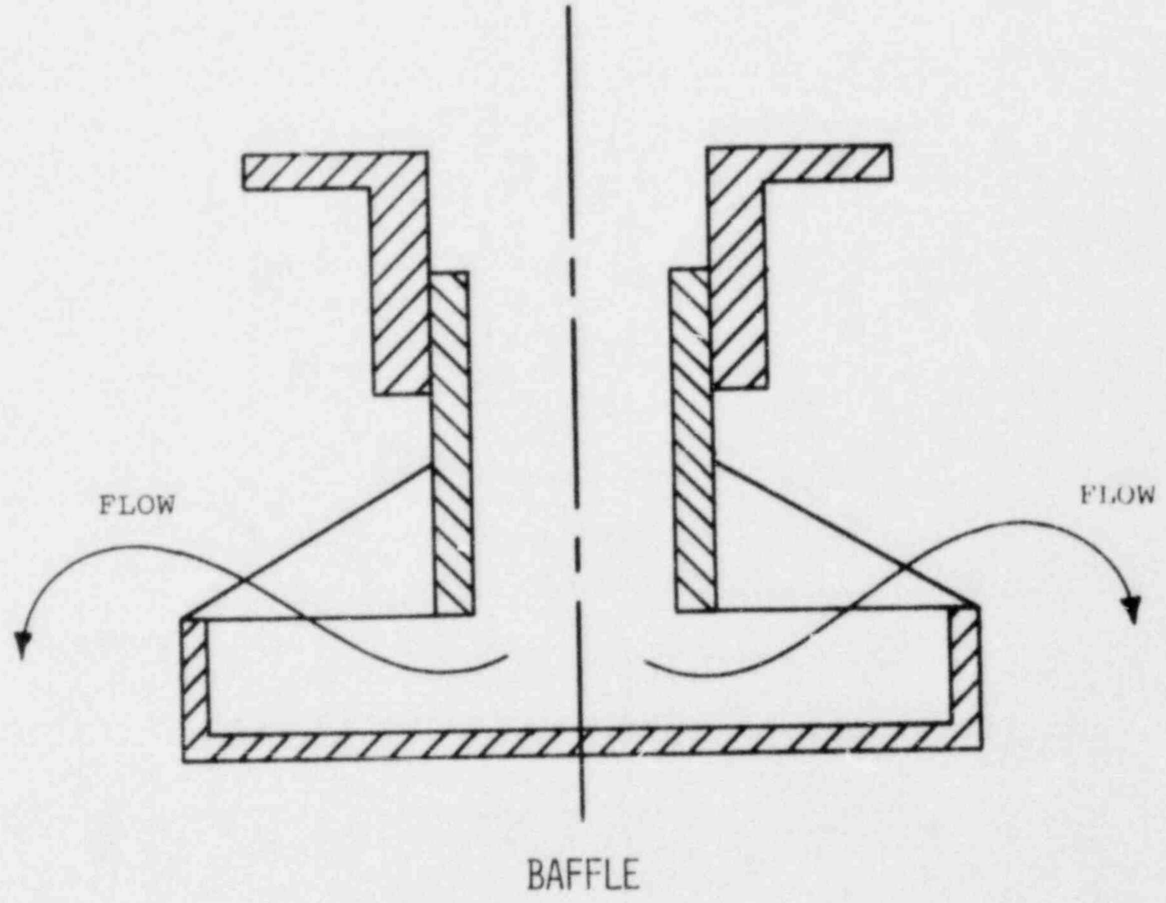


Figure 5-3. UFTR Coolant Storage Tank Aluminum Bucket Baffle.

The "dump valve" (see Figure 5-1) is a solenoid operated valve which opens automatically when a scram signal is generated by the control system, allowing water in the fuel boxes to drain into the coolant storage tank. Only "nuclear type" scrams open the dump valve (high power, fast period, loss of neutron chamber HV). These scrams are now called Full Trips.

A sight glass located on the north wall of the reactor room allows visual check of the reactor core water level. An electric level switch located behind the sight glass is wired to the reactor protection system actuating a reactor trip when the water level in the core falls below preset limits.

The system is further protected by a graphite rupture disc set to burst at 7 psi, two pounds above the normal operating pressure. Should a power excursion occur, this diaphragm will rupture causing the water from the core to be drained into the equipment storage pit, shutting down the reactor. (2)

The primary reactor cooling system does not contain any valves which could be inadvertently left in the wrong position and restrict or shut off the flow of cooling water for the system without activation of the reactor protection system. (2)

5.1.1 Coolant Storage Tank

The primary coolant is stored in the coolant storage tank (see Figure 5-1) which has a capacity of 200 gallons of water; approximately six times the capacity of the reactor. (5)

The storage tank has several features designed to optimize the overall performance of the reactor cooling system and to eliminate undesirable water surges in the core. Special storage tank features include the diffuser illustrated in Figure 5-2 and the baffle illustrated in Figure 5-3. (5)

The diffuser forces the water in the coolant storage tank to diffuse through the input line to the primary coolant pump; the diffuser eliminates the formation of vortices inside the storage tank as a result of the pump's suction. The design specifications of the diffuser are included on the drawing in Figure 5-2. The second storage tank feature is a diffuser or aluminum "bucket" baffle shown in Figure 5-3. This baffle is designed to suppress the splashing of the primary water coming into the coolant storage tank and to change its direction of flow (see Figure 5-1 for location in the coolant storage tank). This eliminates air being trapped in the coolant flowing through the system. (5)

5.1.2 Heat Exchanger

The heat exchanger is a 316 stainless steel water-to-water heat exchanger designed to circulate from 150 to 250 gpm of well water through the shell side and 75 gpm of reactor coolant water through the tube side for removal of up to 500 Kw thermal heat load.

The tubes were seal welded to the tubesheet to minimize leakage. The heat exchanger complies with ASME code, Section III, Class III on the tube side (primary water) and with ASME Section VIII standards on the shell side (secondary water).

The design specifications for the heat exchanger are listed in Table 5-1.

Table 5-1

UFTR Heat Exchanger Design Specifications

Manufacturer:	The Whitlock Manufacturing Company, West Hartford, Connecticut
Model:	Whitlock size 8 - y - 48 Type MAHT-4-B-55
Material:	all 316 stainless steel
Shell Side:	Well water, or back up city water, 1 pass
Tube side:	Primary water, 4 passes
Design Pressure:	100 psi
Test Pressure:	150 psi
Design Temperature:	200°F

5.1.3 Primary Water Makeup System

Demineralized water is used as makeup to the primary coolant system. The makeup system consists of two demineralizers in series filled with amberlite, H-OH, nuclear grade resin. The unit has a hose connection to the coolant storage tank, supplying primary coolant whenever necessary. The schematic of the UFTR primary water makeup system is shown in Figure 5-4. The makeup orifice for the primary system is located on the side of the coolant storage tank as illustrated in Figure 5-1.

5.1.4 Primary Purification System

The primary purification system loop is included in Figure 5-1; it is supplied with a separate pump allowing continuous purification flow. The purification pump is interlocked with the primary coolant pump in a manner that prevents operation of the purification pump when the primary coolant pump is running. The feed of the primary coolant pump is sufficient to maintain a flow through the purification loop when it is in operation.

The purification system is arranged to provide the reactor with continuous monitoring of the resistivity of the primary water and the functioning of the amberlite-nuclear type resin (H-OH; pH-control) in the purification system. The in-line, wall-mounted resistivity bridge is set up to accept two conductivity cell signals--one before the demineralizer and one after the ceramic filter. The location of the purification system and a schematic showing its components is depicted in Figure 5-1.

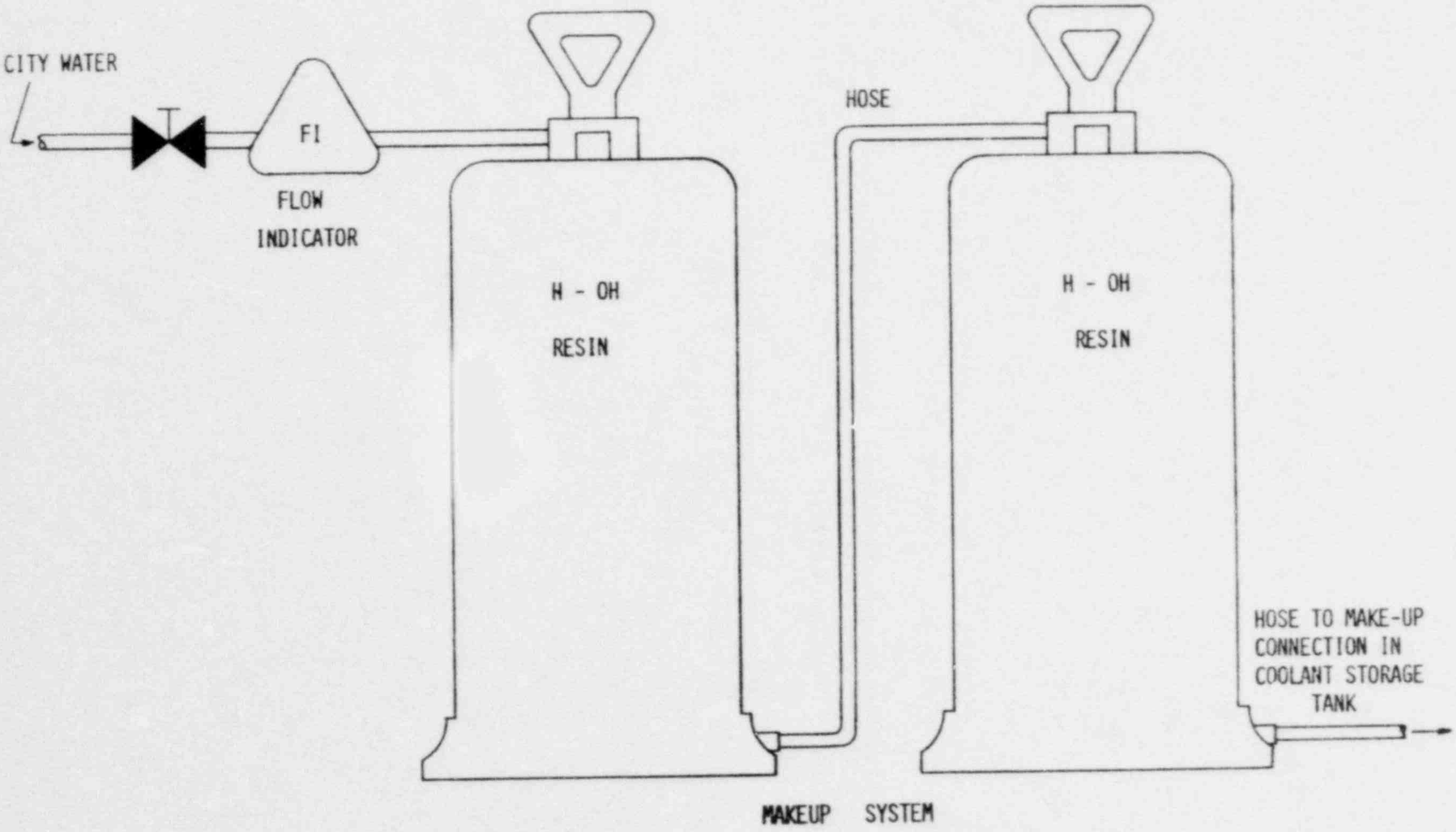


Figure 5-4. Diagram of UFTR Primary Water Makeup System.

5.2 Secondary Cooling System

A system schematic diagram of the secondary cooling system of the UFTR is shown in Figure 5-5. This figure depicts two sources of water for this secondary cooling system: the deep rock well, used during principal operation; and the city water used as a back-up system during operation above 1 kw (thermal). The well water is pumped by a Goulds, Series UG Submersible pump, 10 horsepower. The design specifications of this pump are as follows:

Manufacturer:	Goulds Pump, Inc.
Pump Series:	220 GPM Series UG66L, 10 H.P.
Pump Model:	4 stg. MODEL 2361339000
Capacity Specifications:	218 GPM at TDH, at 0 tank pressure
Operation/Control:	pump on-off from the reactor console

The deep well is 238 ft. deep with a casing diameter of 3", the static water level is approximately 87 ft. below grade. The well pump has approximately 200 gpm pumping capacity for this arrangement. The well water flows through a basket strainer, with a stainless steel mesh of approximately 1/16". This water flows into the shell side of the heat exchanger and subsequently into the storm sewer as depicted in Figure 5-5.

There is a sample flow valve in the heat exchanger discharge line which continuously bleeds a small sample flow into the hold-up sample tank. A second sample valve normally kept closed is used for actual sample collection.

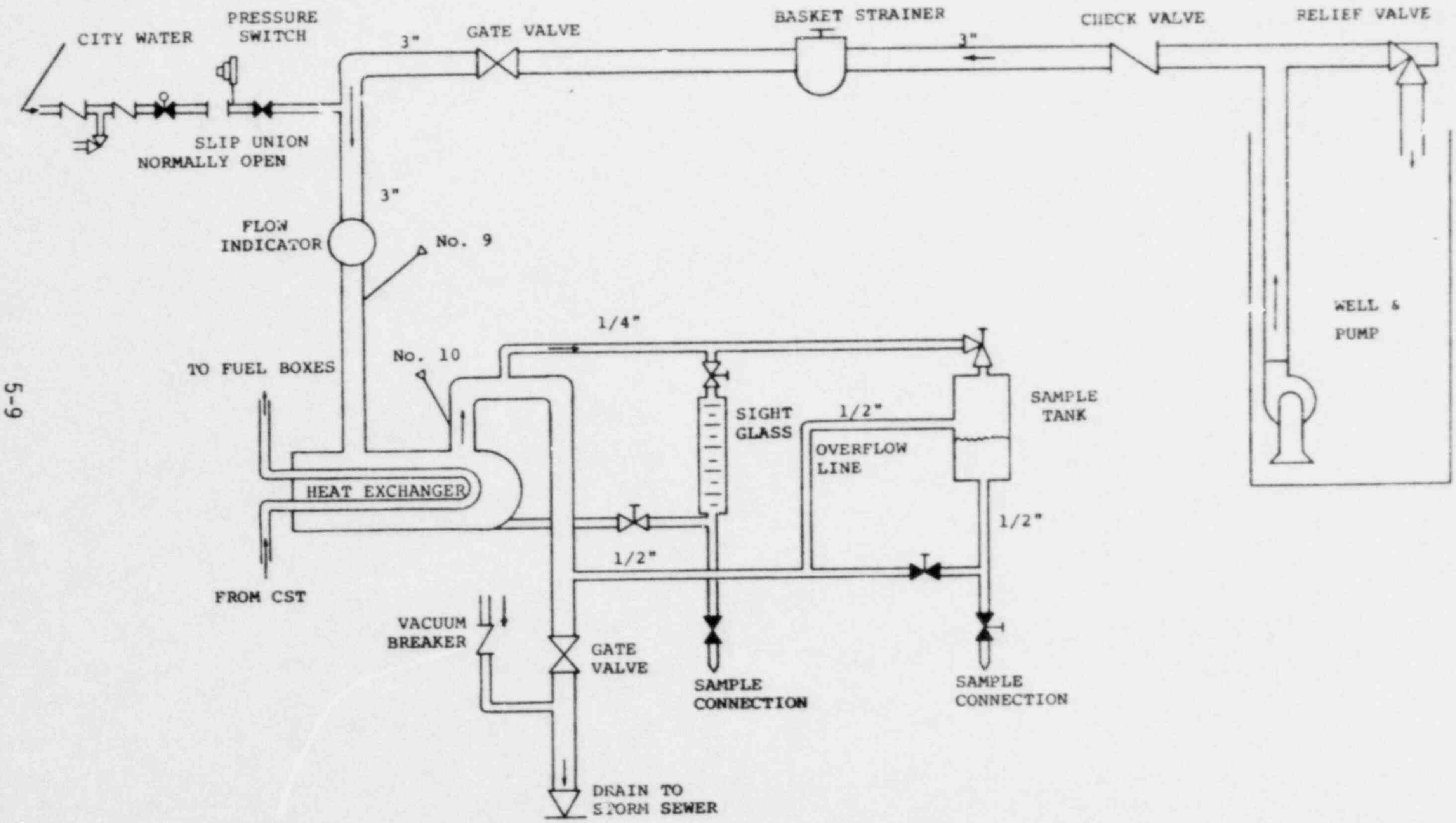


Figure 5-5. Schematic of UFTR Secondary Water Cooling System.

6. ENGINEERED SAFETY FEATURES

The UFTR reactor is a self-limiting research and training reactor which requires no additional engineered safeguards beyond those designed into the reactor core or incorporated into the main cooling, safety, control and radiation monitoring systems. All requisite safety features are described in appropriate places in the rest of this Safety Analysis Report.

7. INSTRUMENTATION AND CONTROLS

7.1 Introduction

The reactor i strumentation monitors several reactor parameters and transmits the appropriate signals to the regulating system during normal operation, and during abnormal and accident conditions to the reactor trip and safety systems. Since the UFTR is a low power, self-limiting reactor, the instrumentation and associated controls are considerably simplified when compared to instrumentation and control systems of large power reactors.

7.2 Identification of Safety-Related Systems

The safety-related instrumentation and controls for the UFTR include the control console, the control and safety channels, the facility interlock system, control drive switches, and the reactor scram circuitry. Table 7-1 contains a list of abbreviations used in the UFTR instrumentation and control diagrams; it is repeated from Chapter 1 for completeness and ease of reference in this chapter. Figure 7-1 shows a block diagram of the nuclear instrumentation and scram logic of the UFTR.

7.2.1 Console

All the functions essential to the operation of the UFTR are controlled by the operator from a desk-type control console. The reactor console is conveniently located near the reactor to allow the reactor operator to monitor activities in the reactor cell during operation. All instrumentation contained in the console accepts or sends signals from or to the control rod drives, the reactor interlock system, and various detectors and transducers located around the reactor core and the reactor coolant system.

The reactor control panel contains the following control and indicating instrumentation:

1. A console power (POWER ON) switch.
2. A three-position OFF/OPERATE/RESET key switch.
3. A set of four control-blade switches for the three safety blades (1,2, and 3) and the regulating blade. One set of switches for controlling the secondary system city water valve.
4. Four control blade position digital indicators
5. A MODE SELECTOR switch (mode switch) for automatic or manual operation.
6. A REACTOR POWER range switch (range switch).
7. A dual-pen strip-chart recorder.
8. A %-DEMAND control potentiometer.
9. A manual SCRAM bar.
10. A REACTOR PERIOD meter and calibrate/test controls.
11. A set of scram (14) and blade interlock (3) annunciator lights, left panel.
12. Safety Channel Meter #1 and test controls.
13. Safety Channel Meter #2 and test controls.
14. Log Power Meter and calibrate controls.
15. Reactor cell entrance/exit door monitors.
16. Reactor equipment control switches and annunciator lights, right panel.

TABLE 7-1

ABBREVIATIONS USED IN UFTR

INSTRUMENTATION AND CONTROLS DIAGRAMS

AMM	AMMETER
AMP	AMPLIFIER
AUTO	AUTOMATIC
B/S	BISTABLE
CAL	CALIBRATE
CIC	COMPENSATED ION CHAMBER
COMPA	COMPARATOR
COMPUT	COMPUTER
CPS	COUNTS PER SECOND
DN	DOWN
HV	HIGH VOLTAGE
INT'LK	INTERLOCK
LIN	LINEAR
LOG	LOGARITHMIC
MAG	MAGNETIC CLUTCH
MAN	MANUAL
NI	NUCLEAR INSTRUMENTATION
P/S	POWER SUPPLY
PA	POWER AMPLIFIER
PC	PRIMARY COOLANT
PWR	POWER
REG	REGULATING ROD
RPI	CONTROL BLADE (ROD) POSITION INDICATION
UIC	UNCOMPENSATED ION CHAMBER
W/D	WITHDRAWAL
W/R	WIDE RANGE DRAWER (CHANNEL)

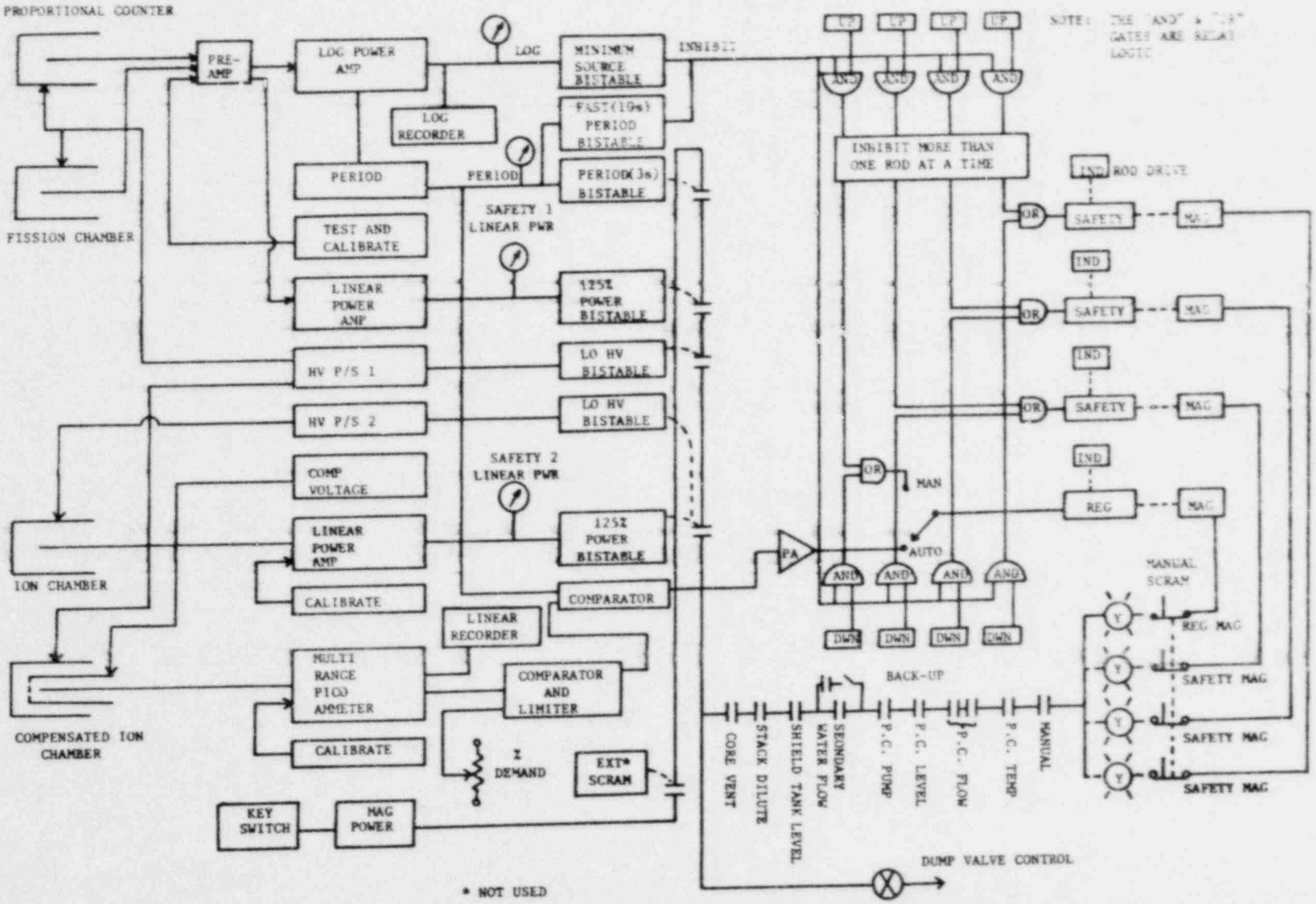


Figure 7-1. Overall UFTR Instrumentation and Scram Logic Diagram.

The functions of these control and indicating devices are summarized in the following paragraphs.

The console POWER ON switch controls a.c. power to all control and equipment circuits. The nuclear instrumentation channels receive power from the circuit breaker on the console rear center door.

Control blade magnet power is controlled through the three-position OPERATE key switch.

The control blade switches (UP, DOWN, and ON) are provided for the safety 1, safety 2, safety 3, and regulating blades. The positions of the control blades relative to their lower limits are indicated on individual digital blade POSITION indicators mounted on the control panel.

A two-position MODE SELECTOR switch is located in the lower left corner of the central control panel. The switch is used to select one of two modes of operation for the reactor: MANUAL-AUTOMATIC.

A REACTOR POWER range switch, with seventeen steady-state positions (.001 watts to 100 Kw), zero and calibrate, is located in the lower right corner of the horizontal portion of the control panel. It is used in conjunction with the linear amplifier. The dual-pen strip-chart recorder is centrally located in the upper center portion of the console. The red pen provides a linear indication of power as a percentage of the range switch's position and the purple key provides a 10 decade logarithmic display of reactor power level.

THE %-DEMAND control in the upper right center section of the console is used in conjunction with the steady-state automatic control servo to maintain the desired power level during operation above 1 watt.

The SCRAM BAR provides a means of manually scramming the reactor. This is a safety-related provision considered necessary for all licensed reactors.

A LOG POWER meter ranging from 0.1 to 10^3 counts/second and a power range of 10^{-8} to 125% rated power, is located on the left side of the control panel along with the REACTOR PERIOD meter, which provides an indication of the rate of power change and ranges from periods of -30 seconds (subcritical) to infinity to +3 seconds (supercritical).

The SAFETY CHANNEL meters #1 and #2 range from 0 to 150% power and are located on the right side of the control panel. A set of 17 annunciator lights is located on the left side of the dual pen recorder. These annunciate all scrams and blade interlocks. Three indicators on the right side of the panel indicate use of the three possible entrances and/or exits to the reactor cell and control room.

7.2.2 Nuclear Instrumentation Channels

The two channels of neutron instrumentation shown in Figure 7-1, the Nuclear Instrumentation and Scram Logic Diagram, provide the UFTR with independent, separate neutron monitors of the reactor power level. Figure 7-2 shows the operating ranges of the detectors used to monitor UFTR power levels.

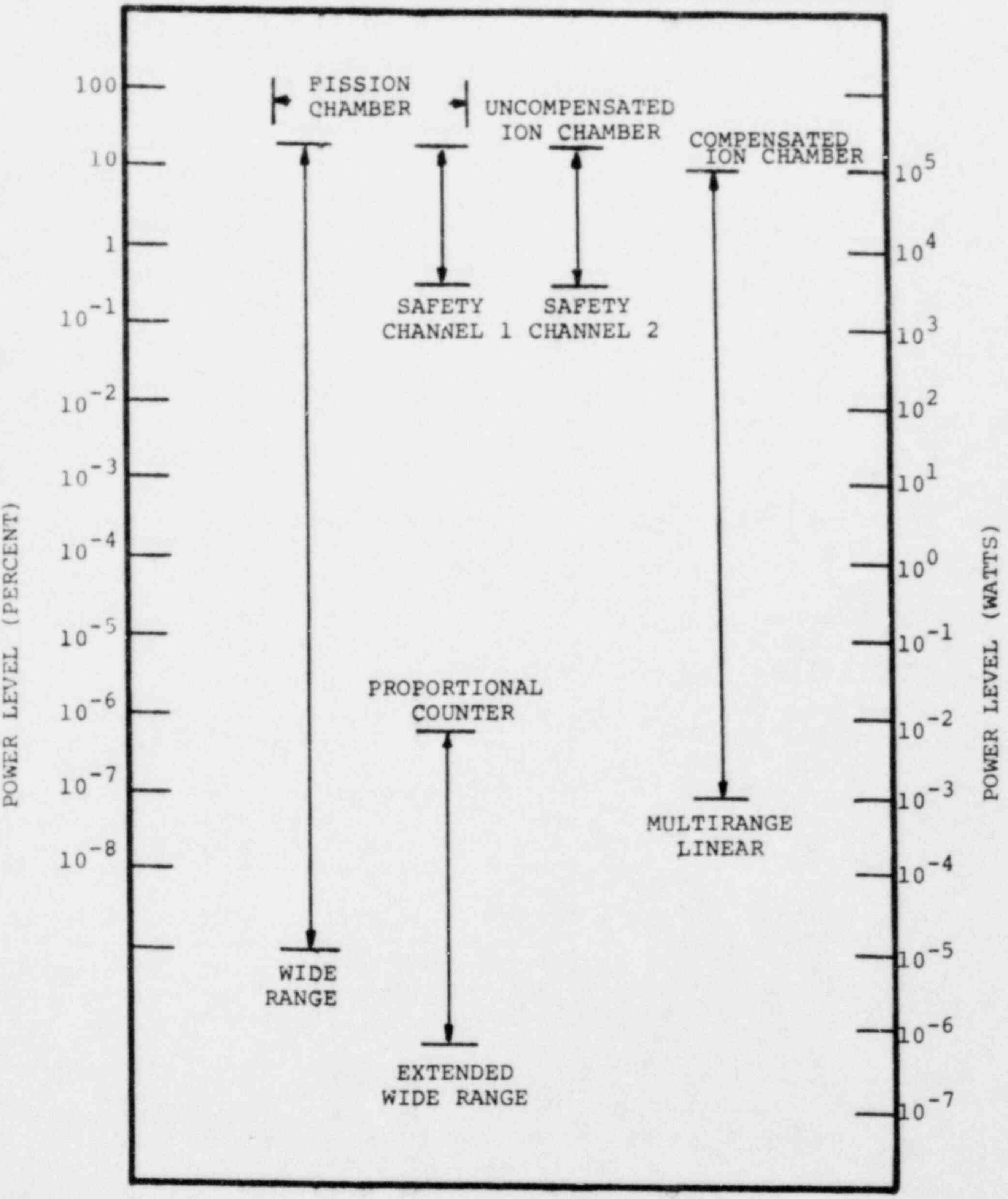


Figure 7-2. Operating Range of UFTR Neutron/Power Level Detectors.

7.2.2.1 Nuclear Instrumentation Channel 1. As indicated in Figure 7-3, Nuclear Instrumentation Channel 1 monitors the rate of growth of the neutron flux or power level. Reactor trips which operate on a one-out-of-one logic, are provided in this channel for any of the following three occurrences:

1. A fast period (3 seconds),
2. UFTR Reactor Overpower (125% rated power/125 Kw),
3. A 10% loss of high voltage to the neutron detection chambers.

These reactor trips are present to insure the safety of the UFTR facility by preventing the reactor power from exceeding design levels. A 2 cps neutron interlock assures that a reactor start-up can be made only if neutron source counts are sufficient and the low level neutrons monitoring channel is properly functional. The main components of Nuclear Instrumentation Channel 1 and their functions are described in the following three subsections.

7.2.2.1.1 Log Power (Wide Range Channel). The log power channel depicted in Figure 7-3 provides the reactor operator with a continuous display and record of neutron flux from source level to full power. The circuit consists of a B-10 proportional counter(for low levels), a pre-amplifier, a log and period fission chamber, a derivation amplifier, and the purple pen (second) channel of the dual pen recorder.

7.2.2.1.2 Period Channel. For the period channel shown in Figure 7-3, the log-n amplifier produces a voltage proportional to the logarithm of neutron flux. A derivative circuit produces a voltage proportional to the inverse of the reactor period, which is then amplified and displayed on a control panel meter that ranges in seconds from -30 to ∞ to + 3 sec. An adjustable bi-stable trip activates a scram, currently set at +3 seconds, as determined by the Technical Specifications.

7.2.2.1.3 Safety Channel #2. The linear channel shown in Figure 7-3 is applied as a safety channel by using the D.C. component of the signal from the wide range fission chamber. As shown in the NI Channel 1 diagram of Figure 7-3, the linear amplifier accepts the linear current signal from the pre-amplifier. The output signal is then displayed as the power level on a linear scale ranging from 1 to 150% of rated power. A reactor trip is set at 125% rated power (125 Kw) resulting from operation of a bi-stable trip.

7.2.2.2 Nuclear Instrumentation Channel 2. As shown in Figure 7-4, Nuclear Instrumentation Channel 2 is used to monitor the neutron level or power level of the UFTR and maintain a steady power level through the reactor steady-state automatic control servo system. The main components of the NI Channel 2 are described with their functions in the next two subsections.

7.2.2.2.1 Linear Power Channel. The linear power channel provides power level indications from just above source level to 100 Kw. As indicated in Figure 7-4, the linear power circuit consists of a neutron-sensitive compensated ion chamber, a pico-ammeter with a 17 position range switch and the red pen channel of the 2 pen recorder which reads the power as a percentage of where the range switch is set on the recorder. The pico-ammeter sends a signal, which is a function of a linear indication of reactor power, to the servo amplifier as a part of an automatic reactor control circuit. At the

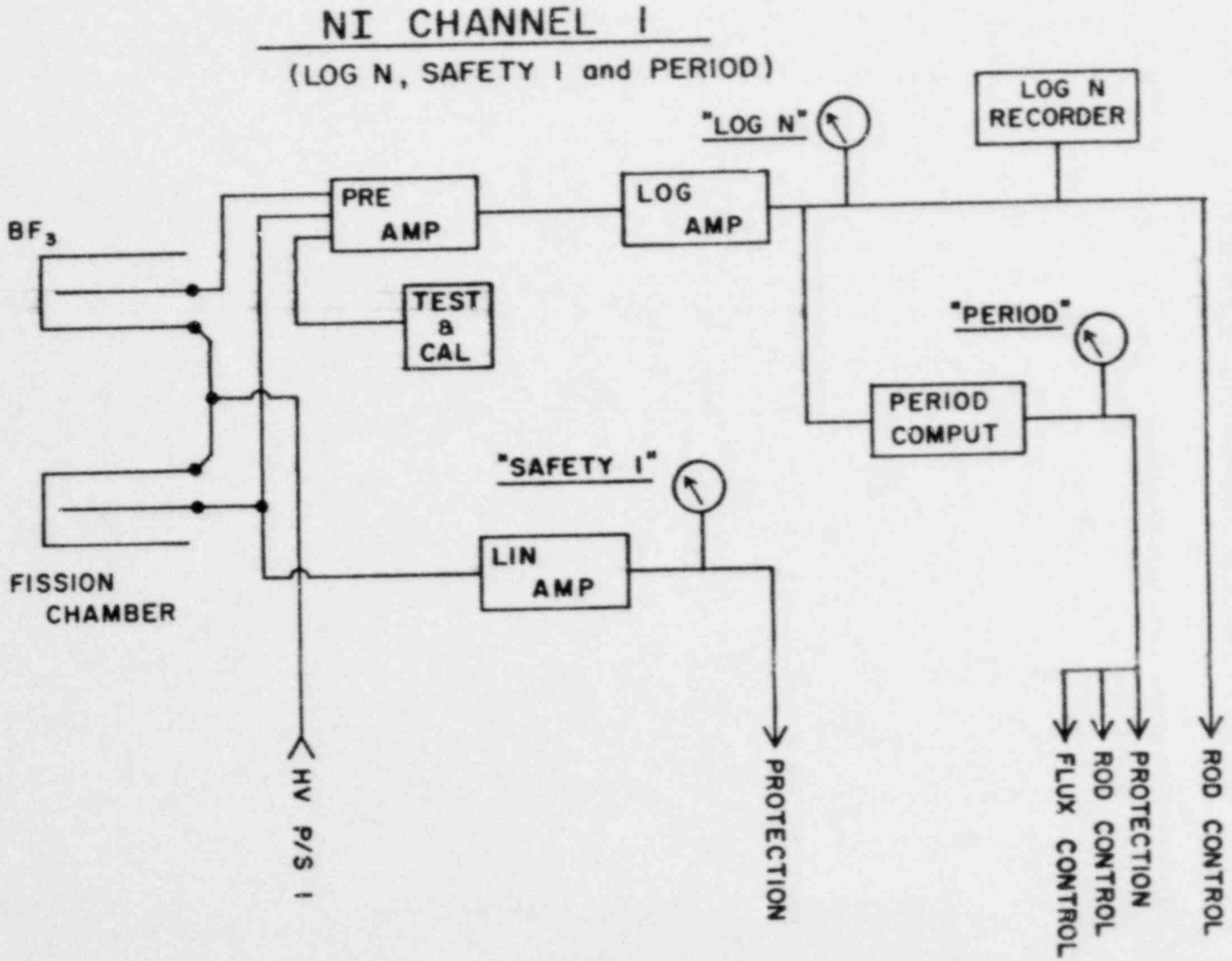


Figure 7-3. UFTR Nuclear Instrumentation Channel 1 Schematic (Log N, Safety #1 and Period Channels).

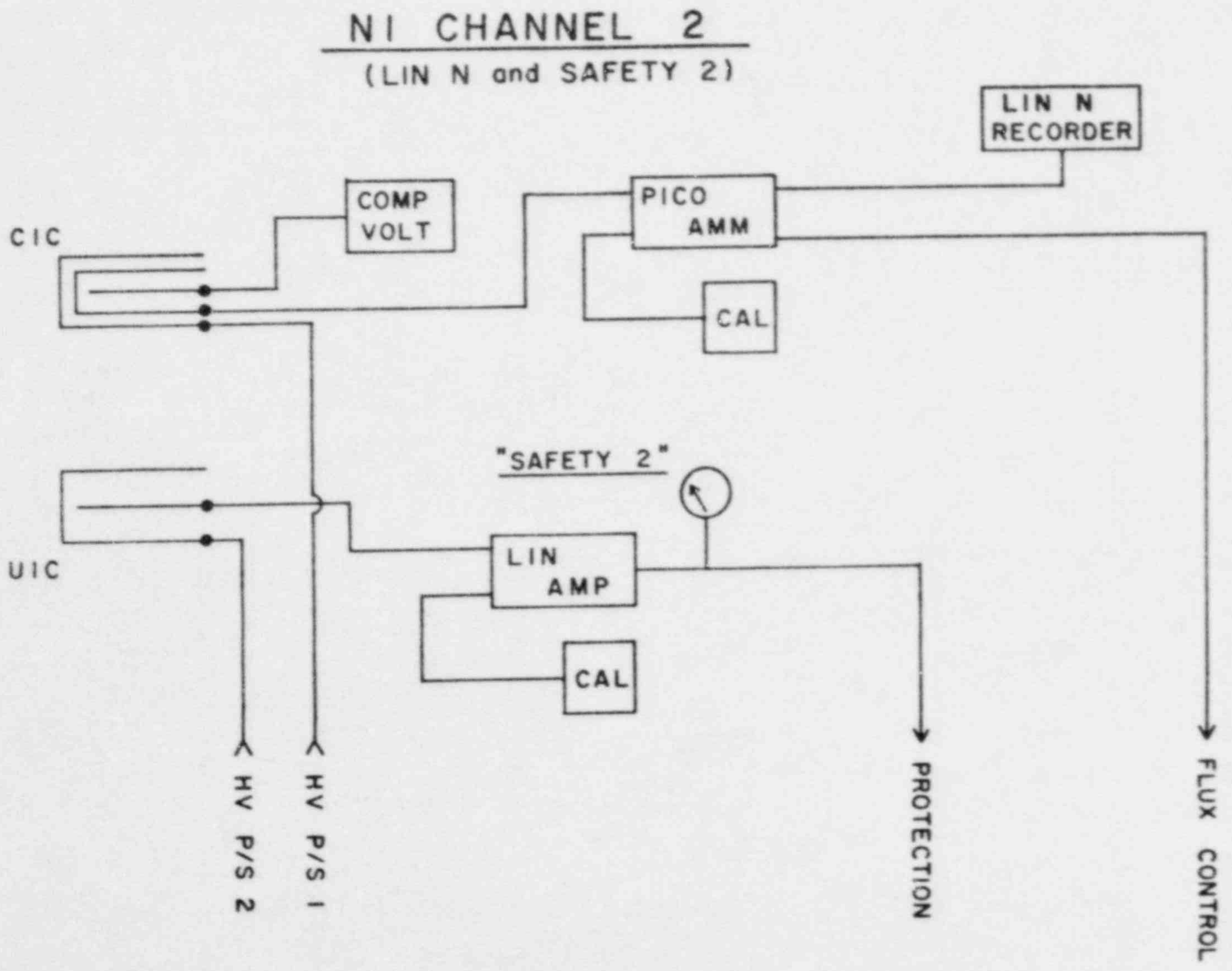


Figure 7-4. Nuclear Instrumentation Channel 2 (Linear N and Safety #2 Channels).

servo amplifier, the signal is compared with the signal from the servo flux control.

7.2.2.2.2 Safety Channel #2. As indicated in Figure 7-4, the safety channel receives a signal from an uncompensated ion chamber and consists of the ion chamber (with an independent high voltage supply), an operational amplifier, an adjustable bi-stable trip, and a meter ranging from 1% to 150% rated power. The Safety Channel #2 system initiates a reactor trip at 125% power. Safety Channel #2 also initiates a reactor trip whenever the high voltage applied to the chamber drops by 10%. The channel also generates test signals to check the functioning of the channel.

7.2.3 Non-Nuclear Instrumentation Channels

The UFTR is supplied with several process instrumentation channels to monitor the normal operation of the various systems; to aid in maintaining a steady-state power level, and also trip the system whenever an unsafe situation occurs or an instrument fails. Other channels supply information needed to safely operate the reactor but do not have protective functions. These Non-Nuclear Instrumentation Channels are described in the next three subsections.

7.2.3.1 Control-Blade Drive System. The control-blade drive circuit is shown in Figure 7-5; it consists of switches and indicating devices used in operating the four control blade drives. The twelve illuminated push button switches are arranged in the center of the control panel in four vertical rows, one row for each control rod. Each row of switches contains a white DOWN switch, a red UP switch, and a yellow ON (magnet on) switch.

When the ON push buttons are depressed, magnet current is interrupted and the ON lights extinguished. If the control rod is above its down limit, the blade will gravity fall back into the core. Turning off the reactor key has the same effect. In the event of a loss of power, these blades fail safe, falling into the core by gravity.

7.2.3.2 Control-Blade Withdrawal Inhibit System. The Control Blade Withdrawal Inhibit System is depicted in Figure 7-5; this Inhibit System is part of the reactor protection system and functions in the following situations:

1. Test switches are not in operate position to insure the monitoring of the neutron level increases as the blades are raised.

2. Insufficient neutron source counts to insure the proper function of the source level instrumentation. A minimum of 2 counts per second is required by the technical specifications.

3. A multiple blade withdrawal interlock is provided to prevent exceeding the reactivity addition rate authorized by the UFTR Technical Specifications.

4. A period of 10 seconds or faster prevents control blade withdrawal.

5. Power is raised at a period in the automatic mode faster than 30 seconds. The automatic controller drives the regulating blade down until the period is slower than 30 seconds.

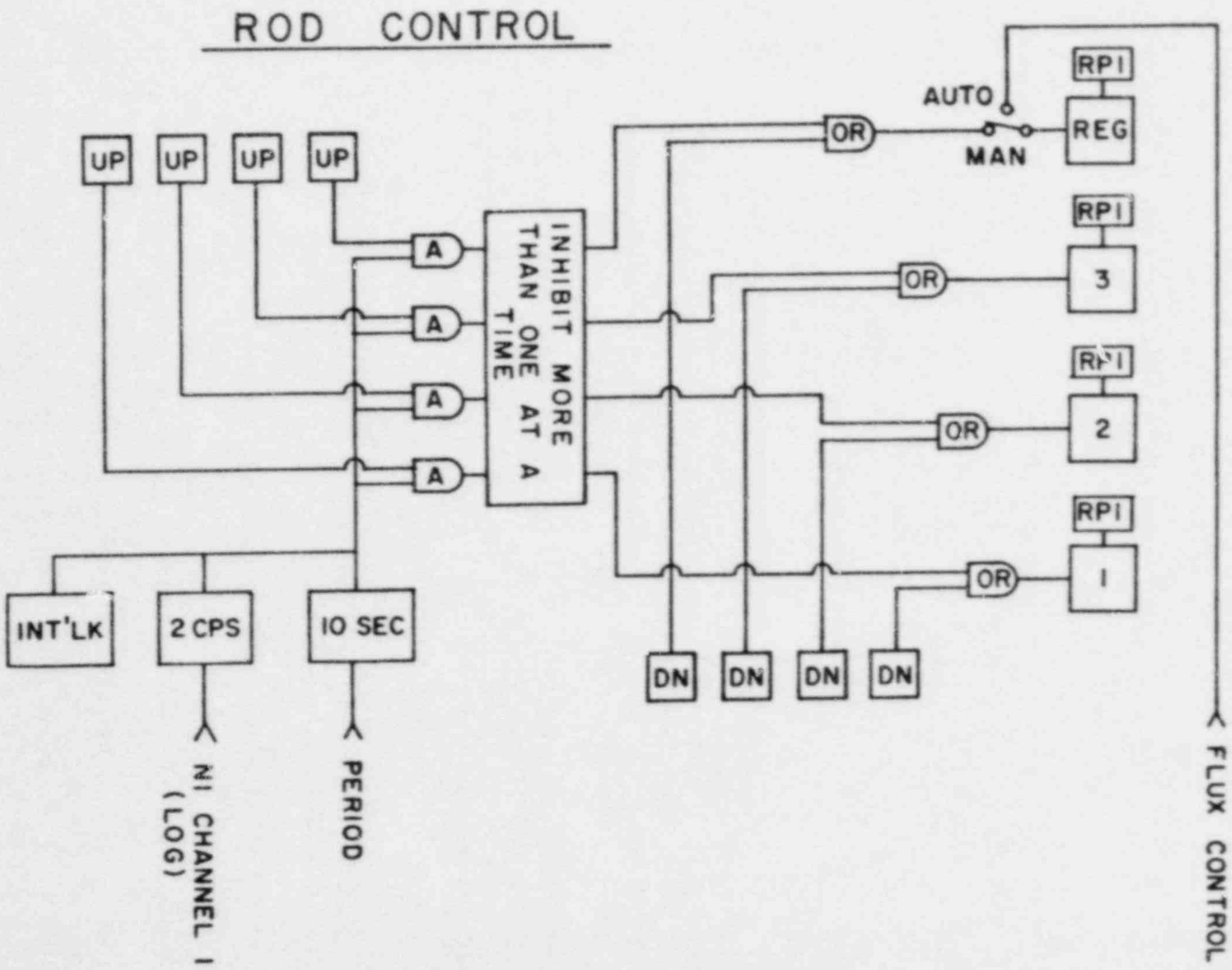


Figure 7-5. ROD CONTROL: UFTR Control-Blade (Rod) Withdrawal Inhibit System.

7.2.3.3 Automatic Control System. The UFTR Automatic Control System is used to hold reactor power at a steady level during extended runs and may be used to make minor power changes within the range of the switch setting. A manual override of the Automatic Control System by operation of the blade drive switches is not possible while the control mode switch is in AUTOMATIC. The control mode switch must be placed back to MANUAL before the Automatic Control System can be overridden; for this prupose, the neutron flux shown in Figure 7-6 controller compares the linear power signal from the pico-ammeter with the power demand signal and moves the regulating blade in order to reduce any difference, therefore maintaining a steady power level.

A primary coolant flow monitor, with a sensor located in the primary coolant fill line, prevents reactor operation or trips the reactor if flow is below the set point of 30 gpm (normal flow 40 gpm).

A coolant flow switch, located in the return line of the primary coolant to the primary coolant storage tanks, initiates a reactor trip in case of a loss of return flow. This flow switch serves as a backup for the low flow reactor trip in the fill line and activates only after the return line has been drained of water or the flow is reduced to less than about 10 gpm.

A sight glass, attached to the north wall of the reactor room, at the east side of the primary equipment pit, shows the water level in the core allowing a visual check of the primary coolant level.

An electric switch which is located behind the sight glass is wired to the reactor protection system. It prevents reactor operation, or activates the reactor trip system, when the water level in the core is below pre-set limits.

Type "T" (copper-constant) thermocouples are located at each of the fuel box discharge lines to monitor water temperature from each fuel box to the primary coolant storage tank and 2 thermocouples monitor the temperature of the bulk-primary water to and from the core. The temperature information is sent to the 12 point recorder in the reactor control room. If any temperature point of the recorder exceeds preset levels, an audible alarm is set off at 150^{o}F, and the reactor trips at 155^{o}F. Conditions which may cause this excessive increase in the primary coolant temperature include:

1. Restriction or reduction of primary coolant flow,
2. Reduction of or restriction of the secondary coolant flow,
3. Malfunction of the primary heat exchanger,
4. Excessive power level, or
5. Instrument malfunctions or thermocouple failure.

A water level switch in the top of the reactor shield tank will trip the reactor when the water level drops below a preset value. This prevents reactor operation because of water loss due to evaporation or leakage.

7.3 Reactor Trip System

The UFTR facility is provided with two types of reactor trips, both initiating the gravity insertion of all the control blades into the core. These reactor trips can be classified into two categories:

FLUX CONTROL

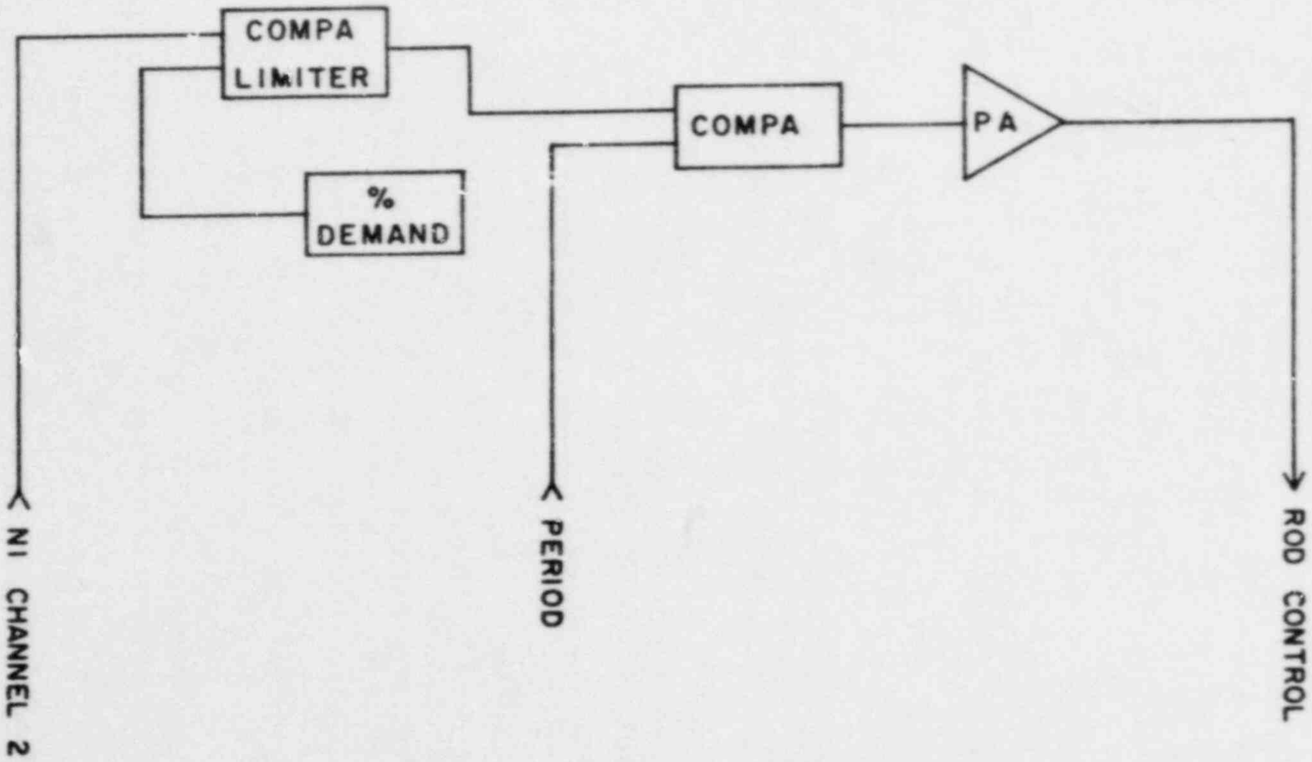


Figure 7-6. FLUX CONTROL: Schematic for UFTR Neutron Flux Control System.

1. Nuclear Instrumentation Induced Trips, which involve the insertion of the control blades into the core and the dumping of the primary water into the storage tank (this type of trip will dump primary water only if 2 or more control blades are not at bottom position);

2. Process Instrumentation Induced Trips, which involve only the insertion of the control blades into the reactor core (without dumping of the primary water). Figure 7-7 shows a schematic diagram of the Protection System provided for the UFTR.

7.3.1 Nuclear Instrumentation Induced Trips (Full Trips)

One of four conditions must exist for the initiation of the Reactor Trip System with dump of primary water (Nuclear-Type Trip); these four conditions include:

1. Fast Period (3 seconds or less),
2. High power, safety channel #1 (125%) or safety channel #2 (125%),
3. Reduction of high voltage to the neutron chambers of 10% or more,
4. Turning off the console magnet power switch.

7.3.2 Process Instrumentation Induced Trips (Rod-Drop Trips)

The conditions which must exist for the initiation of the Reactor Trip System without dump of primary water (process type trips) include:

1. Loss of power to the Reactor Vent Blower System.
2. Loss of power to Reactor Vent Diluting System.
3. Loss of power to the secondary system deep well pump when operating at or above 1 Kw and using this system for secondary cooling.
4. Dropping of secondary flow below 60 gpm (normal flow 200 gpm, alarm at 140 gpm) when operating at or above 1 Kw when using this system for secondary cooling.
5. Dropping of secondary flow below 8 gpm when at or above 1 Kw when using city water for secondary cooling.
6. Drop in water level of the shield tank (about 4 in.)
7. Loss of power to primary coolant pump.
8. Reduction of primary coolant flow (normal 40 gpm, trip at 30 gpm); flow sensor is located in the fill line.
9. Loss of primary coolant flow (return line).
10. Reduction of primary coolant level.
11. High temperature primary coolant return from the reactor (alarms at 150°F, trips at 155°F).
12. Manual reactor trip button depressed.
13. A.C. power failure (fail-safe criterion).

A set of annunciator lights located on the left side of the control console indicates all scrams and 3 interlock conditions. In case of high reactor temperature, an audible alarm is set off at 150°F and the reactor trips at 155°F. The alarm continues to sound until the indicated temperature drops below 150°F.

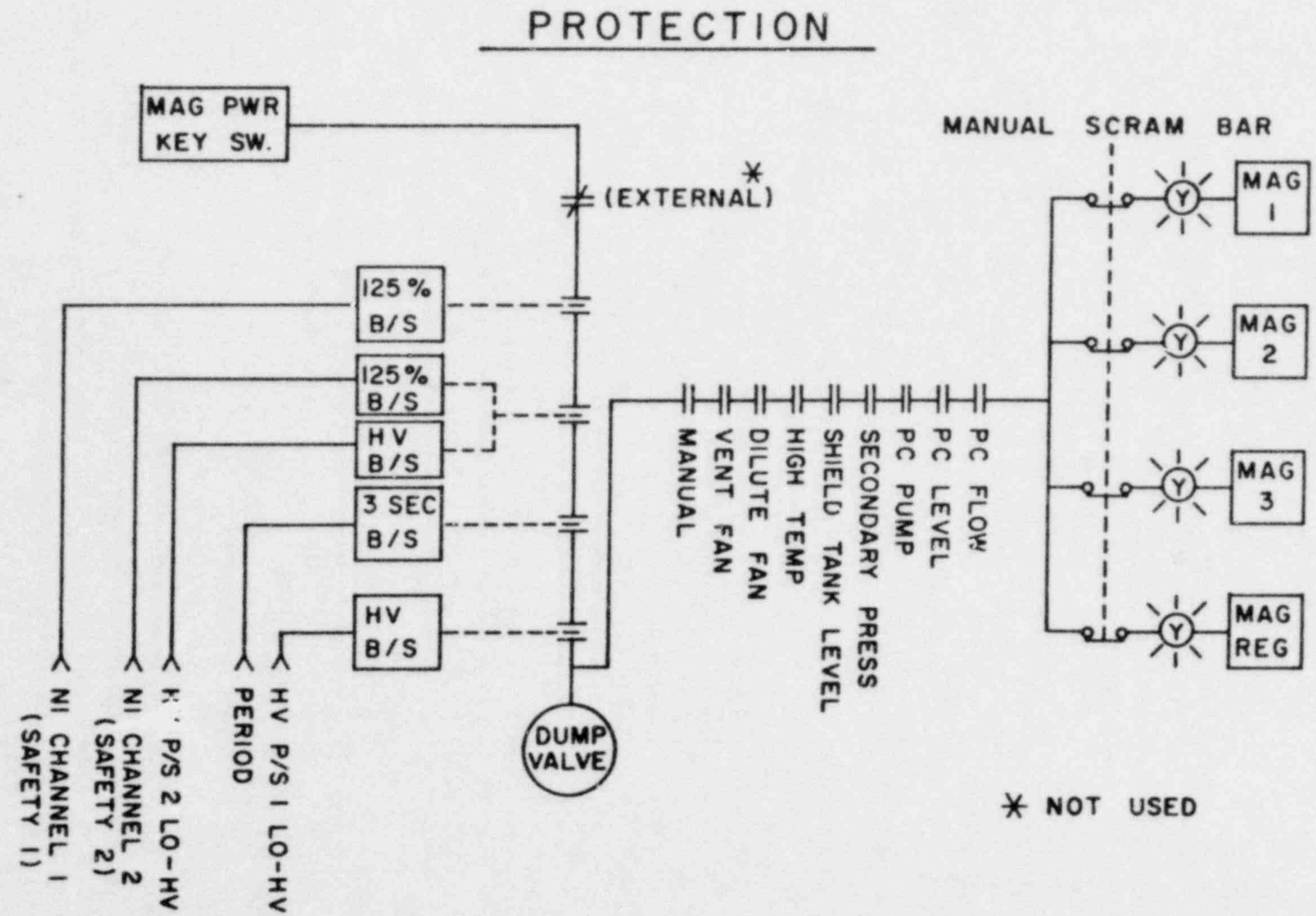


Figure 7-7. PROTECTION: Schematic Diagram of the UFTR Reactor Protection System.

A red rotating beacon located in the reactor cell together with three "reactor on" lighted signs located on the outside of the east side of the Reactor Building on the second floor level, on the entrance hallway leading to the control room, and on the north outside reactor building wall, are all energized whenever the console key switch is turned to the "ON" position.

7.4 Engineering Safety Feature System

As explained in Chapter 6, there are no separate Engineered Safety Features required in the UFTR aside from those built-in into the facility. Therefore, no instrumentation or control system relative to this system is present.

7.5 Systems Required for Safe Shutdown

The only system required for normal safe shutdown is the safety-control blades drive instrumentation channels allowing the operator to insert the blades into the core to shut the UFTR system down. Proper rod movement can be observed in the display panel where the four rod position indicators are located. In addition, the nuclear instrument channel read-outs provide another way for determining proper decrease in power for reactor shutdown. Nevertheless, the only system really necessary for reactor shutdown is the control rod drive system. In case of failure of this system on a loss of power, the control rod system is designated to fail safe; the blades drop by gravity into the system to shut the reactor down. A semi-annual measurement is made of blade drop times which must be less than 1 second. Normal times are about 0.5 second. If the control blades do not function properly and the core overheats, the negative void and temperature coefficients will cause the core to go subcritical and shut down even without insertion of the control blades. Therefore, instrumentation is not an absolute necessity for shutting the UFTR down because of its inherent safety features. In addition, the reactor can be made subcritical and power reduced by the operator initiated action of dumping the primary coolant.

7.6 Safety-Related Display Instrumentation

Readouts from all of the nuclear instrumentation and non-nuclear instrumentation channels are displayed on the reactor console as described in Section 7.1.1.1.

The reactor vent system has a GM detector and preamplifier, which transmits a signal to the control room to monitor the gamma activity of the effluent in the downstream side of the absolute filter, before dilution occurs. If the activity reaches alarm level preset in the control room, the monitor will actuate an audible alarm in the control room. The data from this monitor is continuously recorded.

The stack monitoring system consists of a GM detector, a log rate meter, and a strip chart recorder. It also provides a log rate meter with an alarm setting capability for the different powers of operation, monitoring the gross activity concentration of radioactive gases in the room effluent air entering the stack.

A complete area radiation monitoring system consisting of three independent area monitors with remote detector assemblies and interconnecting cables, and strip chart recorders and count rate meters are available. The signals from these detectors are sent directly to the log count rate meter and recorder, monitoring the gamma activity in the reactor room. Each detector has an energy compensated Geiger Counter with built-in Kr-85 check source which can be operated from the control room. The stack monitor and 3 area monitor modules in the control room are equipped with test switches and green "NO FAIL" lights that go out if the modules do not receive signal pulses from the detectors. Floating battery packs supply power to the units in the event of electrical power loss.

The air monitoring system is equipped with a flow indicator (LPM), a strip chart recorder and an audible and visible alarm setting. The monitor is a lead-shield, compact airborn particulate Geiger Counter.

The portal monitoring system in the airlock leading from the reactor cell is a Beta-Gamma Portal Monitor Model PCM-4A console and portal frame. It contains eight channels of geiger tube detectors providing complete head to foot coverage of beta-gamma radiation plus individual alarm lights for each channel. An audible alarm will be activated any time the preset radiation field limit is exceeded.

7.7 All Other Instrumentation Systems Required for Safety

There are no other instrumentation systems required for the safe operation of the UFTR; all the necessary instrumentation has been covered in previous sections of this chapter.

7.8 Control Systems Not Required for Safety

There are no control systems in the UFTR facility which do not have safety-related functions as considered in this Safety Analysis Report. Consequently, all UFTR control systems have already been described in the preceding sections. Even those controls which do not have a safety operational function do have a safety function in the sense of providing information on safe UFTR operation through read-outs supplied by the appropriate monitoring control.

8. ELECTRIC POWER

8.1 Introduction

The UFTR is a research reactor presently licensed to operate at only 100 Kw (thermal), and it does not generate electric power. Since the UFTR does not generate electrical power, there is no impact on the power grid. The reactor is designed to withstand any credible accident and is designed to shut itself down safely through operation of the reactor safety systems in case of loss of primary coolant.

8.2 Offsite Power System

During operation, the electric power requirements for the UFTR reactor will be supplied by the regional utilities servicing the University. The reactor facility requires power of 230v and 115v-AC at 60 cycles. The facility requires power of 115v-AC at 60 cycles for the reactor console and auxiliary equipment and 230v-AC at 60 cycles for all motors.

Since the system is fail safe, no auxiliary power is needed for the operation of post-shutdown safety systems. The loss of electrical power drops out the scram relays and de-energizes the magnetic clutches to trip the reactor by dropping the control rods under gravity completely into the core. Therefore, there is no need to consider offsite sources of emergency power.

8.3 Onsite Power System

The electrical supply to the reactor and console is supplied by the Regional Utility System of Alachua County. This offsite power is supplied onsite to operate the various non-nuclear reactor safety and monitoring instrumentation channels, as presented in Section 7.1.1.3. These channels are all dependent on the utility system A.C., power for proper operation. However, they will only be needed during operation to perform monitoring and scram functions. In a "loss of power" situation, the nuclear instrument channels and the fail-safe nature of the control rod system provides the proper trip and shutdown of the reactor.

Interruptions in power from the regional utilities system are quite common. Although such trips associated with loss of power are bothersome from a training or research stand-point, such a loss of power has no bearing upon the safe operation of the UFTR system. When power is lost, the reactor automatically trips. Since these interruptions in power are usually of short duration, there is no simple remedy for the loss of power problem. Therefore, secondary power systems are not considered in this report.

8.4 D.C. Power Systems

The radiological area monitors and stack monitors are powered by 24vDC power supplies backed up with a "floating" battery pack. In the event of loss of A.C. power, the battery packs will automatically power the monitors with the ability to maintain operation for at least 12 hours. This provides the system with an ability to monitor radiation activity in the reactor area at all times. Emergency lighting is located throughout the reactor building and the reactor cell. There is a two lamp emergency spotlight within the reactor cell to provide light in the event of a loss of power. The security system itself is also equipped with a battery power supply to maintain operation in the event of a loss of all electrical power.

8.5 Diesel Electrical Power

The UFTR will be connected to an A.C. Diesel Electric Generator located in the rear of the Reactor Building in the near future. The design of the switching system and the necessary purchase orders have been made and the work is expected to be completed by April 1, 1981. The Diesel Generator will provide backup electrical power for all reactor systems, including the radiation monitoring and physical protection systems, as well as emergency lighting.

No credit is taken for the back-up electrical Diesel Generator for safety analysis considerations.

For additional information on the Diesel Generator refer to Chapter 9, Section 9.5.4.

9. AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

Unirradiated reactor fuel is normally stored in a 5-drawer, fire-resistant Diebold Safe equipped with a combination lock. Supports are provided to space the plates in such a manner that no more than 56 plates can be placed in a drawer. The bottom of each drawer is lined with cadmium. The fuel storage safe, which is locked at all times except during transfer of fuel or inventory is located in the reactor cell. An authorized person is present at all times when the reactor cell (which comprises the reactor room and the control room) is unlocked. The reactor cell is protected by a security system which alarms at the University of Florida campus police headquarters.

Loading and unloading of the fuel into and out of the reactor will only be performed by qualified reactor operators and staff, and under the supervision of the reactor supervisor as specified in the UFTR SOP C.1 and C.2.

9.1.2 Spent Fuel Storage

Irradiated fuel is removed from the reactor in a lead transfer cask using the crane and special handling tools (Section 9.1.2.1); a continuous radiation survey is made while the fuel is being transferred. Irradiated fuel assemblies or plates are stored in the spent fuel storage area located in the concrete floor at the northwest corner of the reactor cell as shown in Figure 3-2. This storage area is readily accessible to the crane and contains 27 steel-lined storage pits, each of which is 4" in diameter x 4 ft. deep. These storage pits are arranged so that k_{eff} will be less than 0.8 under optimum conditions of reflection and moderation. Padlocked shield plugs are provided for these storage pits and are keyed to the University of Florida Proprietary Keyway, Sargent Grand Master Series. The key is kept in a safe, available to the Reactor Administration and under established conditions can be used by qualified reactor operators. Therefore, all reactor fuel which is not in the reactor will be locked either in the fuel safe or in the fuel storage pits, or in active transfer between these places.

Fuel plates are replaced when necessary. The irradiated fuel can be shipped to a fuel reprocessing plant after sufficient cooling.

9.1.3 Bridge Crane

A 30-ton bridge crane is provided for handling shield blocks, lead casks, and other heavy equipment. The crane travel allows coverage of the entire area of the reactor cell as shown in Figure 3-2. Maximum clearance of 11 ft., 9 in. can be obtained between the top of the reactor, which extends 11 ft., 10-1/2 in. above the floor, and the crane hook. The clearance is reduced to 8 ft., 9 in. over the water tank which extends 3 ft. above the top of the reactor. This clearance is adequate for use of the lead transfer cask to remove irradiated fuel elements from the reactor and also for the installation of any experimental equipment which might be desired over the internal thermal column. A balcony over the control room serves as a maintenance area for the crane and also as a shield to prevent damage to the control room from the crane hook or heavy objects moved with the crane. (See Figure 3-3). (1).

9.1.4 Fuel Handling Systems

9.1.4.1 General Precautions. Whenever fuel is loaded into or removed from the reactor, the following requirements shall be met:

1. All fuel transfer operations shall be supervised by the reactor supervisor or his duly authorized representative, who shall hold a Senior Reactor Operator License.

2. All the required logs, diagrams, records, and forms shall be maintained as specified by the UFTR SOP-C-1 and SOP C-2.

3. Adherence to the UFTR Technical Specifications criticality safeguards criteria shall be enforced at all times.

4. Minimum personnel requirements shall be met at all times during fuel movement-related operations as specified in SOP-C-1.

5. Radiation Control Personnel shall be present to perform periodic checks to assure the operability of the survey instruments, take swipe surveys, take air samples and perform radiation field surveys. Records of the above checking activities shall be maintained and adherence to limits set forth in 10 CFR 20 shall be observed as set forth in the UFTR SOP's. (3)

9.1.4.2 Fuel Loading Initial Conditions. The following initial conditions shall be observed and assured prior to the beginning of the fuel loading process:

1. The reactor will be operational with top shield blocks removed.

2. All the requirements specified in the UFTR Technical Specifications must be satisfied.

3. The pre-nuclear testing program, as defined in SOP-C.2, must be satisfactorily completed.

4. Neutron source(s) must be installed and a minimum count rate established on the start-up channel.

5. Visual inspection and clearing of any fuel assembly or dummy assembly must be performed before insertion into the core.

6. All operations must be previously approved by the Reactor Supervisor.

7. Only licensed reactor operators shall insert fuel into the reactor.

8. Minimum personnel requirements must be met as specified in SOP-C.2.

9.1.4.3 Fuel Loading to Critical and Operating Reactivity. The following conditions apply for fuel loading to critical and operating reactivity:

1. All fuel loading (including dummy assemblies) will be performed with the water out of the core and all the control blades fully inserted.

2. All counts for subcritical multiplication will be taken with the primary water up, and as specified in SOP-C.2.

3. At no time will the reactor core be loaded with a reactivity in excess of 2.3% $\Delta k/k$.

4. Fuel loading increments must be carefully controlled. Regulations and limitations for both an unfueled and partially fueled UFTR core must be followed as outlined in SOP-C.2. These regulations and limitations are designed to assure that the amount of fuel loaded in any one step will not exceed the critical mass for water-up and two safety blades fully withdrawn.

5. All fuel loading for Step 4 shall be made from the most reactive to the least reactive location as a further safety precaution.

6. Full or partial dummy assemblies may be used during fuel loading to occupy empty positions to support assemblies.

7. Full or partial dummy assemblies must be used to fill any vacant position in the core after fuel loading is completed. (3)

9.1.4.4 Fuel Removal and Storage. Before attempting fuel removal operations, two preliminary precautionary measures must be taken. First, precautions must be taken to limit the vertical movement of the fuel. The necessary safety line and its length will be determined using a dummy fuel element. Second, all necessary monitoring and alarm systems shall be checked for operability.

The following requirements must be met before actual operations for removal of fuel from the core are undertaken:

1. The shield tank must be prepared to receive fuel for inspection as specified in SOP-C.1.

2. Fuel pits must be prepared as necessary to receive the fuel.

3. All neutron and radiation monitoring systems must be in operation.

4. The Reactor Vent System must be in operation.

5. The neutron source must be installed in the reactor to assure the detection of fission events by the instrumentation.

6. Reactor shielding must be unstacked as necessary to permit core area accessibility.

7. Reactor primary coolant must be up and the console key must be removed from the console.

8. A reactor operator must be at the console.

9. Removal of shield plug and wedging pin from the fuel box shall be performed under direct supervision of the person in charge and radiation control personnel must be present for surveying at time of shield plug removal.

When removing fuel from the fuel pit, the shield tank shall be prepared if inspection of fuel is required. In addition, other fuel pits shall be prepared if change of fuel locations within the fuel pits is the only required operation.

Detailed descriptions of the procedural steps to be followed during transfer of fuel to and from the fuel transfer cask and for fuel inspection are contained in SOP-C.1 for the UFTR facility.

9.2 Water Systems

9.2.1 Station Service Water System

The cooling of the UFTR is accomplished by two systems: the primary and secondary cooling systems.

The primary system uses demineralized water from the coolant storage tank to transfer the heat from the reactor to the heat exchanger. The storage tank has a capacity of 200 gallons of water which is approximately six times the capacity of the reactor. Whenever necessary, make-up water is supplied from the city water line through demineralizers contained in the UFTR Water Coolant Make-up System sketched in Figure 9-1.

The secondary cooling system has two sources of water as indicated in the diagram of Figure 9-3:

1. The deep rock well used during normal operation.

2. The city water supply used for back-up operation or for training purposes.

The deep well is 238 ft. deep with a casing diameter of 3", the static water level is approximately 87 ft. below grade. The well water flows through a well strainer, which has a stainless steel mesh of approximately 1/16", and then into the shell side of the heat exchanger from where it goes to the storm sewer. (5)

9.2.2 Shield Water Tank

The shield water tank is a 5 ft. x 5 ft. x 14 ft. high water tank placed against the west face of the reactor, opposite the thermal column (see Figure 1-8). Shield water tank components include:

1. Water level indicator,

2. Pump,

3. Ceramic filter,

4. Flow water indicator,

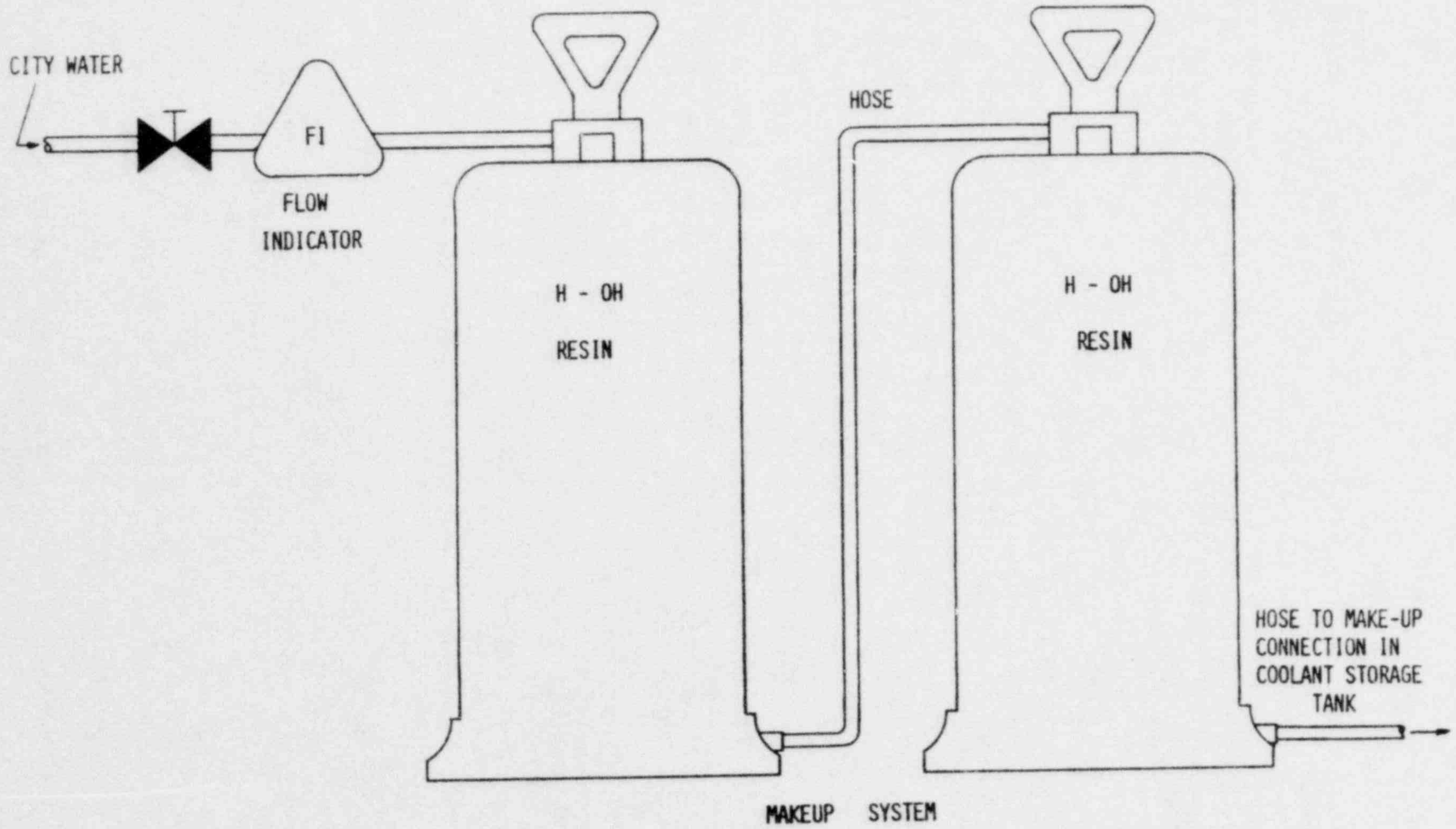


Figure 9-1. Diagram of UFTR Primary Water Makeup System.

5. Demineralizer,

6. Sampling valve.

This test tank is primarily used for experimental purposes. If necessary, the tank can be drained and lifted out of the way with the bridge crane. All water drained from this tank will go directly to the reactor sink and the holdup tanks where it will be monitored. It will then be released to the University of Florida Sanitary Sewage System if, as expected, the activity level is below those established by the Radiation Control Office. If activity levels exceed those established by the Radiation Control Office, then the water will be held up in the reactor sink until activity levels have decayed sufficiently to allow release.

9.2.3 Demineralized Water Makeup System

Demineralized water is used as makeup to the primary coolant system. The makeup system consists of two demineralizers in series that are filled with Amberlite, nuclear grade resin, as is the demineralizer in the primary loop. The unit has a hose with a connection that can be made to the primary tank when water is needed. As indicated, the schematic of the makeup system is shown in Figure 9-1. The makeup connection for the primary system is found on the side of the coolant storage tank, and is located on the top of what is called the "ice chute."

9.2.4 Purification System

The purification loop is provided with a separate pump in order to maintain a continuous purification flow. The purification pump is interlocked with the primary coolant pump in a manner which shuts off the purification pump when the primary coolant pump is running.

The arrangement of the purification loop provides the system with continuous monitoring of the resistivity of the primary water and the functioning of the Amberlite nuclear-type resin (H-OH; H-control) in the purification system. The in-line, wall-mounted resistivity bridge is set up to accept two conductivity cell signals--one before the demineralizer and the other after the ceramic filter. A schematic diagram of the primary loop purification system is presented in Figure 9-2, showing the feed and bleed nature of the system and its various components. (5)

9.2.5 Potable and Sanitary Water Systems

The UFTR Building does have potable and sanitary water system connections. Tap water and a utility sink are located in the northwest corner of the reactor cell. A "back flow preventer," as required by the National Plumbing Code, is installed in the city water line ahead of any industrial type use of this water.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

An air compressor and associated components is located in the Air Conditioner Equipment Room on the north side of the Reactor Building. This system

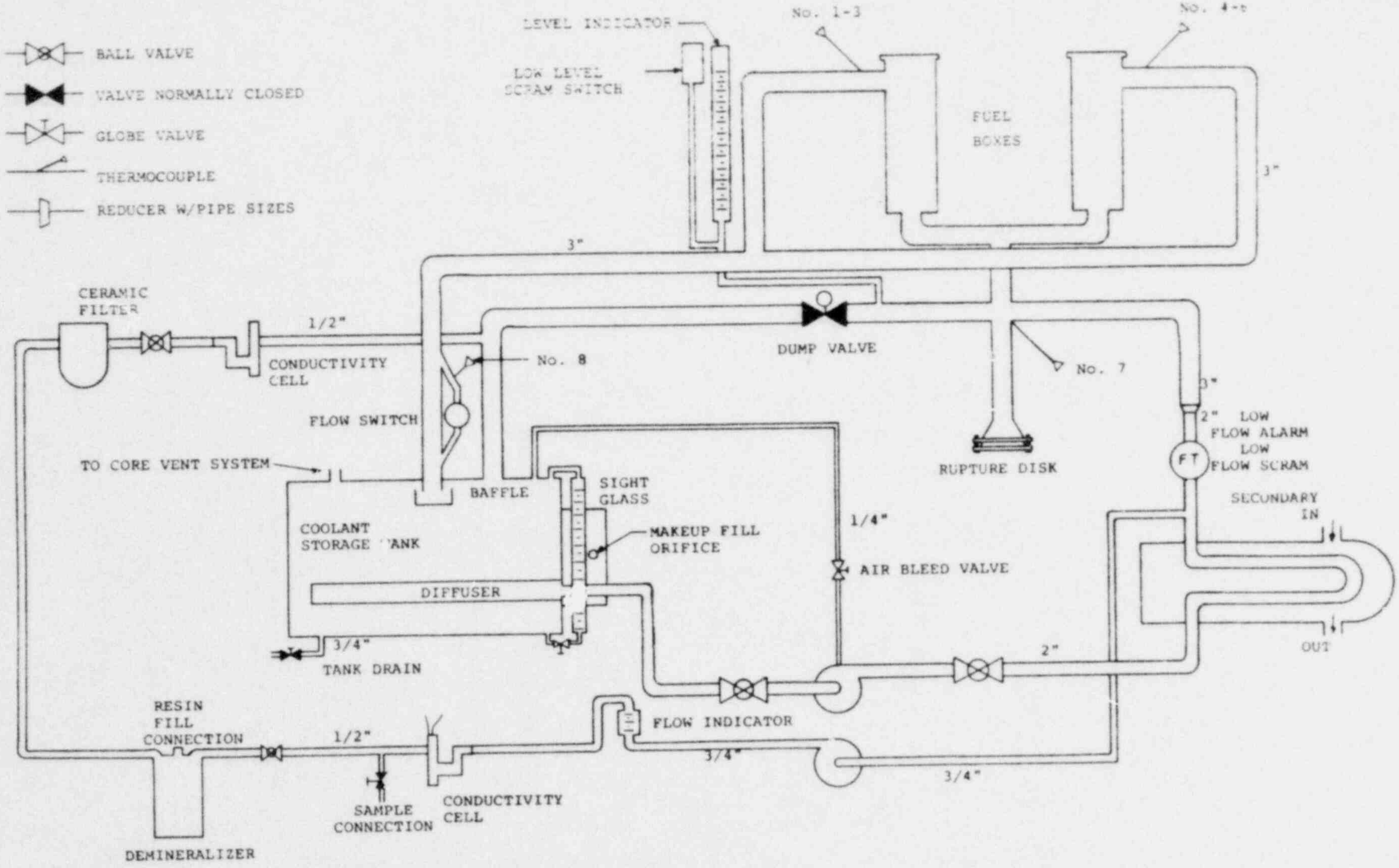


Figure 9-2. Schematic of UFTR Primary Coolant Loop and Purification Systems.

supplies compressed air for the laboratories in the Reactor Building, and for operation of the thermostats and valves of the air conditioning system.

9.3.2 Process Sampling System

The process sampling system for the UFTR consists of several sample valves found in the primary and secondary coolant loops, and in the purification system as labeled in Figure 9-2 showing the Primary Coolant Loop and Purification System and in Figure 9-3 showing the Secondary Loop Cooling System. Process sampling is done routinely on a weekly basis as part of the Weekly Pre-operational Check (SOP-A.1).

For the primary system, water samples are taken from the sample valve located in the equipment pit. Two samples are required. One sample is used in the reactor cell to check the water resistivity; the second sample is taken to Radiation Control for analysis. For the secondary system, two water samples are taken to check for primary to secondary coolant leaks. There is one sample flow valve in the heat exchanger discharge line which continuously bleeds a small sample flow into the hold-up sample tank. A second sample valve, which is normally closed, is used for collecting a sample directly from the heat exchanger, as shown in Figure 9-2. Water samples to check the shield water resistivity are taken from the sample valve located in the shield tank system.

9.3.3 Equipment and Floor Drainage System

The reactor building floor drainage system is designed so that all liquid effluents will go directly to the hold-up tanks. There are no drains leading directly to the hold-up tanks; therefore, all the water must be pumped or drained to the reactor sink which drains directly into the hold-up tanks.

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

9.4.1 Control Room Area Ventilation System

The reactor cell is completely air conditioned with a recirculating type system designed to provide an atmosphere suitable for reliable operation of electronic instruments and for human comfort. The air-conditioning unit has a design capacity of 1500 c.f.m. (approximately 2 changes per hour) with a total air delivery of 6050 c.f.m. at 75°F, dry bulk temperature, and 50 percent relative humidity, summer and winter. All inlet and circulated air is filtered through a 2 in. thick, dry, spun glass, cleanable-type roughing filter capable of removing particles of 5 microns or larger in size with an efficiency of 85 percent or better. The inlet air duct is provided with a motor-operated damper to close the duct whenever the unit fan is not operating.

The room exhaust air is used to ventilate the reactor structure. The vent flow from the reactor cavity is adjusted within limits conducive to minimization of releases of Argon-41 to the environment and exposures to personnel within the reactor cell. The vent flow is controlled by the operation of a small blower fan and an electrically actuated damper. This air is passed through a roughing filter and an absolute filter to an outside stack where it is diluted with approximately 12,000 c.f.m. of outside air. It is then discharged through the stack extending from the roof of the building where a further 200 to 1 atmospheric dilution is effected.

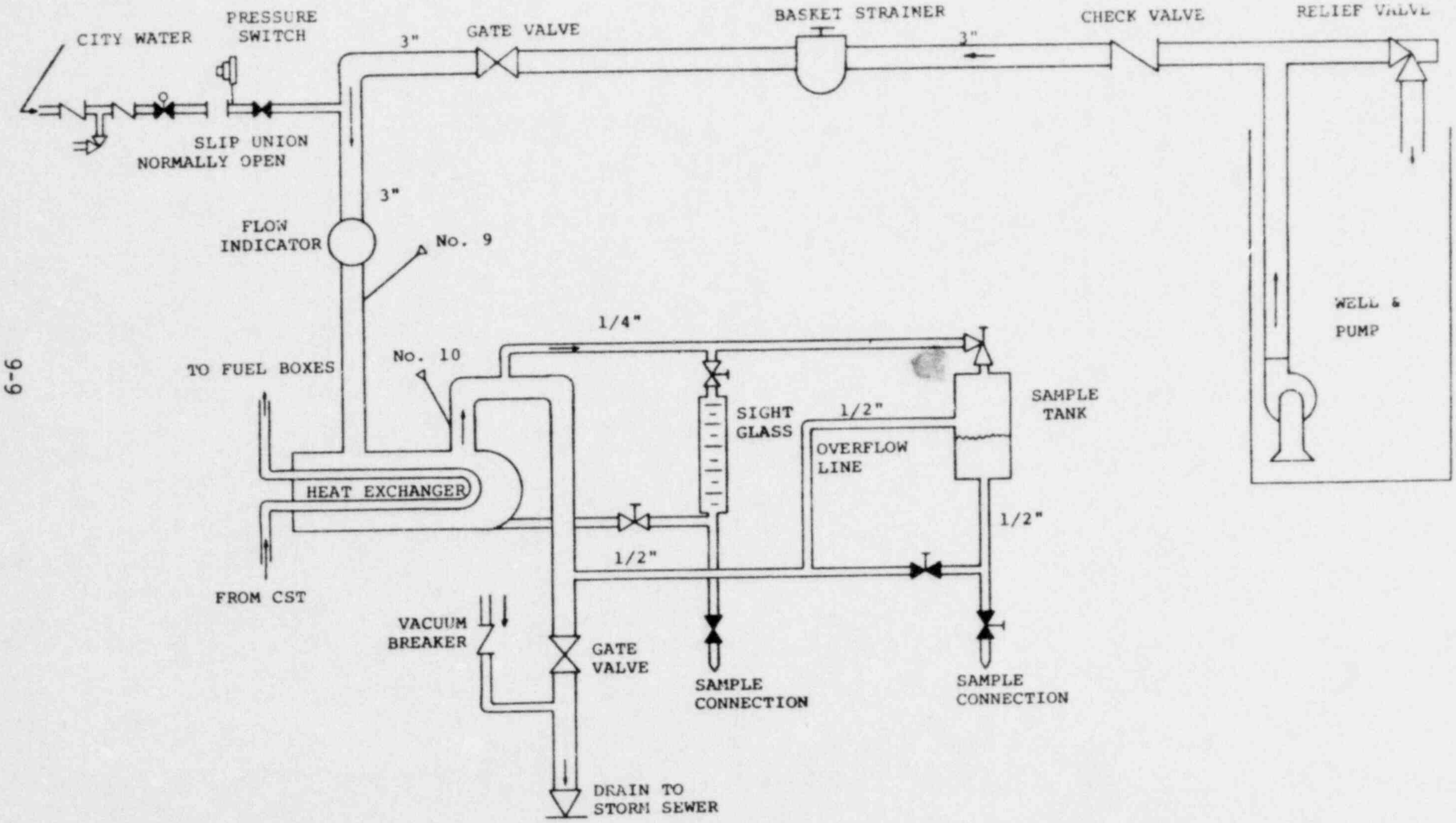


Figure 9-3. UFTR Secondary Water Cooling System.

9.4.2 Core Vent System

As indicated in Section 9.4.1, in order to prevent radioactive gases and particulate matter formed in the reactor from escaping into the reactor room, the air surrounding the reactor core structure is withdrawn by the core vent system and then through a rough and an absolute filter. The air is then discharged through the stack where it is diluted with about 12,000 c.f.m. of outside air before it is released to the atmosphere.

Vacuum breaker vent lines (1" diameter) connect the tops of the fuel boxes to the coolant storage tank to provide an air-return path allowing rapid dumping of the water from the boxes. The coolant storage tank vent connection to the reactor ventilation system is shown in the diagram of Figure 9-4 giving a vertical section view of the physical arrangement of the UFTR Core Vent System. The vent lines are positioned between the graphite blocks that surround the fuel boxes and the concrete shield tank. A schematic flow diagram of the core cooling and vent system is presented in Figure 9-5.

On-line measurement of the vent flow rate is accomplished by a pitot tube in the outlet line of the core vent. A differential pressure, proportional to the square of the flow rate, is displayed on inclined manometers on the north wall of the reactor. The differential pressure across the rough filter is indicated by another inclined manometer, and the differential pressure across the absolute filter is indicated by a "Magnehelic" gauge. These three instruments display differential pressure in inches of water head.

Gamma activity of the gaseous effluent release is monitored by a GM detector located on the downstream side of the absolute filter after the pitot tube (see Figure 9-4) at the base of the stacks before dilution occurs. An audible alarm will be actuated in the control room, in the event the vent flow activity reaches a preset level. The data from this monitor is continuously recorded. In the exhaust duct there is a motor opened, spring-closed damper valve which automatically closes whenever the fan is not operating.

The Reactor Vent System prevents diffusion of radioactive gases or particulate matter into the reactor room during reactor operation. Loss of electrical power to either the reactor vent damper or the dilution fan motor will result in a reactor trip without dumping primary water. The vent damper is electrically interlocked with the dilution fan motor control circuit so that the damper control cannot be opened unless the dilution fan is energized. This interlock prevents the discharge of undiluted air effluent via the stack.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

Since none of the materials of construction of the reactor are inflammable, and since the reactor building is fireproof construction and will not be used for storage of quantities of inflammable materials, a fire of any consequence is considered very unlikely.

Conventional fire equipment is located in the reactor cell and throughout the reactor building. Five CO_2 extinguishers are available in the reactor room itself, and one more is located in the control room at the control

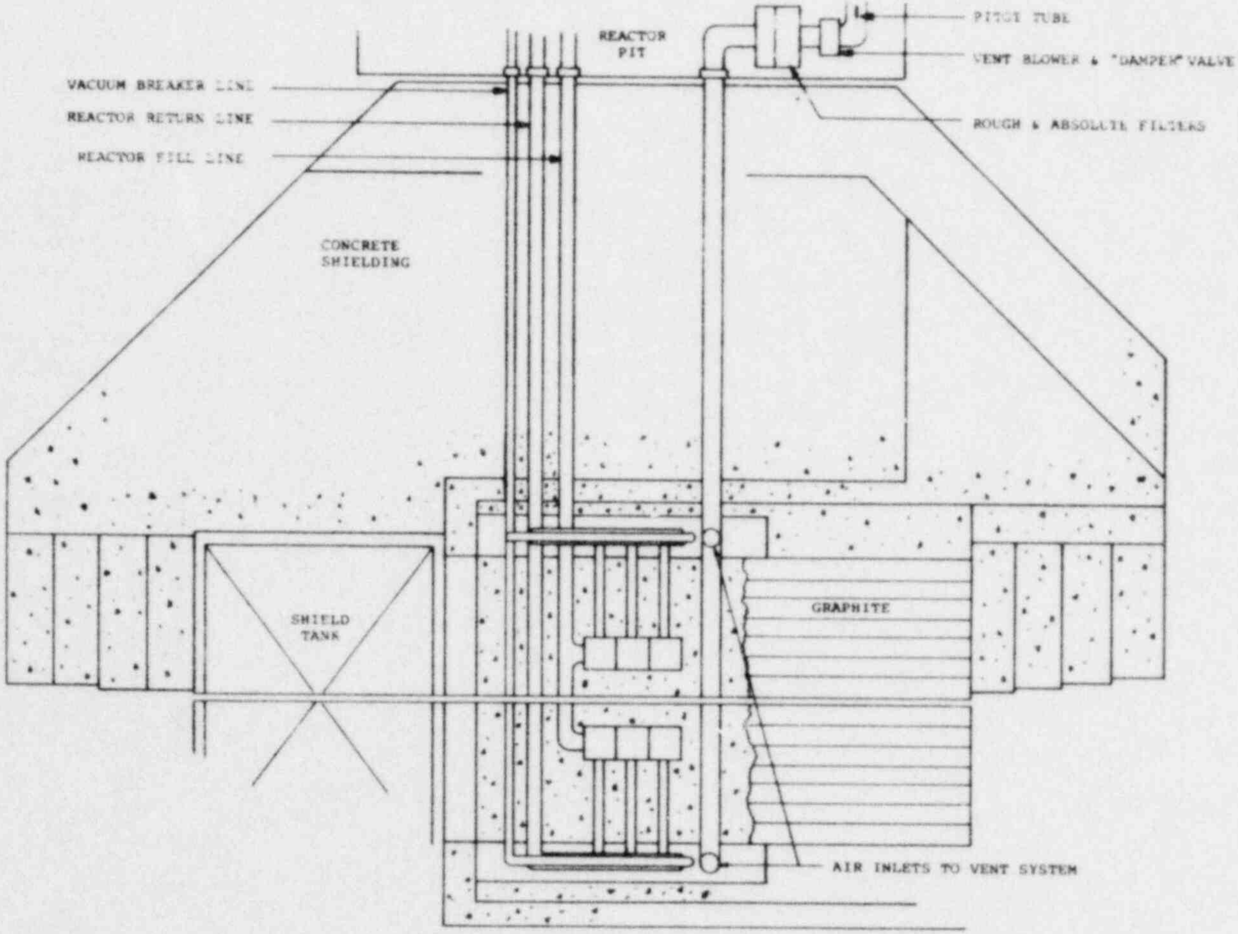


Figure 9-4. Vertical View Schematic Flow Diagram Showing Physical Arrangement of UFTR Core Vent System.

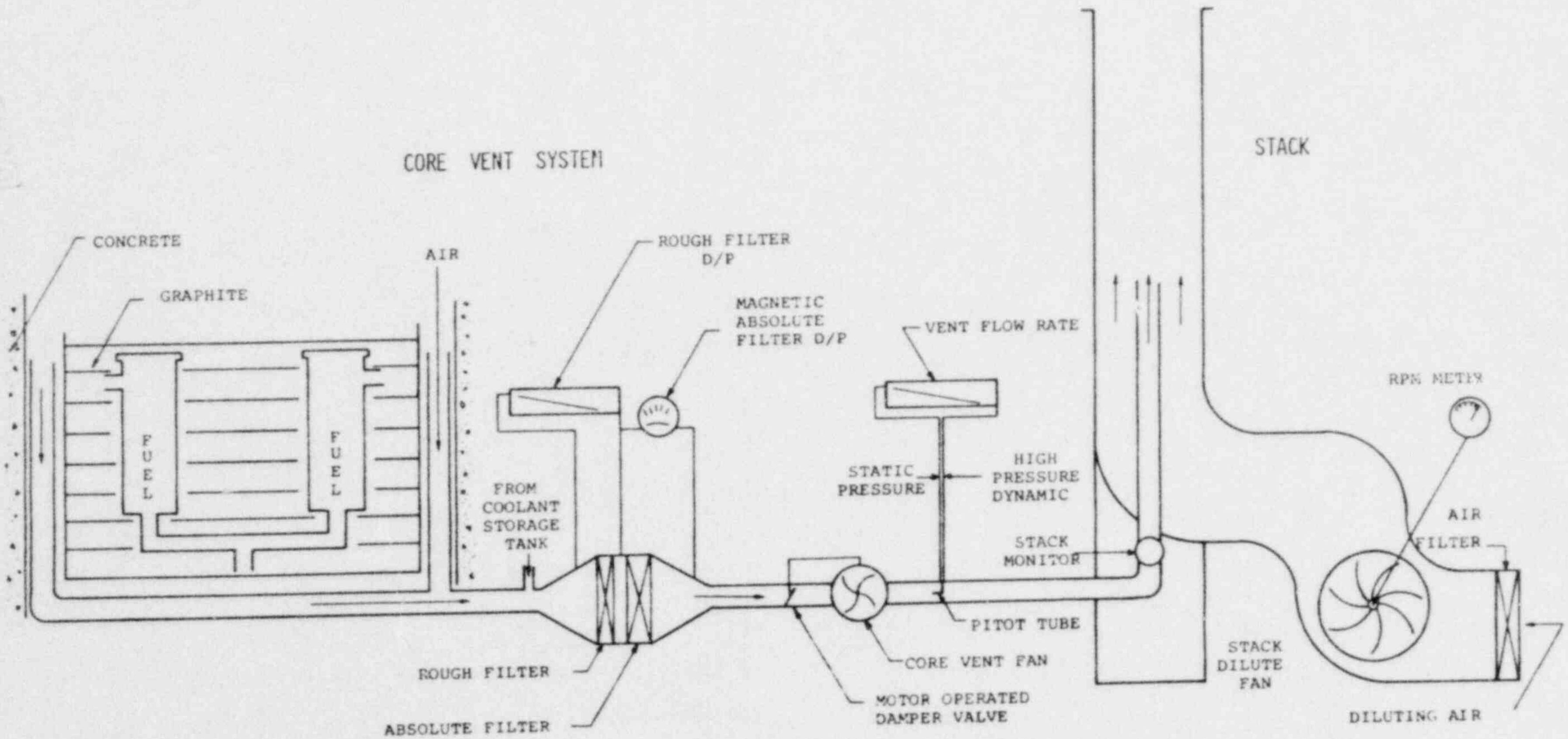


Figure 9-5. Side-View Schematic Diagram of UFTR Core Vent System.

console. A fire hose and fire extinguisher are also located outside the control room in the ground floor foyer area referred to as the Limited Access Area in Chapter 3 of this report.

An automatic fire alarm system monitors the reactor cell continuously and the reactor building. The system in use is a Simplex, Type 4207, completely-supervised system with Emergency Battery Back-up. The following equipment is installed:

1. Two (2) Ionization Detectors

2. Two (2) Thermal (Heat) Detectors

3. Seven (7) Pull Stations

4. Six (6) Horns

This system alarms at the Campus Police Station. Operation of this system will turn on the emergency light in the reactor room (for illumination).

9.5.2 Communications Systems

A full-service telephone is installed within easy reach of the reactor operator at the console. This provides direct communication within the building, on and off-campus including: The Reactor Supervisor, Radiation Control Office, University of Florida Police Department, Gainesville Fire Department and Senior Reactor Operator.

An intercom system is set up providing direct communication from reactor console to the Reactor Supervisor, Senior Reactor Operator (not present in effect) and the Health Physics Office.

In case of a power failure, the telephone will be available for communication within the building as well as on and off-campus.

9.5.3 Lighting System

The reactor building is provided with overhead fluorescent lighting. Additional supplementary lighting is possible via 115v wall outlets.

In case of a power failure, emergency lighting is provided automatically throughout the building by the emergency diesel generator located outside the reactor building.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

The diesel generator is a Turbo-Charge D-6 Caterpillar type generator and is available for emergency conditions in case of a power failure. The system is designed to come on line automatically within 10 seconds after the power failure, operating 10 to 11 minutes after power recovery, as a back-up power supply in case of repeated failure within this short period of time. The automatic starting system provides for three start-up events within a 90 second period, after which it goes into a manual stand-by condition with the option of a manual start-up or a reset mechanism for start-up.

Fuel oil storage provisions consist of an underground tank with a capacity of approximately 2000 gallons. Fuel oil transfer is accomplished by an electrical motor system with a manually operated hand-pump as a secondary backup. Cooling of the system is provided by a radiator assembly. Inspection of the Diesel Generator System is carried out on a routine weekly basis by the Plants and Grounds Division of the University of Florida; preventive maintenance is provided by the Ring Power Company - Ocala Division.

10. STEAM AND POWER CONVERSION SYSTEM

The 100 Kw University of Florida Training Reactor operates at low power levels, low temperatures, and near ambient pressure levels; by design, the UFTR produces no steam and no electrical power. There is no working fluid cycle. Therefore, since no steam and power conversion system is needed, this chapter is not considered necessary.

11. RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

The UFTR is designed to minimize leakage of gaseous and particulate radioactive materials from the core area into the reactor room. A core vent and exhaust system draws air from the reactor room through the various openings and cracks and crevices of the reactor structure, around the graphite reflector, and into the two vent intake lines. This air is then passed through a rough and an absolute filter, through an automatic damper valve, past a pitot tube for flow sensing, and into a plenum chamber at the base of the stack where it is monitored for radioactivity by a GM detector. The activity level of the air is indicated and recorded in the control room, with adjustable audible and visual alarm level.

After leaving the chamber, the core vent air is diluted with about 12,000 c.f.m. of outside air which enters the stack above the plenum chamber. As the effluent plume leaves the stack, a further atmospheric dilution factor of 200 to 1 is applied for the purpose of determining radioactivity concentrations in the environment.

This ventilation, filtering and dilution process assures a reduced likelihood of radioactive gases escaping into the reactor room and reduces the amount of particulate and the concentration of effluent. Because of the system design and characteristics of the reactor, there is no significant danger of fission product release under normal (and accident) conditions. Calculations discussed in Chapter 4 show the low power density provides a large safety margin before fuel or clad temperatures would produce radioactive releases.

The sources of radiation which are the basis for required radiation protection during operation are primarily the core neutron and gamma ray fluxes. Sources activated for experimental purposes in the UFTR experimental ports are also a concern but only after reactor shutdown. Previous radiation exposure measurements indicate that the radiation hazard in the reactor due to both thermal and fast neutrons is negligible; therefore, the main concern is the gamma exposure. (23)

The only normal isotope of concern is the Ar-41 produced in the UFTR as a result of neutron activation of the Argon-40 in the air drawn in through the crevices in the concrete and the graphite reflector. (This topic is addressed in Section 11.3.) Since Argon-41 production is proportional to thermal power produced by the UFTR, a historical summary of UFTR energy generation is presented in Table 11-1. This summary contains total energy generated (Kw-HR) and hours at full power for all reporting years after the UFTR October, 1971.

The natural atmospheric argon is responsible for virtually all of the neutron-induced radioactivity released to the stack. (30) The other gaseous components of air are either too rare, have small activation cross sections, or produce activated products having half-lives too short to be of significance. The combination of argon properties shown in Table 11-2 accounts for the fact that Argon-41 provides essentially all of the radioactivity to be found in the reactor ventilation air leaving via the building stack from the Core Vent System. (31)

TABLE 11-1

HISTORY OF UFTR ENERGY GENERATION SINCE REACHING THE LICENSED 100 KWth POWER LEVEL FOLLOWING SYSTEM MODIFICATIONS IN 1970*

YEAR	KW-HR GENERATED	HOURS AT FULL POWER
Sept. 1, 1971-Aug. 31, 1972	29,873.67	Not Abstracted
Sept. 1, 1972-Aug. 31, 1973	23,039.54	Not Abstracted
Sept. 1, 1973-Aug. 31, 1974	8,904.37	78.8
Sept. 1, 1974-Aug. 31, 1975	43,835.15	425.18
Sept. 1, 1975-Aug. 31, 1976	12,388.62	116.74
Sept. 1, 1976-Aug. 31, 1977	25,388.14	243.67
Sept. 1, 1977-Aug. 31, 1978	26,375.80	248.02
Sept. 1, 1978-Aug. 31, 1979	9,079.30	84.85
Sept. 1, 1979-Aug. 31, 1980	9,800.14	90.97

*The license amendment to upgrade UFTR rated power to 100 KWth was granted in 1964. After a number of years operation, system repairs and modifications were made in 1970. Following these modifications, the UFTR first reached 100 KWth in October, 1971.

TABLE 11-2

SELECTED PROPERTIES OF ARGON (32)(33)

Atmospheric Abundance (By Volume)	0.934%
Isotopic Abundance of Argon-40	99.6%
Argon-40 Activation Cross Section (n,γ)	0.53 barns
Activation Product	Argon-41
Product (Argon-41) Half-Life	110 minutes
Radiation Breakdown	β_1 - 2.49 MeV (.8%) β_2 - 1.20 MeV (99%) γ_3 - 1.29 MeV (99%)

Two experimental determinations provide sound evidence to support the contention that the measured activity is due to Argon-41 decay. (31) First, the photopeak energy of the stack air samples was determined to be 1.29 MeV using an energy calibrated gamma scintillation spectrometer which corresponds to the gamma energy associated with Ar-41 decay shown in Table 11-2. Second, a half-life determination was made and the experimentally determined decay curve presented in Figure 11-1 verifies that the sample activities decay with a half-life of about 110 minutes. Since these values are characteristic of Argon-41 and no other radionuclides are detectable, the radioactive contribution o' Argon-41 is demonstrated.

Lochamy has performed extensive Ar-41 sampling studies for the UFTR. (31) Lochamy's samples were drawn at the base of the stack prior to dilution. Each sample was passed through a drying apparatus made of silica gel and spun glass prior to collection in an evacuated flask. The analysis was performed in a low-background counting room by gamma scintillation spectrometry. The results of the stack activity experimental measurements taken at different power levels up to 90 KWth are presented in Figure 11-2 where each value is the average of several measurements taken at the different equilibrium UFTR power levels. Analysis of the dependence of measured stack activity on UFTR power level resulted in the expected linear fit shown in Figure 11-2 which has a slope of 6.7 pCi/cm^3-Kw.

The following is a summation of Lochamy's data:

Vent Flow rate:	376 cfm
Exhaust Flow Rate:	13,402 cfm
Average Argon-41 Stack Activity per Unit Power Level:	6.7 pCi/cm^3-KW

At 100 KWth, a stack sample activity value of 670 pCi/cm^3 is extrapolated from the data presented in Figure 11-2. This value, when the dilution factors of 35.6:1 for the stack and 200:1 atmospheric dilution authorized by the Nuclear Regulatory Commission are used, yields 9.4×10^{-8} $\mu C/ml$ which is a factor of 2.35 above the MPC for releases to an uncontrolled area at any instant of time at the 100 KWth power level.

In June and July of 1977, the UFTR staff conducted another survey. Using a 92 cm^3 Navy type gas collection chamber, samples were drawn at different points at the horizontal plane at the stack discharge at the 100 KWth equilibrium power level. The samples were counted using an Ortec Ge-Li detector whose signals were processed by a Tracor Northern TN-11 computer based multichannel analyzer using a spectrum analysis system.

The activity results at the stack discharge at the 100 KWth power level are as follows:

Average Argon-41 Sample Activity:	2.48×10^{-5} $\mu Ci/ml$
Activity with Authorized Atmospheric Dilution (200/1):	1.24×10^{-7} $\mu Ci/ml$
MPC:	4.0×10^{-8} $\mu Ci/ml$

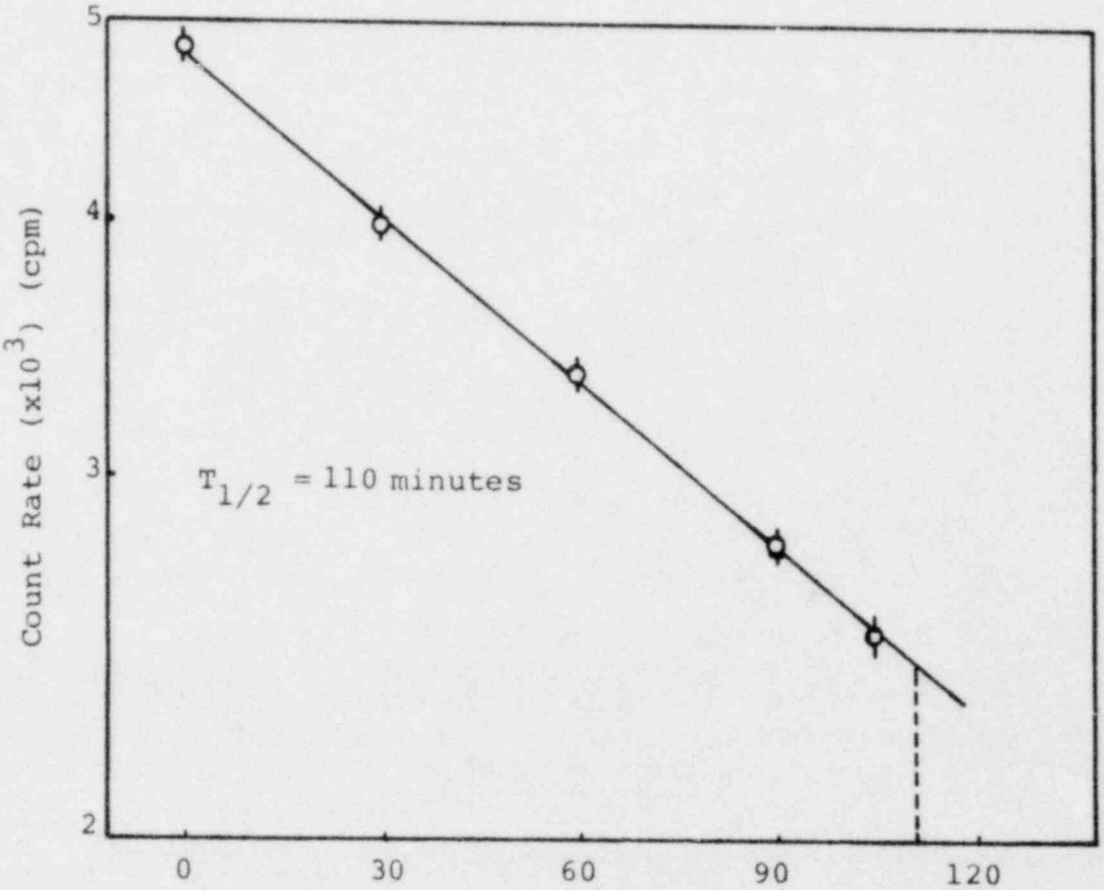


Figure 11-1. Data for Half-Life Determination of UFTR Stack Sample. (31)

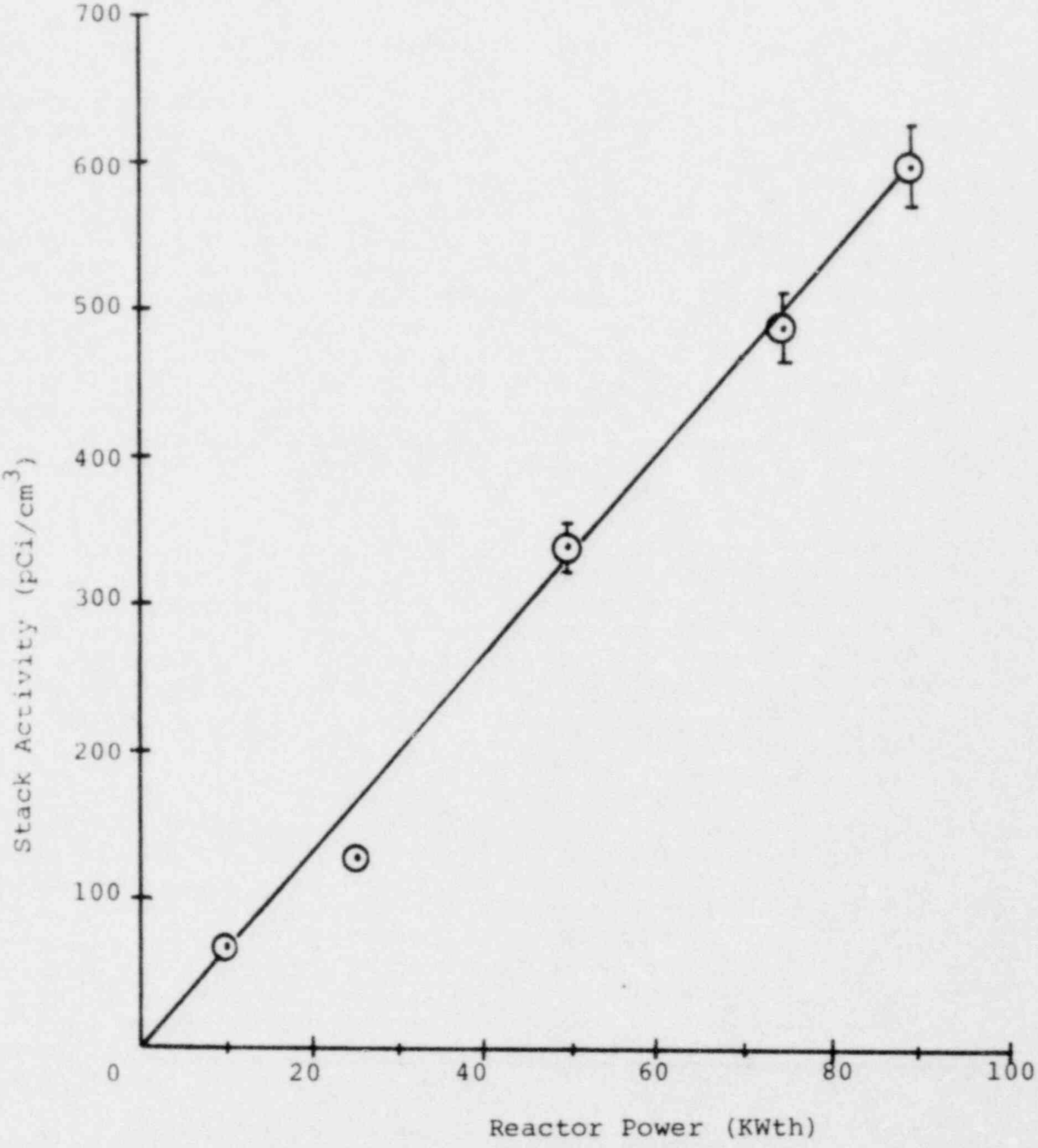


Figure 11-2. Experimental Determination of Argon-41 Stack Concentration Variation With UFTR Operating Power. (31)

Because of this analysis showing the diluted Ar-41 release concentrations at 100 KWth to be a factor of 3.1 above the MPC for this activity, certain restrictions have been placed upon UFTR operations as discussed in Section 11.3.

11.2 Liquid Waste Management

All liquid waste from the UFTR is pumped or drained into the reactor sink and subsequently stored in the Waste Water Holdup Tanks, sized to hold 80,000 liters of liquid. Periodic samples of the collected liquid waste are taken by the Radiation Control Office and assayed to determine the total activity level present. If, as expected, activity levels are within the acceptable levels for release, then the contents of the tank are released into the University of Florida Sanitary Sewage System where the released water is further diluted by an average flow of approximately one million gallons per day. The liquid wastes do not present any problems during operation of the UFTR. Acceptable activity levels for release from the waste water holdup tank have been established by the Radiation Control Office. Based on an average daily flow of 1,000,000 gallons of sanitary sewage, not more than 1/1000 of the maximum amount of radioisotopes specified by 10 CFR 20, Appendix B, shall be released to the University of Florida sanitary sewage system in any one day.

Any liquid waste which must be shipped from the UFTR facility will be placed in appropriate containers suitable for permanent storage and will be properly labeled according to Radiation Control Technique #3, "Instructions for Disposal of Radioactive Waste." As necessary, the containers will be stored on-site and occasionally in the NRC-approved storage area for low-level waste, until the activity has decayed sufficiently to permit safe shipment and until sufficient quantity is accumulated to warrant pickup and ultimate disposal by an NRC-approved disposal agency.

11.3 Gaseous Waste Management

Precautions are taken to insure that no radioactivity is above established tolerance levels when released to the surroundings. Radioactive Argon-41 and Nitrogen-16 are produced in the UFTR. Argon-41 is produced as a result of neutron activation of air containing 1% Argon-40 drawn through crevices in the concrete and graphite shielding while Nitrogen-16 is produced from oxygen-15 activation (O-16 in water). Leakage of these activated gases into the reactor cell is prevented by drawing air from the cell, through the reactor and out the exhaust stack by the core vent described in Chapter 9. Thus, the negative pressure maintained in the shields assures air flow to the Core Vent System from the reactor cell. Air from the core is drawn by the core vent and exhaust system passed through a rough and absolute filter, and discharged through the 30 foot stack where it is diluted with approximately 12,000 c.f.m. of outside air before it is released to the atmosphere. Whenever the reactor vent system is operating, air drawn from this system is continuously monitored for gross concentration of radioactive gases and recorded in the control room. Upon failure of the air monitoring system, the reactor vent system is isolated. If the activity level reaches 120% of that found to be normal, the monitor will actuate a warning light and an audible alarm in the reactor control room. As part of the reactor safety system, any loss of power to the reactor vent or dilution system will cause a reactor trip.

As indicated in Section 11.1, studies conducted by the UFTR staff in June and July of 1977 showed an average Ar-41 concentration at the stack discharge of 2.48×10^{-5} μCi/ml which with the authorized 200/1 atmospheric dilution became 1.24×10^{-7} μCi/ml which is 3.1 times the MPC governmental limit. Since the MPC as recommended by the Florida Division of Health for 168 hour non-occupational exposure to Ar-41 is 4.0×10^{-8} μCi/m^3, the UFTR is limited to 23,535 Kw-hrs energy generation per month in order to comply with State of Florida regulations. (29) This restriction will be enforced until changes are made in the Core Vent System and new Ar-41 release data is obtained and analyzed to show no need of the restriction. As indicated by the UFTR energy generation history in Table 11-1, this restriction is not expected to present any problems. A continuous film badge monitoring system is maintained by the UFTR in areas adjacent to the UFTR complex. Since exposures typically indicate less than 10 millirem per month (approximately background), radioactivity releases from the UFTR facility are not considered to be a problem.

11.4 Solid Waste Management System

Solid wastes are generated from irradiated samples, contaminated gloves, filter traps on the waste water holdup tank and other similar sources. All solid wastes are collected in accordance with Radiation Control Technique #3, "Instructions for Disposal of Radioactive Waste." These wastes are expected to be low-level and less radioactive than wastes already generated on campus by research efforts in other disciplines such as the biological sciences in the Biology Department or the medical sciences at Shands Teaching Hospital. Normally, only solids will be shipped from the UFTR site; therefore, any liquid waste which must be shipped will be placed in vessels suitable for permanent storage and sufficient absorbent will be added to take up the entire volume of the liquid--effectively reducing the waste to solid form.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

There are two normal effluents channels connected with operation of the UFTR: radioactive effluents from the Waste Water Holdup System and gaseous effluents from the UFTR Core Venting System. Both effluents are monitored as released; the UFTR Stack Monitoring System is always in operation and is required for normal reactor operations. In addition to these two effluents, the secondary coolant discharge is monitored (through a sample tank) to assure that no primary-to-secondary coolant leaks exist.

11.5.1 Effluent Channel 1 - Waste Water Holdup Tanks

The first ordinary effluent channel for the UFTR consists of Water Holdup Tanks thorugh which liquid "waste" is periodically released when either of the two tanks contains a significant quantity of water. Usually one or the other of the two tanks is dumped approximately 10-12 times per year. The UFTR normally releases to the holding tanks approximately 1500 milliliters of primary coolant per week due to waste from primary sampling. For a typical reporting period of Sept. 1, 1977-Aug. 31, 1978, the average activity was 2.1×10^{-6} μCi/ml.

Only liquids meeting the requirements set by the Radiation Control Office based on acceptable activity levels are released into the University of Florida Sanitary Sewage System through Channel 1. After negative monitoring, these releases occur at irregularly spaced intervals about ten times per year.

Periodic samples are taken from the Waste Water Holdup Tanks; the samples are assayed for type and quantity of isotope present. If any of the activity in the holdup tanks is long-lived and above acceptable levels for discharge, the contents are drummed and stored in the NRC-approved storage area for low-level waste until the activity has decayed sufficiently to permit safe shipment and until sufficient quantity is accumulated to warrant pickup and ultimate disposal by an NRC approved disposal agency. Otherwise, when the samples demonstrate acceptable activity levels for release, the liquid in the tank in question is released.

A summary of the liquid waste release from the UFTR Waste Water Holdup Tanks is presented in Table 11-3 for all reporting periods (years) since the UFTR first reached full rated power in October, 1971, following relicensing and system modifications. As noted in Table 11-3, the liquid effluent discharged into the holding tanks comes from approximately twenty laboratories within the adjacent Nuclear Sciences Center as well as from the UFTR building. These sources account for the large volumes recorded in Table 11-3. The maximum activity in any release for each reporting period (year) is also recorded in Table 11-3 to demonstrate the low level of these releases. Following release and combination the usual million gallons per day sanitary sewage flow, the activity level is several orders of magnitude below the limits for activity release established by 10 CFR 20, Appendix B.

11.5.2 - Effluent Channel 2 - UFTR Building Stack

The second ordinary effluent channel consists of the stack leading from the Core Vent System depicted in Figures 9-4 and 9-5.

Because the air in the UFTR shield contains the isotope Argon-40 which undergoes neutron absorption to radioactive Argon-41, the Core Vent System assures that air is pumped one way from the reactor cell through the shield and then out the UFTR building stack.

Table 11-4 contains a summary of gaseous Argon-41 effluents released to the environment from the reactor building stack. This summary is presented for all reporting years after the UFTR was relicensed and reached its current rated 100 KWth power level following system modifications as abstracted from the UFTR Yearly Activity Reports. Data presented include yearly releases of Argon-41 (curies) as well as the maximum monthly recorded Argon-41 concentrations prior to atmospheric dilution (μCi/ml) (Column 3) and after 200 to 1 atmospheric dilution (μCi/ml) (Column 4). Since the maximum permissible concentration allowed by 10 CFR 20, Appendix B for release to an uncontrolled area is 4.0×10^{-8} μCi/ml, the NRC authorized atmospheric dilution ratio of 200 to 1 is shown to meet this requirements.

11.5.3 Monitoring Channel 3 - Secondary Loop Sample Tank

The third possible effluent channel consists of the leakage from the primary loop to the secondary loop. A 100 ml sample of secondary coolant

TABLE 11-3

SUMMARY OF LIQUID WASTE RELEASED FROM UFTR/NUCLEAR SCIENCES COMPLEX* SINCE REACHING THE LICENSED POWER LEVEL FOLLOWING SYSTEM MODIFICATIONS IN 1970

Reporting Period	Volume Discharged to UF Campus Sanitary Sewage System (liters)	Maximum Activity In Any Release (μCi/mlβ)
Sept. 1, 1971-Aug. 31, 1972	-	-
Sept. 1, 1972-Aug. 31, 1973	249,800	$1.2 \times 10^{-}$
Sept. 1, 1973-Aug. 31, 1974	412,600	2.1×10^{-7}
Sept. 1, 1974-Aug. 31, 1975	639,000	2.1×10^{-7}
Sept. 1, 1975-Aug. 31, 1976	605,000	1.3×10^{-7}
Sept. 1, 1976-Aug. 31, 1977	279,200	7×10^{-8}
Sept. 1, 1977-Aug. 31, 1978	340,000	2×10^{-8}
Sept. 1, 1978-Aug. 31, 1979	645,000	5.5×10^{-8}
Sept. 1, 1979-Aug. 31, 1980	618,000	1.7×10^{-8}

*The liquid effluent discharged into the holding tanks comes from approximately twenty laboratories within the adjacent Nuclear Sciences Center as well as from the UFTR building complex.

TABLE 11-4

SUMMARY OF ROUTINE UFTR Ar-41 RELEASES
SINCE LICENSING TO 100 KWTH

Reporting Period	Ar-41 Released Ci	Maximum Monthly Concentration* µCi/ml	Maximum Concentration of Monthly Releases** µCi/ml
Sept. 1, 1971-Aug. 31, 1972	-	-	-
Sept. 1, 1972-Aug. 31, 1973	9.6	8.3×10^{-7}	4.15×15^{9}
Sept. 1, 1973-Aug. 31, 1974	3.7	2.8×10^{-7}	1.4×10^{-9}
Sept. 1, 1974-Aug. 31, 1975	18.0	4.1×10^{-7}	2.0×10^{-9}
Sept. 1, 1975-Aug. 31, 1976	5.03	1.7×10^{-7}	8.5×10^{-10}
Sept. 1, 1976-Aug. 31, 1977	113.2	1.52×10^{-6}	7.58×10^{-9}
Sept. 1, 1977-Aug. 31, 1978	129.53	1.97×10^{-6}	9.84×10^{-9}
Sept. 1, 1978-Aug. 31, 1979	40.46	5.4×10^{-7}	2.7×10^{-9}
Sept. 1, 1979-Aug. 31, 1980	51.87	7.0×10^{-7}	2.0×10^{-9}

*MPC for an uncontrolled area is 4.0×10^{-8} µCi/ml.
**Reflects the Authorized Atmospheric Dilution Ratio of 200 to 1.

is taken weekly from a sample tank that collects a representative amount of the secondary coolant discharge (3 gallons per 40 hour week). These samples are dehydrated and counted for detectable α and β contamination. Excess samples are poured into the Waste Water Holdup Tanks and planchetts used for holding the dehydrated samples are disposed of in contaminated waste if necessary. No leakage has been indicated to the present, and based on the safety limits imposed on the UFTR design, none is expected. Therefore, although this secondary loop is a monitored effluent channel, there is essentially no release above background through this channel.

12. RADIATION PROTECTION

12.1 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)

The UFTR is operated by the Department of Nuclear Engineering of the University of Florida for the purpose of instruction and research. With the increased utilization of ionizing radiation at the University of Florida, the administration established a University-wide Radiation Control Program on September 23, 1960. The Radiation Control Program establishes a Radiation Control Committee and a Radiation Control Officer which, together, ensure that occupational radiation exposures are maintained as low as reasonably achievable (ALARA). Line responsibility for the University of Florida Radiation Control Office derives from the President and resides primarily with the Radiation Control Officer as indicated by the flow diagram presented in Figure 12-1. The primary purpose of the program is to assure radiological safety to all University personnel and the surrounding community, and to make certain that sources of ionizing radiations are procured, utilized and stored in accordance with Federal and State regulations. The Radiation Control Committee is responsible for advising the President of the University on all matters related to radiation safety, for reviewing and approving all proposed procurement and use of radioisotopes and machines generating ionizing radiation including the UFTR. The specific responsibilities of the Radiation Control Committee are enumerated below as set forth in a memorandum from the Office of the President of the University of Florida dated September 23, 1950 and updated and revised in the latest January, 1979 issue of the University of Florida Radiation Control Guide. (34)

1. Review and grant permission for or disapprove the use of radioactive isotopes or other sources of ionizing radiation within the institution from the standpoint of radiation safety.

2. Prescribe special conditions and requirements which may be necessary to assure radiation safety (for example, physical examinations, additional training, designation of limited areas or locations of use, disposal methods, etc.).

3. Prepare and disseminate information on radiological safety including University, State, and Federal regulations governing ionizing radiation for use and guidance of students and staff.

4. Pass judgment on the adequacy of safety measures for safeguarding University research workers. Committee approval of health and safety measures must be obtained before initial use of radioisotopes or other ionizing radiation is undertaken or before substantially different uses from those originally approved by the Committee are undertaken. After the issuance of a restraining order by the Committee, the staff member concerned would have a final recourse to the President after approval for such action by the staff member's Dean or Director.

5. Keep records of the actions taken in approving the use of radioisotopes and other sources of ionizing radiation and other transactions, communications, and reports involved in the work of the Committee.

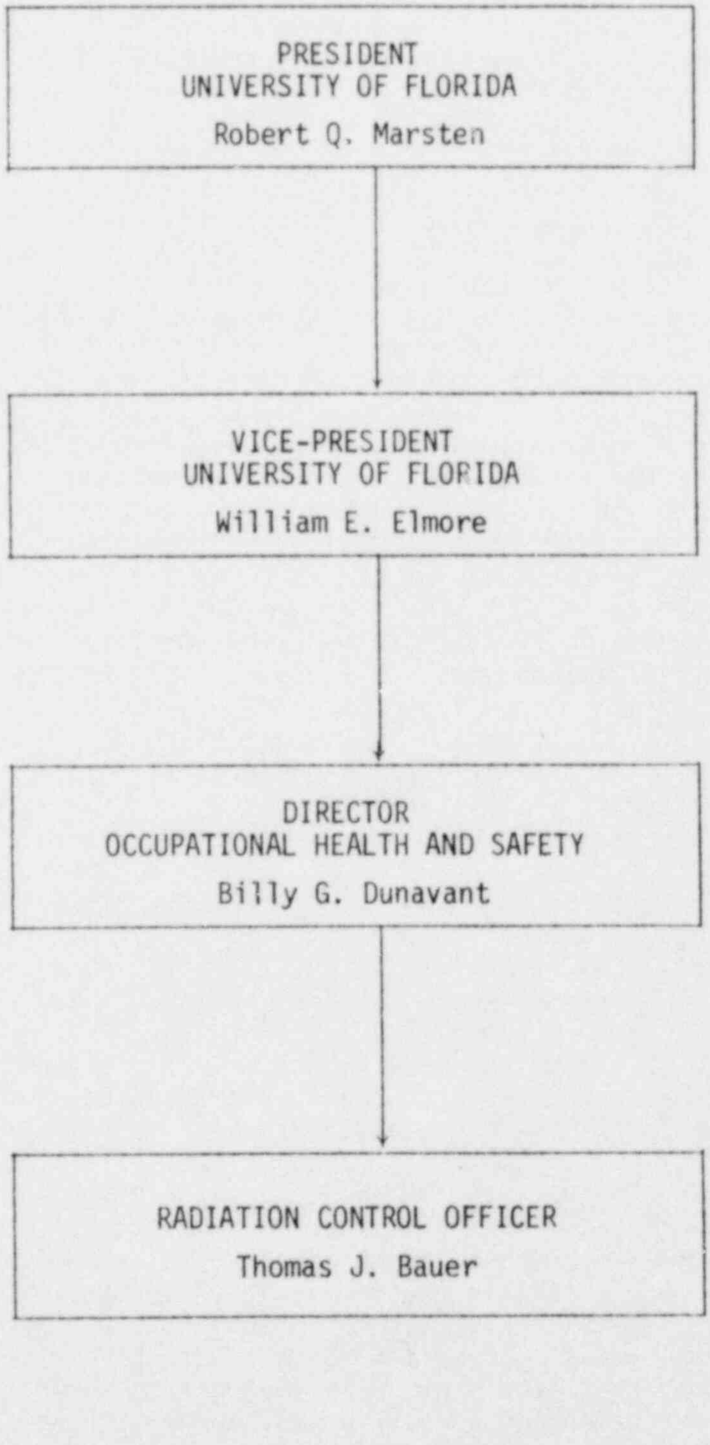


Figure 12-1. Line Responsibility Flow Diagram for the University of Florida Radiation Control Office

6. Delegate to the Radiation Control Officer the authority to act for the Committee between meetings. (Actions of the Radiation Control Officer are reported to the Committee for review at appropriate intervals.).

7. Review plans for all new buildings and modifications of existing structures where ionizing radiation is to be used.

9. Periodically review actions of the following subcommittees of the Radiation Control Committee:

   (a) Training Reactor (UFTR Subcommittee)
   (b) Subcritical Subcommittee

10. Review, at least annually, the activities of the Committee on Human Use of Isotopes from a radiation safety standpoint.

As set forth in the above-mentioned Presidential Memorandum and revised and updated in the January, 1979 issue of the University of Florida Radiation Control Guide, the duties and responsibilities of the Radiation Control Officer include: (24)

1. Administer and be responsible for the overall day-to-day programs of the University's Radiation Control Office.

2. Approve all University procedures which might conceivably involve radiation exposure and all changes in such procedures.

3. Act in a supervisory capacity in all aspects of the University radiation measurement and protection activities, such as personnel monitoring, maintenance of exposure records, survey methods, waste disposal, and radiation safety practices.

4. Consult with all potential radionuclide users and advise on radiation safety practices.

5. Suspend any operation causing excessive radiation hazards as rapidly and safely as possible. (In carrying out this duty, the Radiation Control Officer reports directly to the President.).

6. Maintain a list of all employees who may be exposed to ionizing radiation.

7. Prescribe routine radiation survey and personnel monitoring as deemed necessary.

8. Establish standardized procedures for the procurement of radioactive materials.

9. Serve as an ex-officio member of all radiation safety committees constituted at the departmental, college, experiment station, or University levels.

Administrative Affairs Memorandum No. 22 of May 24, 1974, structures a Radiation Control and Radiological Services (RC&RS) Department, headed by the Radiation Control Officer, under the Environmental Health and Safety Division.

The Radiation Control Officer is specifically responsible for implementing and enforcing the radiological safety program at the UFTR facility. The actual minimizing of occupational radiation exposures to meet the ALARA objective is the direct concern of the staff and faculty associated with the UFTR facility.

12.1.1 Design Considerations

The UFTR has been designed and is controlled to minimize and achieve "as low as reasonably achievable" radiation exposures during normal operation. This level of safety is a result of several basic design principles:

1. Radiation shielding of a type, quantity and sequence to permit maximum experimental irradiation with a minimum of radiation exposure to faculty, students and personnel involved with experimental activities.

2. Complete containment of fuel and fission products within the core and associated auxiliary equipment.

3. Simplicity of the design and operation of the reactor to assure high system reliability.

4. Conservative design demonstrated in a low power density to assure large safety margins in all operating conditions.

Past operating experience with the UFTR facility rated at 100 KWth demonstrates that the system design is compatible with and adequate for minimizing occupational exposures during operation. Further experience with the operation of the similar argonaut-type UCLA training reactor in Los Angeles, California at both 100 KWth and 500 KWth rated power levels further demonstrate that the design features of this UFTR do ensure not only that occupational exposures are kept far below 10 CFR 20 limits but also that occupational exposures meet the ALARA criterion.

12.1.2 Operational Considerations

The UFTR is essentially a minimum release facility excluding the Ar-41 releases; by proper operation, minimal radiation levels are encountered during normal operation. In general, the operating philosophy for maintaining occupational radiation exposures as low as reasonably achievable for the UFTR facility, will follow the guidelines put forth in NRC Regulatory Guide 8.10. However, the management of the UFTR facility is strongly committed to maintaining exposures as low as is reasonably achievable. All facility personnel are made aware of this goal and are required to follow and abide by the procedures and preset limits set forth in the UFTR standard Operating Procedures, Technical Specifications and other documents related to assuring the ALARA criterion is met.

Since the primary purpose of the reactor is to train students, it is necessary to emphasize to the students at the outset the danger of carelessness around the reactor and the need to keep exposures to a minimum. The Radiation Control Officer or the facility personnel instruct students regarding hazards and safety practices during their first session at the reactor.

Radioactive samples are removed from the reactor only under careful supervision of a qualified staff member according to the SOP's and with approval of a qualified reactor operator. When necessary, shielding is used to reduce radiation levels to safe values and conclusive radiation surveys are taken during the transfer of radioactive samples or other materials. All persons handling radioactive materials are instructed in correct procedures, use of survey instruments, and allowable radiation levels. In addition, students and faculty using the facility are kept informed on the subject of radiation protection through the Office of Radiation Control. The Radiation Control Office distributes University of Florida Radiological Control Guides and requires proven training and expertise in the handling and control of radioactive substances before issuing permits for possession of radioisotopes. UFTR personnel release radioactive materials to approved users, as determined by Radiation Control. Facility personnel are trained and yearly qualified on radiation control through the UFTR Training and Requalification Programs.

Detailed specifications, procedures to be followed, and records to be kept in order to assure that ALARA exposures are met and documented during UFTR operations found in the UFTR Standard Operating Procedures. SOP D.1, "UFTR Radiation Protection and Control" describes the general requirements and limits which must be observed to assure the ALARA radiation exposures. Specific procedures to be followed during maintenance operations are included in the E-series of SOP's from E.1 through E.5. Specific procedures and radiation limits related to refueling operations are included in SOP C.1 and SOP C.2. Procedures concerning radioactive waste handling are found in the Radiation Control and Radiological Services - Operating Instructions, Appendix II.

The UFTR subcommittee of the UF Radiation Control Committee performs formal audits periodically to determine ways by which to reduce exposures to individuals based on exposure records and recommendations from the UFTR facility operating personnel. Radiation protection responsibilities at the UFTR facility are assigned as described in Chapter 13 to provide effective radiation protection. These responsibilities of the operating organization at the UFTR facility are defined in detail in Section 13.1.1.

In general, the Radiation Control Officer is given authority to enforce safe plant operation for radiation protection as shown in Technical Organization in Section 13.1.2 Any modifications in facility operating and maintenance procedures which have the potential to reduce exposures are considered for implementation by the UFTR Subcommittee. In general, the Radiation Control Officer and all facility personnel are familiar with sources of radiation exposure and will always try to reduce exposures to a minimum by all means available. The UFTR Subcommittee is a subcommittee of the Radiation Control Committee and functions as the UFTR Safety Committee.

12.2 Radiation Sources

12.2.1 Contained Sources

The sources of radiation which are the basis for the radiation protection during reactor operation are due mainly to the core-produced neutron flux and the gamma ray fluxes originating from the core and from activated sources (mainly, the activation of the biological shield).

Sources activated for experimental purposes in the UFTR experimental ports are a concern after reactor shutdown. Radiation surveys of the reactor cell and adjacent areas at different power levels have shown an essentially linear relationship between radiation levels and operating power levels as shown in Figure 12-2. This linear relationship is to be expected. Radiation survey data taken around the reactor cell and adjacent areas with no external shielding is presented on the UFTR cell floor plan of Figure 12-3.

From previous radiation exposure measurements it is known that both thermal and fast neutron contributions to the radiation field in the reactor cell are negligible and thus the main concern is the gamma exposure. (23)

The UFTR is provided with two startup sources--an Sb-Be source of less than 25 curies fully charged and a one curie Pu-Be source. As sources of exposure, these startup sources can be neglected when compared to the other radiation sources mentioned above.

12.2.2 Airborne Radioactive Material Sources

The UFTR is designed to be a minimum leakage facility. Fission products are contained within the fuel elements and credible escape mechanisms of these fission products during normal or abnormal reactor operation are not available since the integrity of the fuel is not affected at any time during reactor operation.

The only isotope normally considered is the Ar-41 produced in the UFTR as a result of activated Ar-40 produced from air leaking through cracks in the concrete and graphite shielding, as discussed in Chapter 11. Leakage of these activated gases into the reactor cell is prevented by drawing air from the cell, through the reactor and out the exhaust stack with the Ar-41 constituting the only radioisotope of concern. Rough and absolute filters in the Reactor Vent System minimize the discharge of air particulate to the environment.

12.3 Radiation Protection Design Features

The simplified design and low radiation levels associated with the UFTR facility greatly reduce the presentation requirements of this section.

12.3.1 Facility Design Features

The UFTR facility is of the modified Argonaut type, designed to minimize radiation exposure to all individuals. Since the reactor is used as a teaching tool and for research operations, a more stringent safety program has been developed to insure radiation exposures meet the ALARA criterion; all the UFTR SOP's are designed to facilitate the minimization of exposure rates and to insure the health and safety of the people in and around the facility.

To ensure occupational radiation exposures are ALARA, the control console is located at an adequate distance from the reactor and isolated

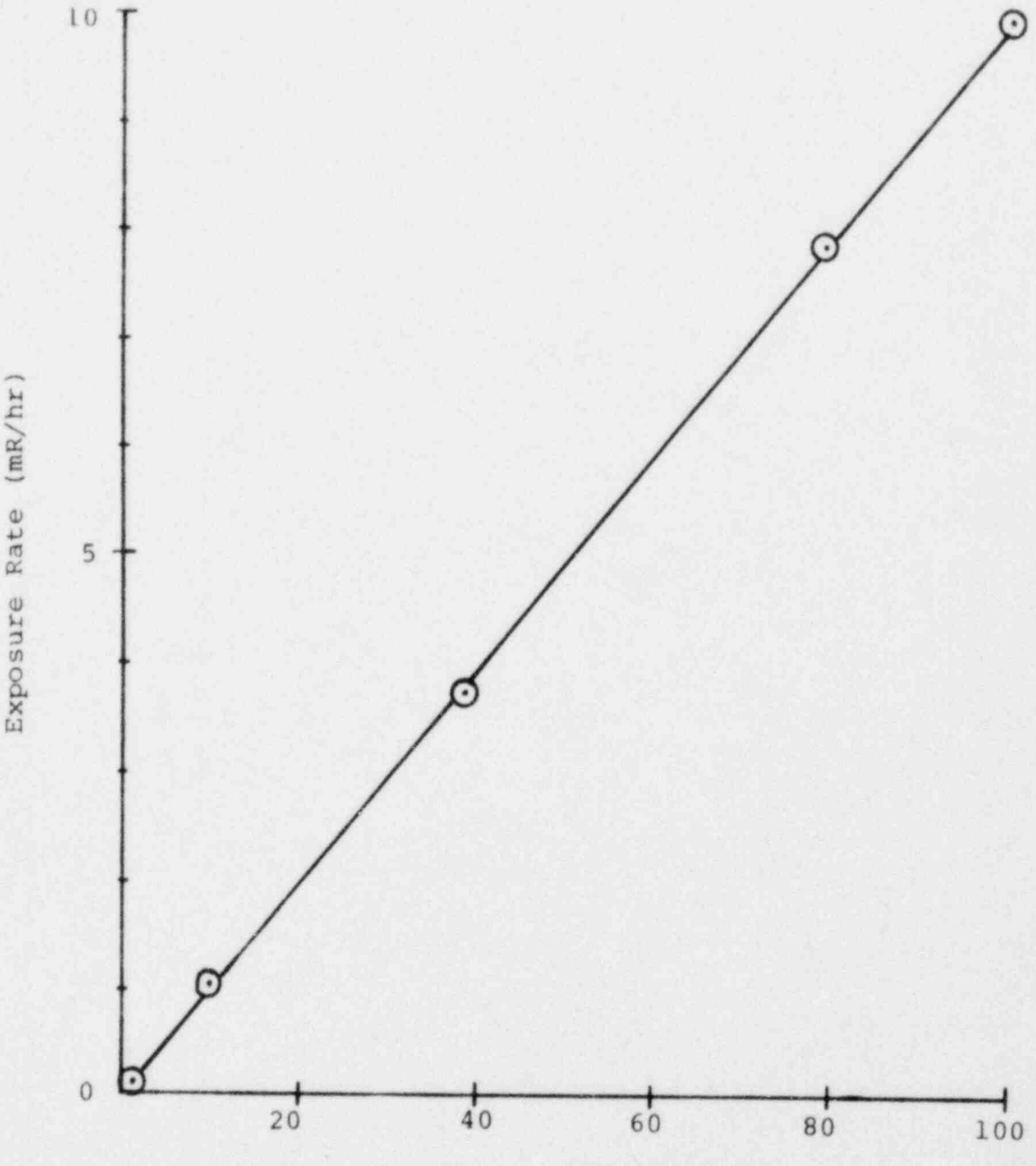


Figure 12-2. Exposure Rate Versus UFTR Operating Power (23)

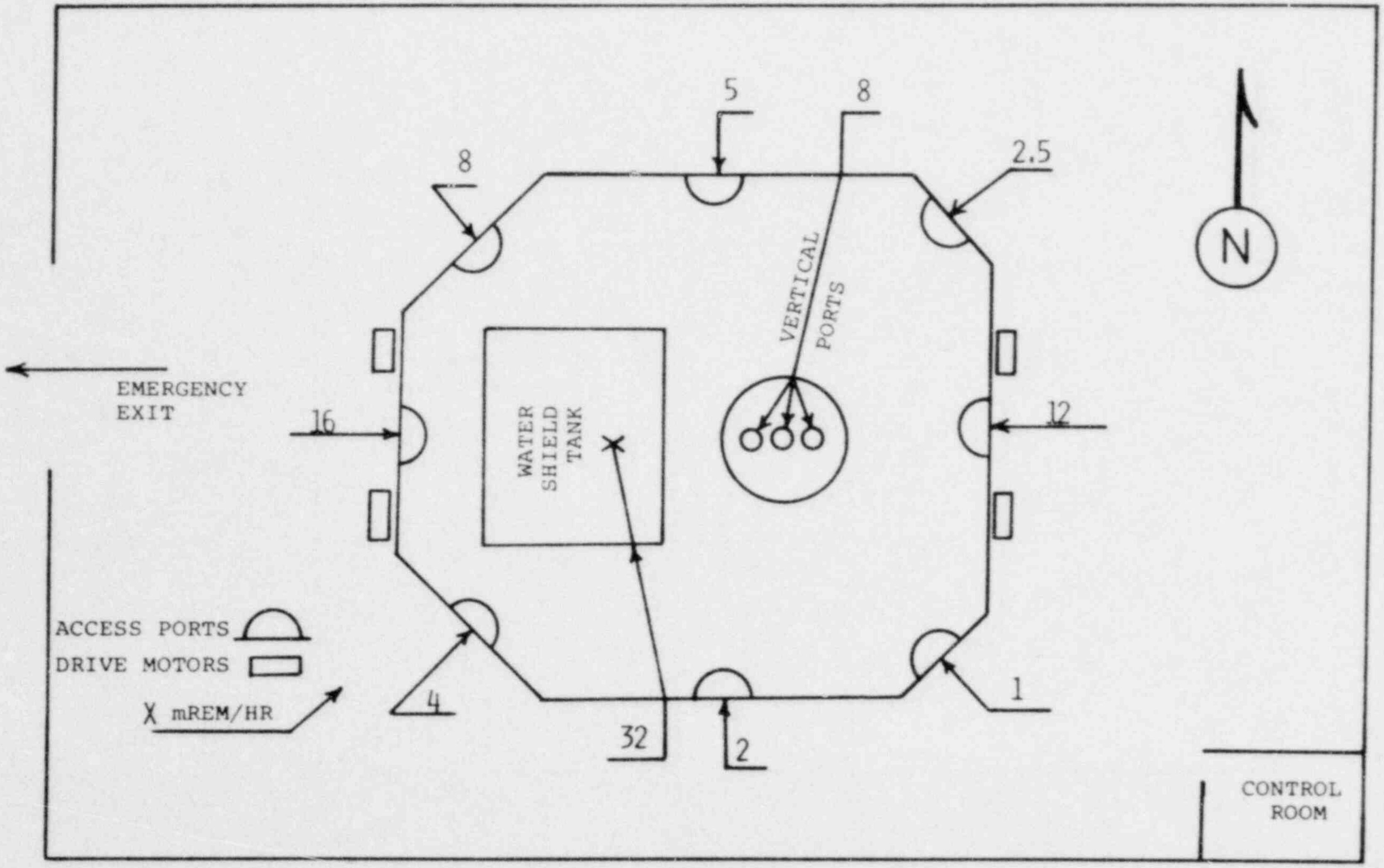


Figure 12-3. Results of Radiation Survey Around UFTR at 100 KWth Power Operation. (23)

from the rest of the reactor cell by a plate-glass wall enclosure which allows good visibility of the reactor cell during operations. No exposure facilities (beam ports) face in the direction of the console. Samples can be changed easily in the beam ports with the reactor shut down. Whenever experimental requirements necessitate operation of the reactor with a shield plug removed, for example, to extract a bea or insert some apparatus, strict health physics supervision is required. All such experiments are approved in advance by the UFTR Subcommittee of the Radiation Control Committee; in addition, adequate shielding must be provided as specified in the UFTR SOP A.5, "Experiments," to assure that the ALARA criterion and safety considerations are satisfied.

Whenever a proposed experiment indicates the need for extraction of a beam from an open port, the associated radiation levels are estimated prior to conducting the experiment. Adequate shielding is then constructed and placed in position while radiation monitoring is required for the experiment itself. These areas around such experiments may also be roped off and posted whenever deemed necessary to minimize the radiation exposure of all personnel in the facility.

All samples activated in the reactor are removed according to the UFTR SOP D.6 entitled, "Removing Irradiated Samples from UFTR Experimental Ports." Additional shielding in the form of lead bricks and concrete blocks is available for any activated sources removed from the exposure facilities. In addition, a hot cave with remote handling facilities is available in the radiochemistry laboratory outside the reactor cell on the first floor of the UFTR building in the event it is needed.

12.3.2 Shielding

During normal operation at the 100 KWth rated power level, the shielding supplied by the present system is adequate for all the "core" and activation (biological shield) sources of radiation discussed in Section 12.2.1. Additional shielding is available in the form of cast concrete blocks which can be used as special shielding during experiments, around activated sources, or during high power operations. In order to reduce the radiation exposure from experiments to ALARA levels, radiation surveys are conducted for all except routine experiments to determine whether special shielding configurations are needed. When necessary for the ALARA criterion, such special shielding configurations are installed.

12.3.2.1 Radiation Surveys. Studies have been conducted in the reactor cell and adjacent areas to evaluate the ability of the UFTR's biological shield to provide adequate radiation protection at the rates 100 KWth power level. Previous exposure measurements have indicated that both thermal and fast neutron contributions to the radiation hazard in the reactor cell are negligible; therefore, only gamma radiation exposures are considered in this report as recorded by Wagner. (23) Results obtained from radiation surveys around the reactor structure during operation at the 100 KWth power level are indicated on the sketch of the reactor cell layout presented in Figure 12-3 discussed in Section 12.2.1. It is important to note that this radiation survey data represents the reactor with no external shielding. Additional temporary shielding is available and used during normal high power operation. This additional shielding consists of lead bricks at the base of the reactor on the north and south faces and a large cast concrete block on the west side.

All gamma exposure rates were recorded with a vibrating-reed Victoreen Survey Meter (Model-440, Serial-440) which was calibrated with a Cobalt-60 source. Additional survey results for various areas within the reactor cell are given in Figures 12-4 through 12-7. Figures 12-4 and 12-5 indicate that a strong source of radiation existed below the reactor resulting from a large void (approximately 3.5' x 0.56' x 8.0') below the fuel box support structure. Following filling of this void with sand, the results of later measurements of gamma exposure rates shown by smaller numbers in Figure 12-5 indicate that the strong source of radiation from the void is essentially eliminated from concern. High levels of radiation are also indicated on the north and south sides of the shield tank as indicated in Figures 12-6 and 12-7. The radiation level above the shield tank is reduced from about 150 mR/hr to 15 mR/hr by the shield block cover. Area monitors which are located on the upper part of the north, east, and south walls of the reactor cell indicate readings of 1.0, 1.5, and 0.4 mR/hr respectively, and show little dependence upon whether or not the shield block is over the shield tank. Additional surveys in this series indicate the following radiation levels for certain special areas at 100 KWth operation:

1. Radiation levels of $\sim$0.4 mr/hr are recorded at the console in the reactor control room.

2. An average exposure rate of $\sim$0.9 mr/hr is recorded in the area directly outside the emergency exit doors to the West of the Reactor.

3. A radiation level of $\sim$0.2 mr/hr is recorded at the surface of the wooden door leading to the workshop to the north of the reactor building which houses the dilution fan for Argon-41 stack releases. Exposure rates are below 0.05 mr/hr outside the concrete barrier surrounding the fan.

The most complete recent survey of UFTR cell radiation levels during 100 Kw operation is summarized in Appendix 12A at the end of this chapter.

12.3.2.2 Shielding Calculations. Several approximate calculations with results performed to estimate the relative importance of prompt, delayed, and capture gammas for the 100 KWth UFTR power level are presented in this Section. All the required information to perform these calculations and the corresponding results obtained are included in Table 12-1. (23) The prompt fission gamma-rays are considered first. The prompt source spectrum used for these calculations is presented in Table 12-1. To compute the fraction of these gammas which escape the fuel box region of the UFTR core, both fuel slabs are considered as one homogenized slab at the center of the core. Volume-weighted, energy-dependent, energy absorption coefficients (34) are then used in conjunction with results of Case (35) to determine fuel slab escape probabilities calculated using the chord method. These probabilities are also presented in Table 12-1. The fraction of prompt gammas escaping the fuel for each energy group is then separately treated as a point source; this treatment is a good approximation since the source width is much less than the attenuation distance. The resultant UFTR reactor shielding model shown in Figure 12-8 allows simple exposure calculations to be performed

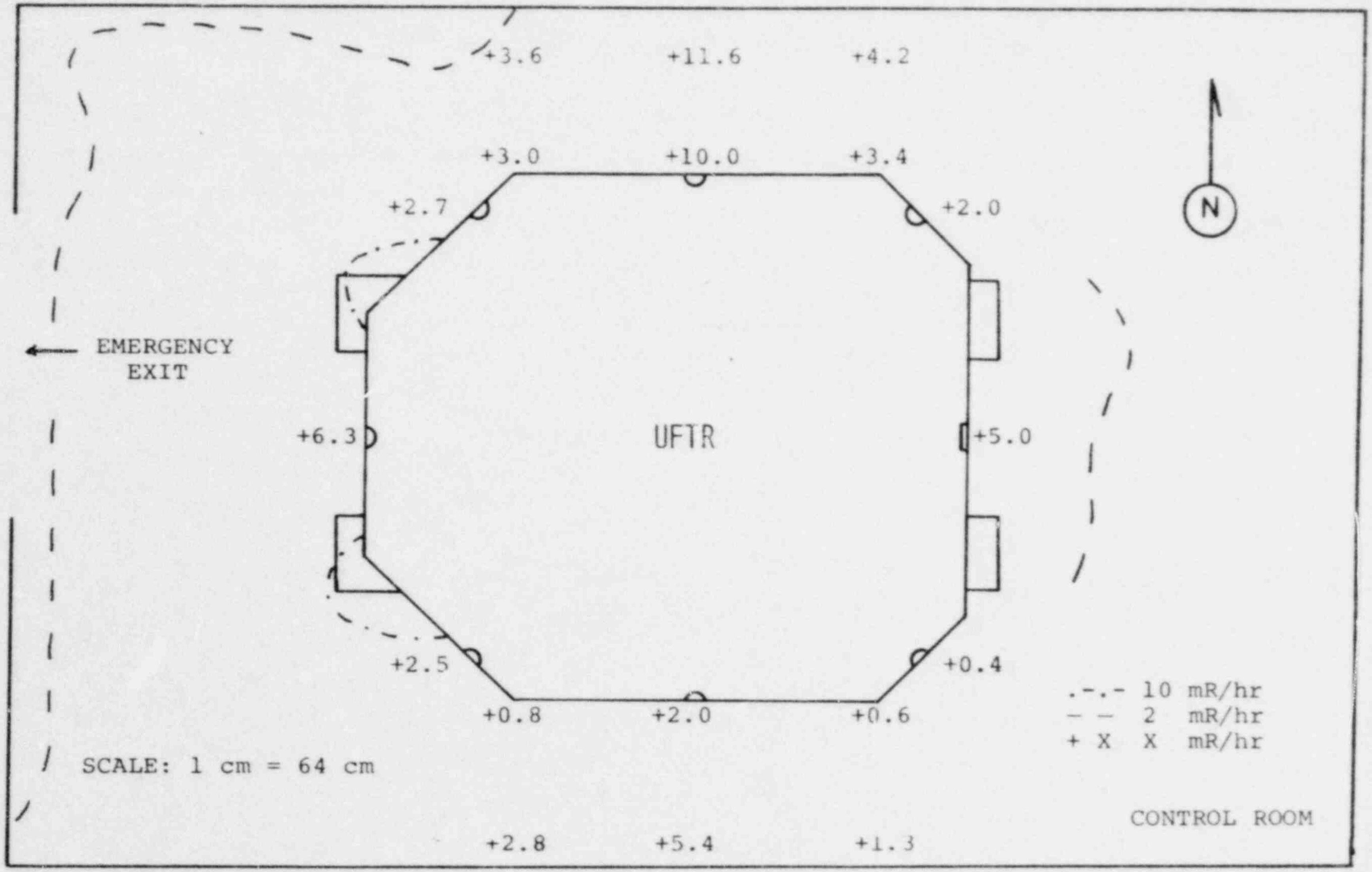


Figure 12-4. Gamma Exposure Rates at Port Level for 100 KWth Operation With No External Shielding and Top Shield Block Removed.

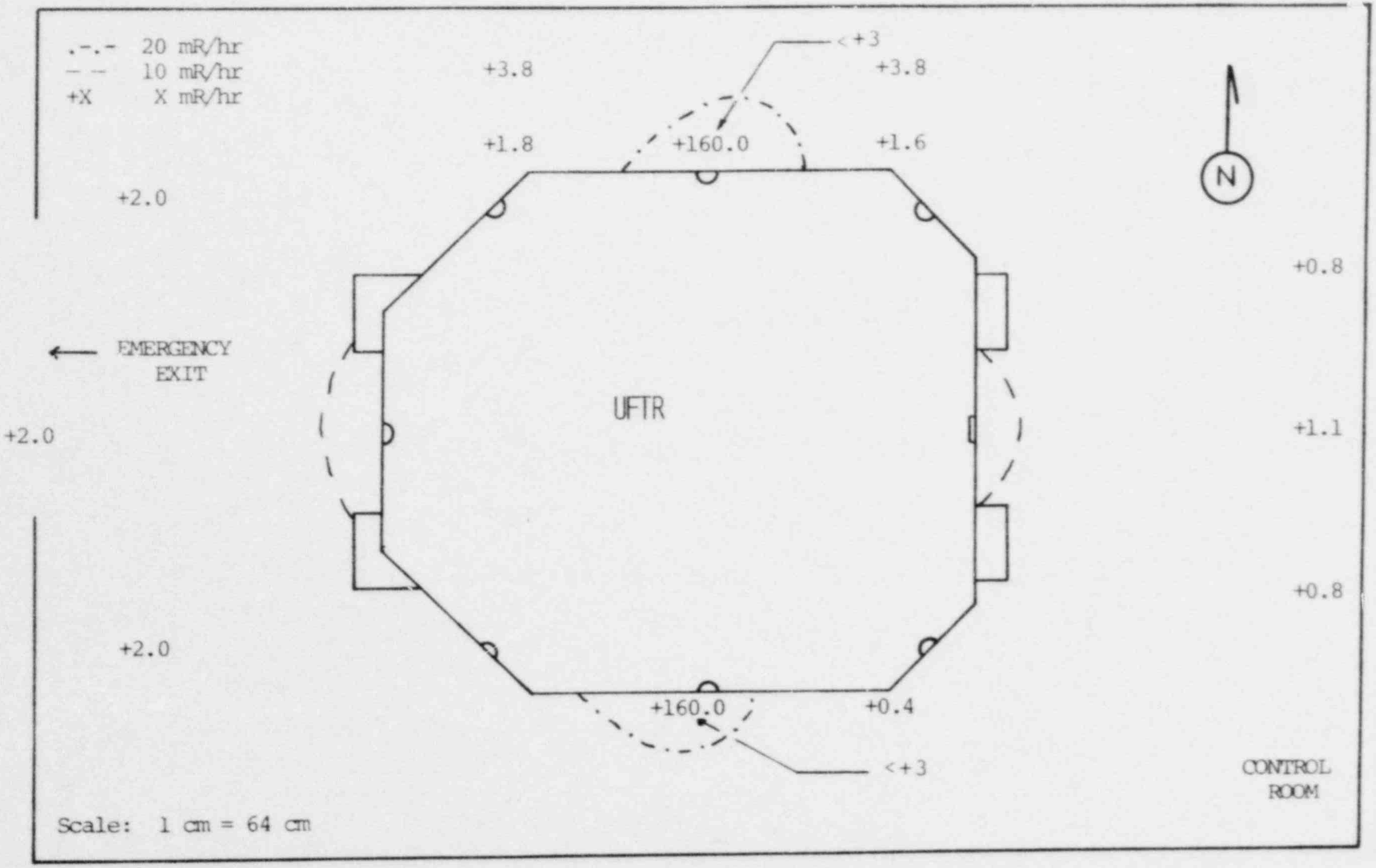


Figure 12-5. Gamma Exposure Rates at Ground Level for 100 KWth Operation With No External Shielding And Top Shield Block Removed.

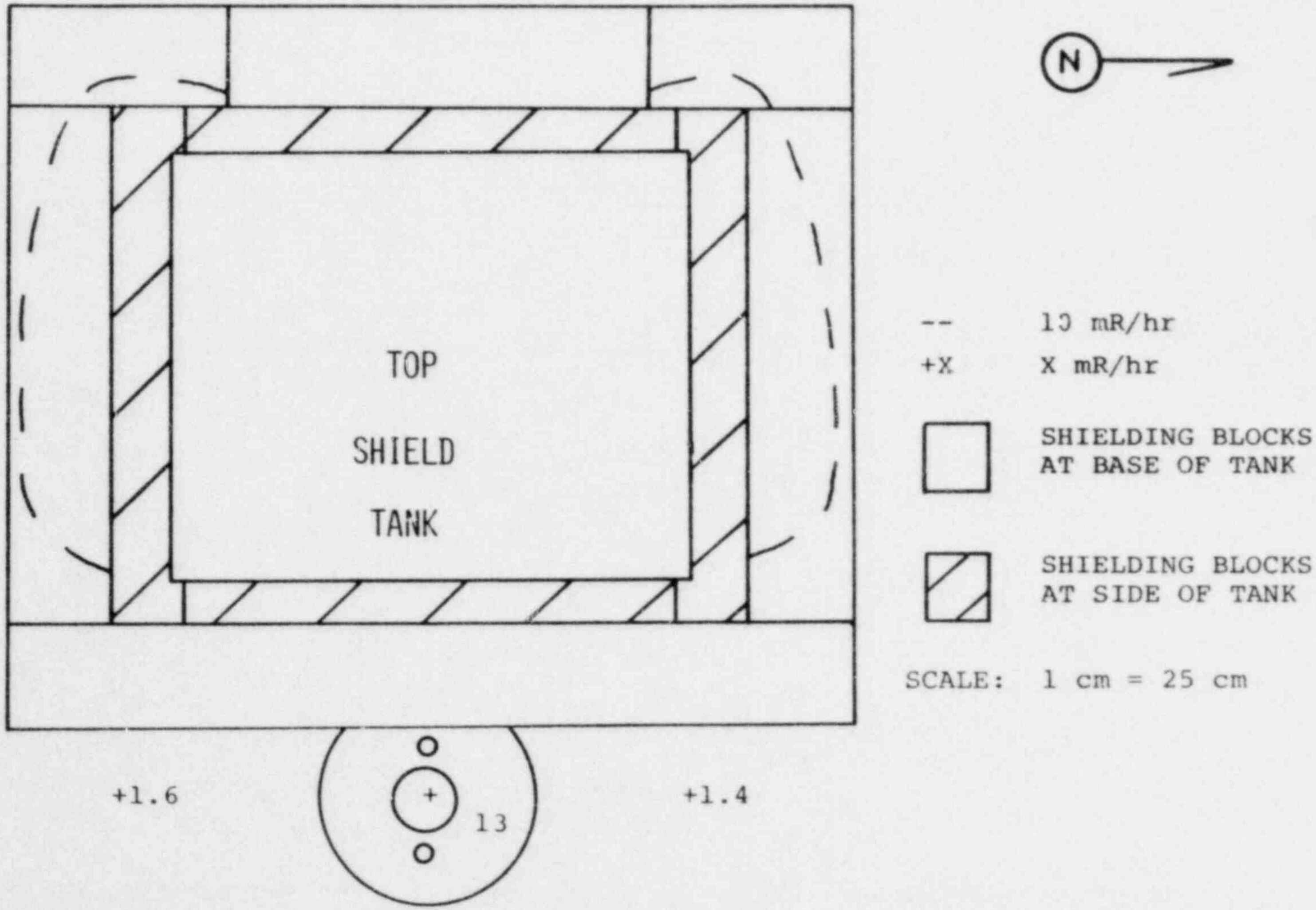


Figure 12-6. Gamma Exposure Rates Around the UFTR Shield Tank for 100 KWth Operation with Readings Made At the Top of Base Shielding (25 cm above the reactor surface).

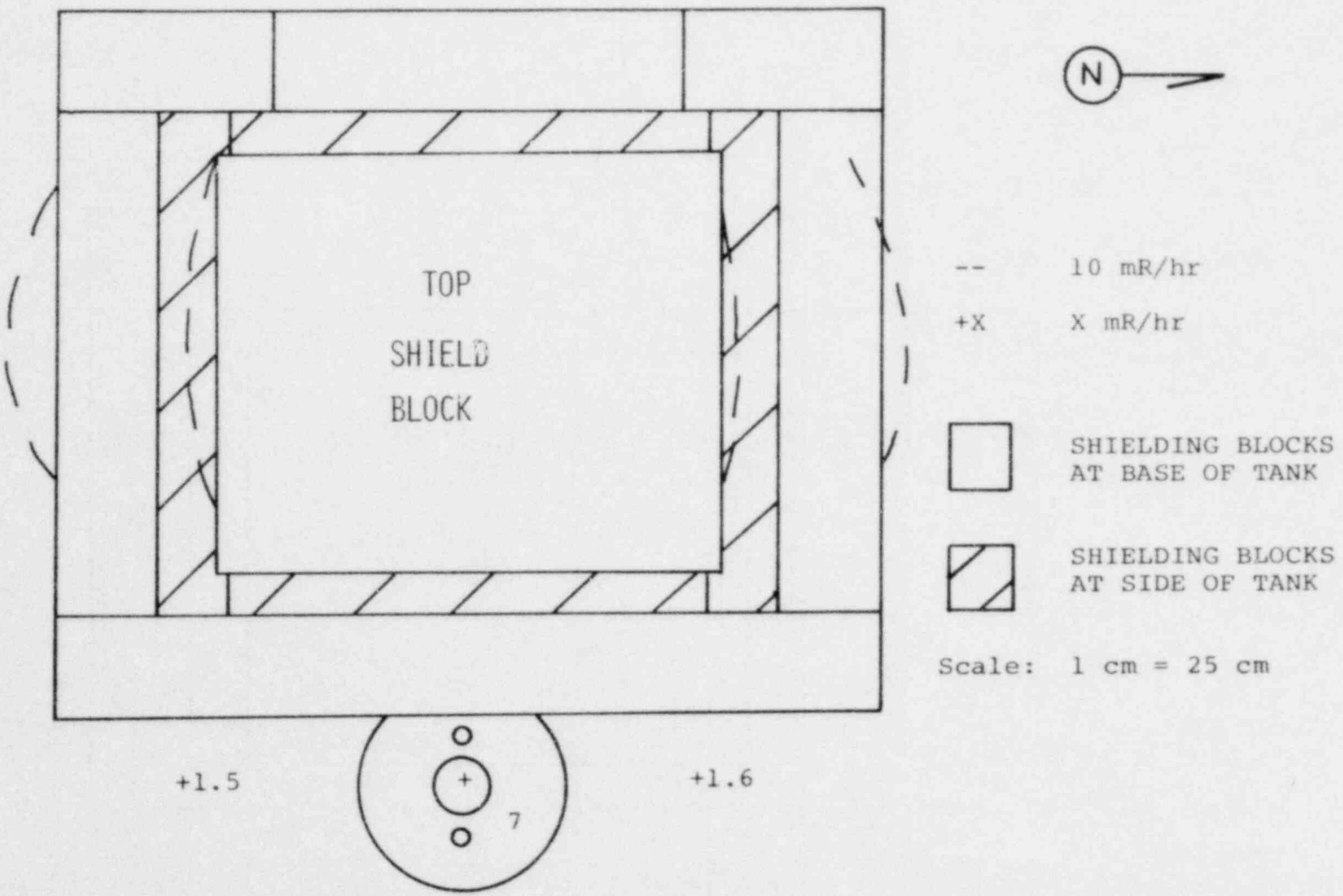


Figure 12-7. Gamma Exposure Rates Around the UFTR Shield Tank for 100 KWth Operation With Readings Made At the Top of the Top Shield Block (101 cm Above the Reactor Surface).

TABLE 12-1

SUMMARY OF SHIELDING CALCULATIONS FOR 100 KWTH OPERATION (23)

Data for Equation (12-1)

Effective Energy (MeV)	Prompt Gammas (MeV/sec)(36)	Fuel Slab Escape Probability (35)	Buildup Factor for Barytes Concrete (37)	k(MeV/cm^{2}sec /mR/hr) (34)
1	1.07 E+16	0.3386	37.0	548
2	9.56 E+15	0.3775	18.0	651
4	3.21 E+15	0.4310	11.0	819
6	7.94 E+14	0.4552	10.3	928

Calculated Shielding Results at North or South Face of Barytes Concrete

Effective Energy (Mev)	Prompt Gamma $\dot{D}_{AT\ P}$	Prompt Gamma $\dot{D}_{AT\ P}$ with addition of 6" Poly-B-Pb	Barytes Concrete Capture Gamma $\dot{D}_{AT\ P}$	Barytes Concrete Capture Gamma $\dot{D}_{AT\ P}$ with addition of 6" Poly-B-Pb
1	0.51	0.05		
2	3.16	0.53		
4	1.26	0.53		
6	0.26	0.26		
	Total = 5.19	Total = 1.47	Total = 7.9	negligible

NOTE: $\dot{D}_{AT\ P}$ = exposure rate (in mR/hr) at point P.

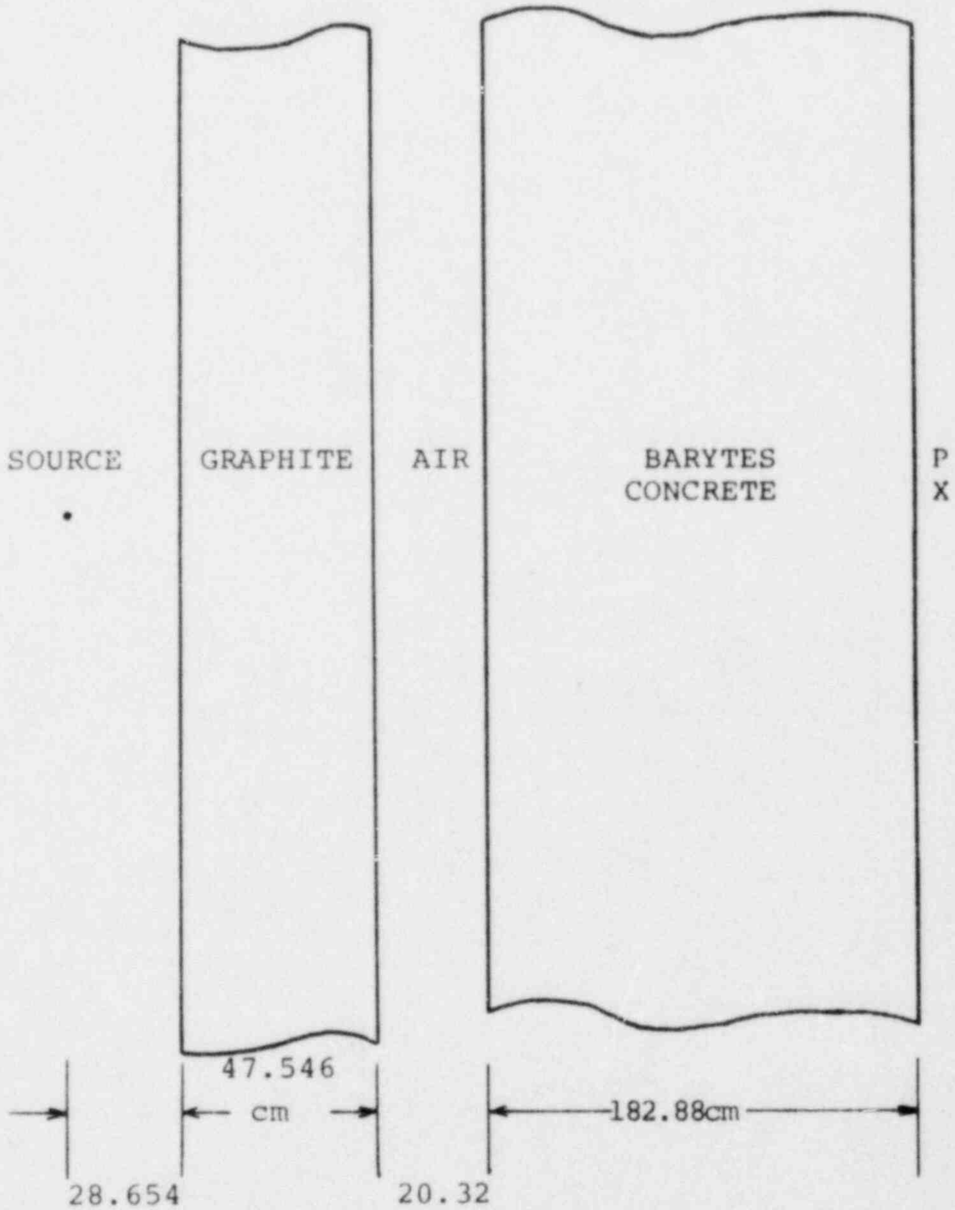


Figure 12-8. Core Gamma Shielding Model for North or South Face of the UFTR.

for the north and south faces of the concrete through use of Equation (12-1):

$$\dot{D}_{A+P} = \frac{BSe^{-b}}{4\pi r^2 K} \tag{12-1}$$

where $\dot{D}_{A+P}$ = exposure rate at point P(mR/hr);

$b = \sum_{i=1}^{n} \mu_i t_i$

μ_i = energy absorption coefficient for gamma rays in group 1 in the shield material (cm^{-1});

t = shield thickness (cm);

n = number of energy groups--four here;

S = strength of point source (MeV/sec);

r = distance from Source to Point P (cm).

B = buildup factor

K = conversion factor (MeV/cm^2-sec to mR/hr).

Equation (12-1) is applied for each gamma energy group to obtain the results outlined in Table 12-1. Buildup is considered only for the barytes concrete sections since it is heavier and larger than the preceding shielding material. (38)

Application of the same calculational method shows that the delayed gammas resulting from fission product decay make a negligible contribution to the exposure rate at point P. This is true for the UFTR since its equilibrium fission product buildup corresponds to an average operating history of no more than one KWth.

A possible method to reduce the prompt core gamma exposure at point P in the event UFTR licensing to higher powers is approved is to add about six inches of polyethylene shielding containing 1 percent boron and 80 percent lead (Poly-B-Pb) in the air gap depicted in Figure 12-8. The resultant exposure rates calculated using Equation (12-1) are significantly reduced as indicated in Table 12-1.

Capture gamma rays produced within shielding materials can represent a significant portion of the radiation hazard for a reactor. For the UFTR this problem arises mainly from thermal neutron capture in the barytes concrete, since capture gammas from the graphite are negligible. (39) To evaluate the exposure rate at point P due to this effect, Equation (12-2) is used: (39)

$$\dot{D}_{A+P} = \frac{\Sigma_a E\phi_o \exp(\mu_e T/2)}{2K[\lambda+(\mu_e/2)]} \Big| \exp(-\lambda-\mu_e/2)T \cdot E_1\{(\mu T-\mu x)\,(1 - \frac{\lambda+(\mu_e/2)}{\mu})\}$$

$$- \exp(-\lambda+\mu_e/2) \cdot E_1\{\mu T-\mu x\}\Big|_{x=0}^{(T-\varepsilon)} \qquad (12\text{-}2)$$

where $\dot{D}_{A+P}$ = exposure rate at point P (mR/hr)

Σ_a = thermal neutron absorption cross section in barytes concrete = 0.0197 cm^{-1} (37)

E = energy of capture gamma rays = 7.2 MeV

ϕ_o = thermal neutron flux density at inner face of barytes concrete = 2.9 x 10^{10} n/cm^2 sec

μ_e = energy deposition coefficient of gamma rays in barytes concrete = 0.0857 cm^{-1} (37)

μ = energy deposition coefficient of gammas in water = 0.046 cm^{-1} (39)

T = thickness of barytes = 182.88 cm

K = conversion factor = 946 MeV/cm^2·sec/mR/hr (34)

λ = attenuation factor for thermal neutron flux in barytes concrete = 0.125 cm^{-1} (37)

$E_1(y) = \int_y^\infty \frac{e^{-x}}{x}\,dx$

ε = incremental value > 0; required to keep $E_1(y)$ finite.

Equation (12-2) assumes the attenuation of neutrons in the barytes is represented by $\phi = \phi_o e^{-\lambda t}$ with an energy buildup factor. ϕ_o is found from the neutron flux computer calculations discussed in Chapter 4 of this Safety Analysis Report (23) and represents the total neutron flux in energy group 4. The assumption E = 7.2 MeV is a conservative approximation to the capture gamma spectrum for concrete. (39) Using $(T-\varepsilon) = 0.9T$ and the fact that $E_1(-y) = -E_i(y)$ (39), where these values are tabulated in Reference 40, the exposure rate at P due to capture gamma-rays in the barytes concrete is found to be 7.9 mR/hr. The accuracy of the approximations used in Equation (12-2) is difficult to evaluate; therefore, it is assumed that this value represents an upper limit for the capture gamma radiation level. Again, the effect of adding six inches of Poly-B-Pb between the core graphite and the barytes is evaluated. Since the attenuation of the thermal flux would be of the order of 10^{10}, with low energy (0.42 MeV) gamma-rays being associated with the neutron capture process (41), it is expected that capture gamma radiation at point P would be negligible.

Radiation measurements at the north and south faces of the reactor show that the actual exposure level is approximately 3 mR/hr for 100 KWth operation, well below that predicted by the above calculations. The poor agreement is probably due to the simplicity of the calculational approach especially with the approximations involved in using Equation 12-2. The conservatism of this analysis is demonstrated, since these calculated exposure results are larger than the actual measured radiation levels.

The east and west faces do not lend themselves as easily to shielding modifications because of the geometry associated with the shield tank and thermal column. Calculations were not performed for these directions.

12.3.3 Ventilation

As presented in Section 3.1 and Section 9.4.1, the Reactor Cell is completely air conditioned. The airconditioning unit has a design capacity of 1500 c.f.m., with a total air delivery of 6050 c.f.m., at 75°F, dry bulb, and 50 percent relative humidity, summer and winter. All inlet and circulated air is filtered through a 2-in thick, dry, spun glass, cleanable-type roughing filter capable of removing particles of 5 microns or larger in size with an efficiency of 85 percent or better. The inlet air duct is provided with a motor-operated damper to close the duct whenever the unit fan is not operating.

The room exhaust air is used to ventilate the reactor structure. This air is exhausted by pulling 1 to 400 c.f.m. of room air from inside the shield. This air is passed through a roughing filter and an absolute filter to an outside stack where it is diluted with approximately 10,000 c.f.m. of outside air. It is then discharged through the stack extending above the roof of the UFTR building as discussed in Chapter 11.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area and Equipment Monitor Detector Assemblies. There are three (3) radiation monitoring systems installed in the reactor cell. These systems are located in the following positions:

1. South side of the reactor cell, ~15 feet above the exit airlock.

2. East side of the reactor cell, centered ~15 feet above the floor.

3. North wall of the reactor cell, centered ~15 feet above the floor.

Each monitoring system includes a detector, a log count rate meter, and a strip chart recorder.

The detector assemblies are of the RT-2 type. The detector is a halogen-quenched G-M detector with a life expectancy virtually unaffected by use. The sensitivity of the detector is approximately 14 cps per mr/hr, with an energy dependence compensated to 20% between 80 Kev and 2.5 MeV.

The log count rate meter is of the RS-2 type. The meter is adjusted for a four to five decade span for use with the G-M detector in the range of 0.2 to 20,000 counts per second. With the model RT-2 detector assembly, the RS-2 count rate meter is calibrated for readings in the range from 10^{-4} to 1 r/hr.

The strip chart recorder is a Gulton Rustrak D.C. Recorder. This D'arsonval meter instrument records signals from the RS-2 log count rate meter for permanent record. All three radiation monitoring systems are calibrated quartery with the assistance of the Radiation Control Office.

12.3.4.2 The Stack Monitoring System. The Stack Monitor System, consisting of a G-M detector, a log rate meter and a strip chart recorder, is equivalent to the area monitoring system. In addition to these, it also includes a log rate meter with an alarm setting capability for different operating power levels, with the information obtained from the RS-2 log count rate meter. The G-M detector is located on the downstream side of the absolute filter, before dilution takes place.

12.3.4.3 Air Monitoring System. A continuous air monitor (CAM) is designed to detect any airborne particulate activity. The Model AIM-3BL monitor manufactured by Eberline Instrument Corporation, is equipped with a flow indicator (LPM), a strip chart recorder and audible as well as visual alarm settings. The monitor is a lead-shielded, compact airborne particulate Geiger counter which may be used in an occupational environment or in confined spaces. The air is drawn through a filter paper by a constant volume pump; the activity is detected by the Geiger counter near the filter and recorded on the strip chart recorder. An alarm is activated whenever a high activity level is detected. There are two settings; one is the X-1 scale with a 3000 cpm full scale setting; the second setting is the X-10 scale with a 30,000 cpm full scale setting.

In the event of a loss of power, all radiation monitors operate on installed battery packs (rated to 8 hours) to insure their availability at all times.

12.3.4.4 Radiation Monitoring. Operators and other personnel working in the reactor wear film badges at all times. If indicated by the type of work, a direct-reading pocket dosimeter, or other dosimeter is worn as specified by the Standard Operating Procedures for the UFTR in SOP D.3 entitled "Personnel Monitoring."

Portable survey meters are also available to be used whenever it is deemed necessary and/or required by the SOP's.

Surveys performed on a weekly basis include swipe surveys, water samples and beta-gamma radiation field surveys. Surface contamination in the room is determined by means of portable instruments and smear tests. Particular attention is given to the equipment pit, experimental areas and the irradiated fuel storage pits during each survey. Periodic surveys by health physics personnel are performed to check for leakage around beam plugs and through the stacked-block reactor shield, providing a check on the proper functioning of the continuous air monitoring (CAM) system. The coolant is checked by evaporating a sample to dryness and counting with a gas flow counter.

As indicated previously, there is an on-going program by the Radiation Control Office and the UFTR facility staff to monitor radiation levels outside the UFTR building in the nearby vicinity. Monitoring is performed by placing film badges at seven (7) locations outside the UFTR building as marked on the sketch of the UFTR building and immediate vicinity presented in Figure 12-9. These film badges are collected by the UFTR staff for Radiation Control and evaluated monthly at R.S. Landauer, Jr., and Co. in Glenwood Illinois.

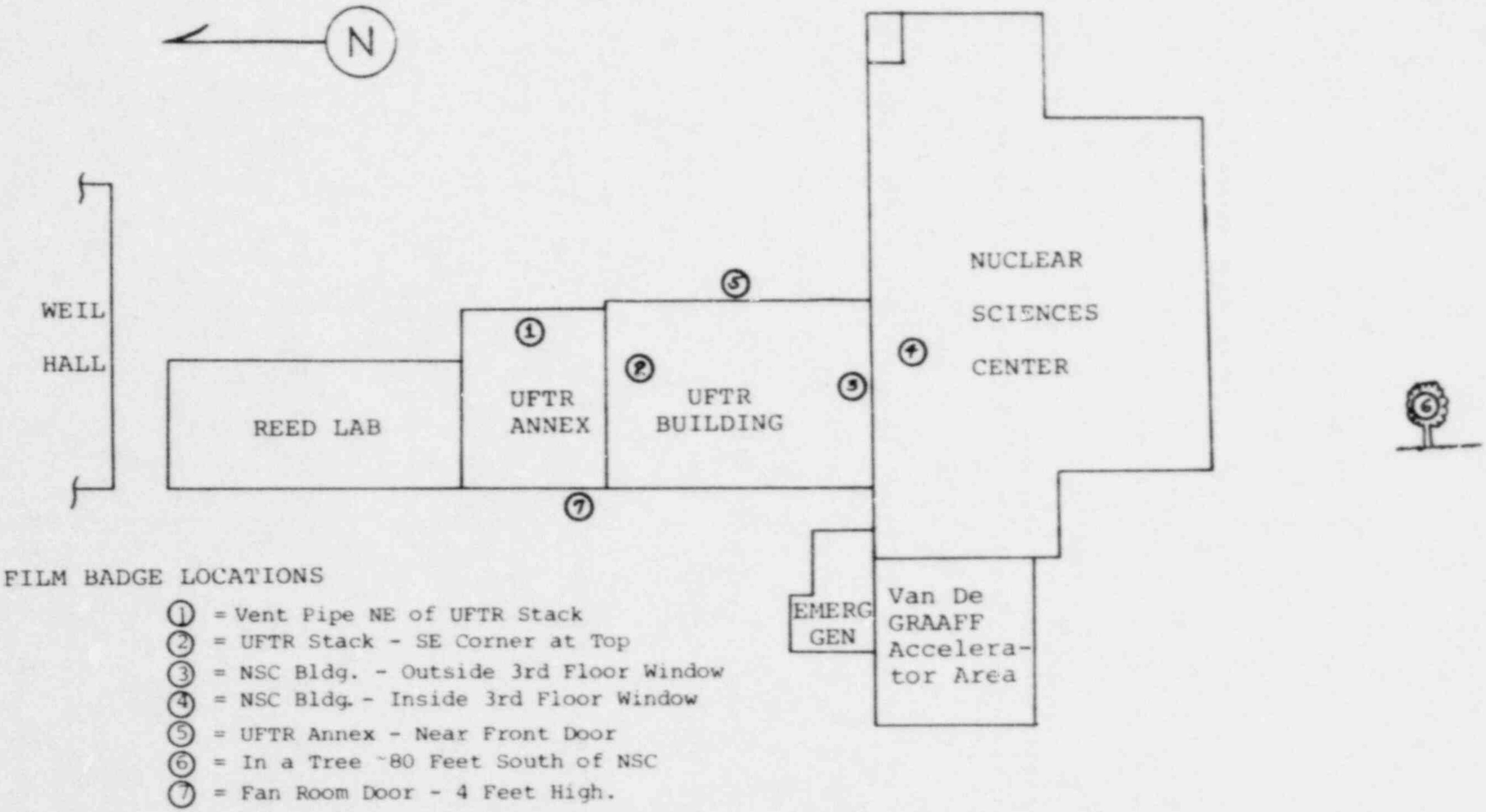


Figure 12-9. Location of Film Badges Used for Monitoring UFTR Site and Environs.

Typically these detection devices show no indications above background ($\sim$ 10 mrem/month) for the UFTR site. Therefore, Ar-41 discharges from the UFTR stack are not considered to present a danger to the general public.

12.4 Dose Assessment

As discussed in Section 12.1, surveys have shown that exposure levels associated with UFTR operation at the rated 100 KWth level are within the required limits. As indicated in Section 12.2, the radiation levels encountered during normal operation of the UFTR are low.

Radioactive effluents, excluding the Ar-41 problem, are essentially nonexistent at the 100 KWth rated power levels. Preliminary calculations indicate that the same situation would hold for operation at levels even up to 500 KWth. The Ar-41 problem has been discussed in Section 1'.3.

In general, the determination of dose levels expected to be received by the UFTR personnel, faculty and students working with the facility, and to the general public depends on the location of the person in question and the length of time the person spends in the area of the reactor. Radiation levels measured around the reactor at 100 KWth are indicated on the UFTR reactor cell sketch presented in Figure 12-2.

12.4.1 Dose Model for Gaseous Effluents

The Dose Model used for this Safety Analysis Report follows the approach outlined in Regulatory Guide 1.109 entitled, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50 Appendix I."(3) Since the only significant gaseous release from the UFTR during normal operations is Argon-41, the GASPAR code was used to calculate the doses from the routine releases assuming Ar-41 to be the only radioisotope released. Two different kinds of calculations were performed:

1. Dose to the hypothetically most exposed individual including highest gamma and beta air dose, and

2. Dose to the population.

12.4.1.1 Methodology for Calculating Maximum Individual and Air Doses. For dose calculations from a ground release, which is practically the case for the 30 foot UFTR stack, the semi-infinite cloud model is utilized. The applicable equation is derived based on the assumption that the energy generated from the decay of the radioisotopes in the air per unit mass is equal to the energy absorbed per unit mass (absorbed dose). To make the geometrical correction for the fact that a person standing on the ground is irradiated only from the air in 2π geometry, the expression is multiplied by 0.5 for the gamma dose.

The radioactive concentration of a vent-released radionuclide at a point x in pCi/m^3 according to the definition of the diffusion coefficient is:

$$\chi_{41}(x) = \kappa \cdot Q_{41} \cdot \chi/Q(x) \qquad (12\text{-}3)$$

where

κ = units conversion factor (3.17×10^4 pCi-yr/Ci-sec)

$\chi_{41}(x)$ = concentration of Ar-41 at position x (pCi/m^3)

Q_{41} = annual release of Ar-41 (Ci)

$\chi/Q(x)$ = diffusion coefficient at distance x (sec/m^3)

x = downwind distance from release point (m)

The corresponding annual gamma and beta air doses as a function of downwind distance are then given by the following two equations:

$$D_{41}^{\sigma}(x) = DF_{41}^{\sigma} \cdot \chi_{41}(x) \qquad (12\text{-}4)$$

$$D_{41}^{\beta}(x) = DF_{41}^{\beta} \cdot \chi_{41}(x) \qquad (12\text{-}5)$$

where

DF_{41}^{σ} = gamma dose conversion factor for Ar-41 (9.30×10^{-3} mrad-m^3/pCi-yr) (18)

DF_{41}^{β} = beta dose conversion factor for Ar-41 (2.69×10^{-3} mrad-m3/pCi-yr)(18)

$D_{41}^{\sigma}(x)$ = annual gamma air dose from Ar-41 (mrad/yr)

$D_{41}^{\beta}(x)$ = annual beta air dose from Ar-41 (mrad/yr)

The annual total body dose and the annual skin dose to an individual in the vicinity of the UFTR during normal operations in then calculated by using the following equations:

$$D_{\infty 41}^{T}(x) = S_F \cdot \chi_{41}(x) \quad DFB_{41} \qquad (12\text{-}6)$$

$$D_{\infty 41}^{S}(x) = 1.11 S_F \cdot \chi_{41}(x) \cdot DF_{41}^{\alpha} + \chi_{41}(x) \cdot DFS_{41} \qquad (12\text{-}7)$$

where

$D_{41}^{T}(x)$ = annual total body dose from Ar-41 (mrem/yr)

S_F = shielding factor due to the walls of the house where the individual lives; 0.7 is assumed based on recommendations from Reference 18.

DFB_{41} = dose conversion factor for total body dose from Ar-41 (8.84×10^{-3}mrem-m^3/pCi-yr)

$D_{41}^{S}(x)$ = Annual skin dose from Ar-41 (mrem/yr)

DFS_{41} = beta skin dose conversion for Ar-41 (2.69×10^{-3} mrem-m^3/pCi-yr)

12.4.1.2 Population Dose Methodology. For the population dose calculations, the surrounding area of the reactor is divided into subregions. The average dose to an individual present in each subregion is calculated for each of the organs, using the average diffusion coefficient for that subregion. The average dose to an individual is then multiplied by the total population living in the subregion. The total population dose is calculated by adding the population dose in every subregion. The resulting equation for the total population dose to an organ is given by:

$$D_j^P = 0.001 \sum_j P_d D_{jd}$$

where

D_j^P = total population dose to organ "j" (manrem/yr)

P_d = population in subregion, d

D_{jd} = dose to organ "j" of an individual living in the center of subregion d (mrem/yr)

12.4.2 Analysis of Past Effluent Releases from the UFTR.

As indicated in Section 11.3 and 11.5.2, the only radioisotope released in significant amounts from the UFTR is Ar-41. Since it is generated by activation of Ar-40 contained in the air used in ventilation of the Reactor Cell and dissolved in the primary coolant, the release is approximately proportional to the annual total energy generated. The total generated energy (Mw-Hrs) from September 1, 1972 up to August 31, 1978 as presented in Table 11-1 and the corresponding measured Ar-41 releases during the yearly reporting periods as presented in Table 11-4 are listed in Table 12-2. During the six-year period summarized by Table 12-2, the total energy generated by the UFTR was 139.92 Mw-Hr, and the average energy generated per year was 23.32 Mw-Hr.

Completely reliable data for Ar-41 releases is available only for the last two years indicated in Table 12-2. In addition, the releases recorded in this two year period are relatively high compared to the other reporting periods. For this two year period from September, 1976 to August, 1978, the yearly average release was 121.4 Ci. The average release per unit energy generated was 4.69 Ci/Mw-hr based on the last two years of release data. Since the facility design was not altered substantially during the six-year period of interest here, this average release per energy generated (4.69 Ci/Mw-Hr) was extrapolated to apply for all six years listed in Table 12-2. This value along with the average energy generated per year were combined to yield a very conservative value of 109.4 Ci/yr as the average yearly release of Ar-41 for the period September, 1972 to August, 1978. This value is very conservative versus the 40.46 Ci and 51.87 Ci of Ar-41 released for the two reporting years since August, 1976 as reported in Table 11-4. This value of 109.4 Ci/yr is the release selected for subsequent dose calculations.

TABLE 12-2

INTEGRATED HISTORY OF UFTR ARGON-41 RELEASES

Reporting Period	Total Energy Generated Mw-Hr	Ar-41 Released Ci
Sept.1, 1972-Aug.31, 1973	23.04	9.6
Sept.1, 1973-Aug.31, 1974	8.90	3.7
Sept.1, 1974-Aug.31, 1975	43.83	18.0
Sept.1, 1975-Aug.31, 1976	12.39	5.03
Sept.1, 1976-Aug.31, 1977	25.39	113.2
Sept.1, 1977-Aug.31, 1978	26.37	129.53
Sept.1, 1972-Aug.31, 1978	139.92	279.06

12.4.3 Population Distribution Around the UFTR.

As indicated in Section 2.13.6 of this Safety Analysis Report, the population distribution around the UFTR for these dose calculations was obtained from "Characteristics of Housing Units and Population by Blocks," which consists essentially of population data from the 1970 census. (8) The urban area of Gainesville extends further than 5 miles from the UFTR, but the population was conservatively assumed to be concentrated within a 5 mile radius around the UFTR. Table 2-1 and Figure 2-9 show the population distribution for each sector of the compass for circles with radii at 1 and 5 miles. The most significant changes to the Gainesville area population after 1980 have occurred in the "suburbs", outside the 5 mile area. Figure 2-9 is repeated here as Figure 12-10 for convenience.

12.4.4 Results of Dose Calculations.

12.4.4.1 Individual Dose Results. Two computer calculations were performed using the releases corresponding to the total 1978 release and the average yearly release for the period from September 1972 to August 1978. The points selected in both cases correspond to the two locations with the highest diffusion coefficients as noted from Figure 2-17 showing the annual average isopleths around the UFTR with Gainesville data repeated here as Figure 12-11for convenience, and from Figure 2-20 showing the annual average χ/Q values at special locations around the UFTR with Gainesville data repeated here as Figure 12-12 for convenience.

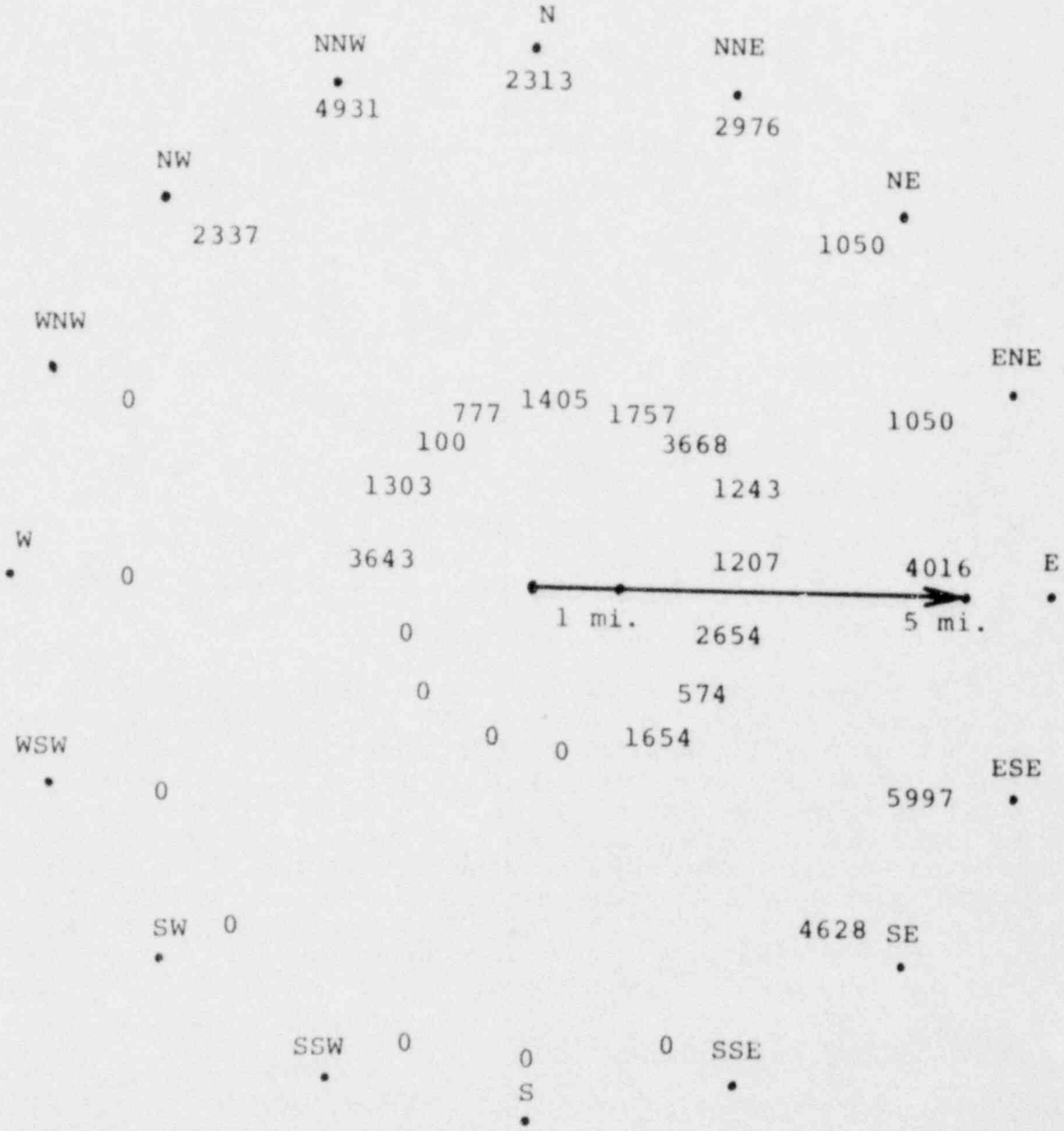


Figure 12-10. Population Distribution Around the UFTR By Sector Based Upon 1970 Census Data.

TABLE 12-3

RESULTS OF INDIVIDUAL DOSE CALCULATIONS AROUND THE UFTR

CASE*	X/Q (sec/m^3)	Ar-41 RELEASE (Ci)	BETA AIR DOSE (mrad/yr)	GAMMA AIR DOSE (mrad/yr)	WHOLE BODY DOSE (mrem/yr)	SKIN DOSE (mrem/yr)
CASE IA: 1978 Release	7.2 E-05	129.5	0.931	2.64	1.76	2.81
CASE IB: 1978 Release	7.1 E-05	129.5	0.917	2.60	1.73	2.77
CASE IIA: 1972-1978 Annual Release	7.2 E-05	109.4	0.786	2.23	1.48	2.37
CASE IIB: 1972-1978 Annual Release	7.1 E-05	109.4	0.775	2.20	1.47	2.34

*Cases I and IIA correspond to a point 0.10 miles West from UFTR,Cases IB and IIB correspond to a point 0.10 miles East from UFTR.

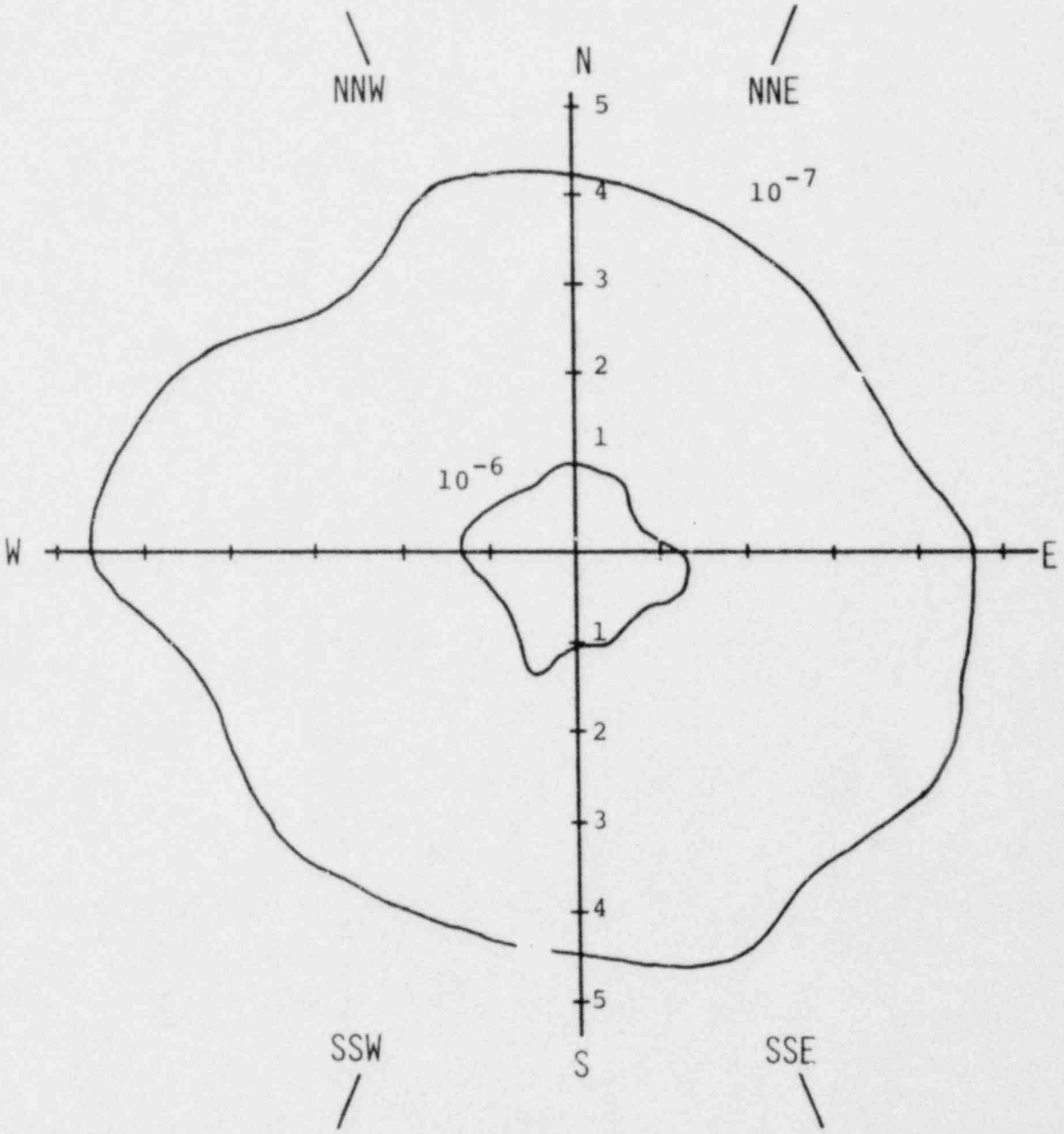


Figure 12-11. Annual Average Isopleths Obtained with Gainesville Data.

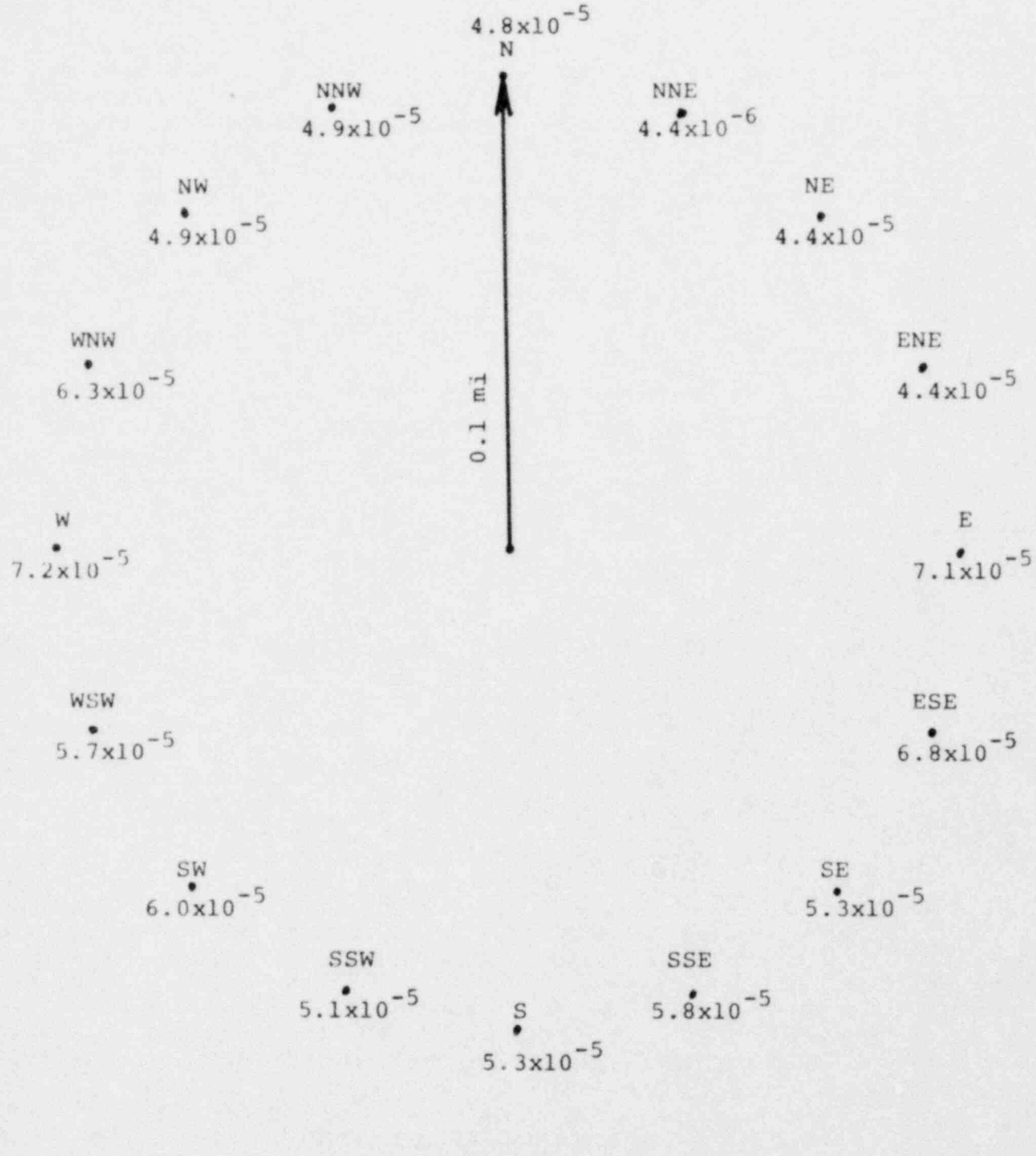


Figure 12-12. Directional Variation of Annual Average Diffusion Coefficients at 0.1 Mile Distance from UFTR.

The results calculated for the beta and gamma air doses, the whole body doses and the skin doses for both locations and both releases are presented in Table 12-3.

12.4.4.2 Population Dose Results. In calculating the population dose during normal operations of the UFTR, two cases were again considered. Case I corresponds to the total 1978 release and Case II corresponds to the calculated conservative average yearly release from September, 1972 to August 31, 1978. Because Argon is a noble gas, the only pathway which results in significant doses is direct irradiation. The results for the integrated yearly population dose for the UFTR are shown in Table 12-4.

TABLE 12-4

INTEGRATED YEARLY POPULATION DOSE FOR THE UFTR

CASE	A-41 RELEASE (Ci)	TOTAL BODY DOSE (manrem)	TOTAL SKIN DOSE (manrem)
I*	129.5	0.861	1.53
II**	109.4	0.727	1.29

*Corresponds to the total release for the September, 1977 - August, 1978 reporting year.

**Corresponds to conservatively averaged yearly release from September 1, 1972 to August 31, 1978.

12.4.5 Assessment of Dose Results for Normal UFTR Operation.

Appendix I to 10 CFR 50 and the Regulatory Guides 1.109 and 1.111 (References 18 and 16) are intended to state, clarify, and quantify the design objectives for commercial Nuclear Power Stations from the standpoint of their radiological impact in normal operations. In the evaluation of Appendix I for these stations, the highest exposed individual is assumed to be located outside the site boundary. The site boundary of the Nuclear Power Stations varies with each plant, but a value of 0.5 miles can be considered typical. This distance is five times the chosen distance from the UFTR vent to the hypothetically most exposed individual.

Because of the difference in site boundaries between the UFTR and typical power reactors, the conditions assumed for the evaluation of the radiological impact in normal operations for commercial Nuclear Power Stations are very different from the assumptions used in this work. However, in the absence of any applicable regulation for the radiological impact in normal operations for Test and Research Reactors, the comparison of the Appendix I Design Objectives for Gaseous Effluents with the actual doses calculated for the UFTR in normal operations from the Ar-41 releases for the highest exposed individual, is shown in Table 12.5. In general, the doses for the most exposed individual around the UFTR are much below those for a typical power reactor.

The population dose results for the UFTR are comparable to the dose resulting from commercial Nuclear Power Stations because, although the average individual doses are much smaller for the UFTR, the population concentration around the reactor is much larger than for a typical commercial Nuclear Power Station (Reference 42).

TABLE 12-5

DOSE COMPARISON BETWEEN APPENDIX I DESIGN OBJECTIVES AND CALCULATED UFTR RESULTS FOR THE MOST EXPOSED INDIVIDUAL AND HIGHEST AIR DOSES

	APPENDIX I DESIGN OBJECTIVE	UFTR HIGHEST CALCULATION
Gamma Dose in Air	10 mrad/yr	2.64 mrad/yr
Beta Dose in Air	20 mrad/yr	0.931 mrad/yr
Whole Body Dose	5 mrem/yr	1.76 mrem/yr
Skin Dose	15 mrem/yr	2.8 mrem/yr

12.5 Health Physics Program (34)

As indicated in Section 12.1, the increased utilization of ionizing radiation at the University of Florida led the administration to establish a University-wide Radiation Control Program in 1960. The primary purposes of this program are to assure the radiological safety of all University personnel and to make certain that ionizing radiation sources will be procured, used and stored in accordance with Federal and State regulations. To assume these ends, the Office of Administrative Affairs established the Radiation Control and Radiological Services Department under the Division of Environmental Health and Safety and headed by the Radiation Control Officer.

The Radiation Control Program provides a Radiation Control Committee and a Radiation Control Officer to carry out the responsibilities and necessary steps to insure radiological safety for the University and surrounding community. The Radiation Control Committee has designed procedures and policies in the form of a document entitled "Radiation Control Guide," (34) in an effort to provide investigators using ionizing radiations with guidelines necessary to maintain their facilities in a manner that assures radiological safety. These regulations and procedures are consistent with regulations of the Nuclear Regulatory Commission and the Florida State Board of Health; they are applicable to all facilities under the administration of the University of Florida including the UFTR facility.

The UFTR Reactor Safety Review (RSR) Subcommittee, a Subcommittee of the University Radiation Control Committee, reviews and audits reactor operations for safety, insuring radiological safety at the facility as determined by the Radiation Control Program. 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 by supervising the actions of the UFTR RSR Subcommittee.

The Radiation Control Committee is comprised of representatives from all departments involved in the use of ionizing radiations. The Radiation Control Officer is a qualified health physicist appointed by the Director of Environmental Health and Safety Division of the University of Florida. The delegation of responsibilities and duties of the Radiation Control Committee, and the Radiation Control Officer have been discussed in Section 12.1. The delegation of responsibilities and duties of the UFTR RSR Subcommittee are discussed in Section 13.1. The basic philosophy of the Health Physics Program is to assure the health and safety of all university personnel directly related to the UFTR. This basic ALARA philosophy is reflected in the UFTR Standard Operating Procedures and Technical Specifications.

APPENDIX 12A

UFTR CELL RADIATION LEVELS
MEASURED AT 100 KWth

12A.1 UFTR Cell Radiation Levels Measured at 100 Kwth

Table 12A-1 contains measured UFTR reactor cell radiation levels measured at 100 Kwth steady-state power levels. The data on radiation levels in this survey was taken using the instruments indicated in Table 12A-1: Cutie Pie 740, Victoreen-440 and the Bonnerball. The position numbers in Table 12A-1 correlate with the survey instrument locations shown on the Reactor Cell Floor Plan presented in Figure 12A-1. Data was taken with all shielding properly emplaced. In general, this survey data shows that the radiation levels in the UFTR cell during full power operation are very low. Such low radiation levels are sufficiently low to assure that occupational radiation exposures are as low as reasonably achievable for all personnel exposed to the radiation environment around the UFTR shields during full power operation.

TABLE 12A-1

UFTR Reactor Cell Radiation Levels Measured
at 100 KWth Steady-State Power Level
on October 9, 1980

Position Number*	Cutie Pie 740 (mrem/hr)	Victoreen-440 (mrem/hr)	Bonnerball (mrem/hr)
1	1.4	0.8	1.0
2	1.4	0.7	0.8
3	1.5	1.2	0.7
4	1.5	1.2	0.7
5	2.0	1.5	1.0
6	3.0	2.3	0.7
7	2.5	2.1	0.5
8	4.0	1.9	0.5
9	1.5	1.0	0.3
10	2.7	2.2	0.6
11	<0.5	0.3	0.3
12 Top of Bricks/ Bottom of Bricks	15/70	15/80	80/x
13	5	5	2.6
14 Top/Bottom	58/25	65/x	25/x
15	0.5	0.5	1.0
16	1.0	0.6	0.6
17	18	18	1.0
18	23	20	0.5
19	6	5	10
20	40	60	25
21	6	5	15
22	40	40	0.8
23	40	42	0.5
24	20	21	0.5
25	0.5	0.5	<1

*Numbers correlate to positions shown on the Reactor Cell Floor Plan in Figure 12A-1.

NOTE: The X on positions 12 and 14 indicates that the instrument would not fit in that location.

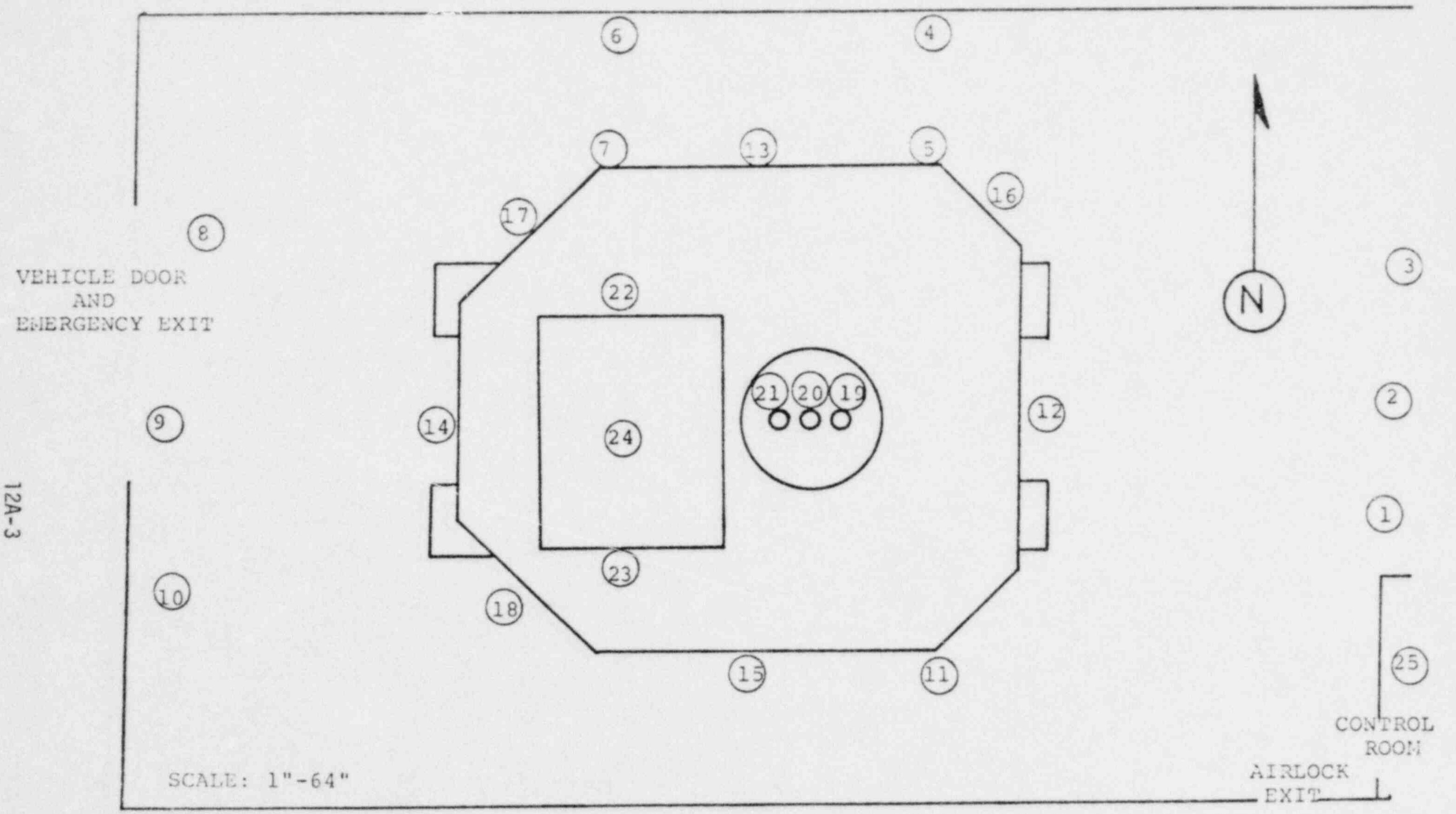


Figure 12A-1. Positions in UFTR Reactor Cell for Measurement of Radiation Levels at 100 KWth Steady-State Power Level with All Shielding Properly Emplaced. (Numbered Circle Indicate Points Where the Survey Was Taken to Correlate with Radiation Levels Recorded in Table 12-A-1.)

13. CONDUCT OF OPERATIONS

13.1 Organizational Structure of the Applicant

13.1.1 Management and Technical Support Organization

The UFTR is operated by the Department of Nuclear Engineering Sciences of the University of Florida for the purpose of instruction and research. The President of the University, the Dean of the College of Engineering, the Chairman of the Department of Nuclear Engineering Sciences, the Director of Nuclear Facilities and the Reactor Manager all have line responsibility for the administrative control of the reactor facility, for safeguarding the general public and facility personnel from radiation exposure and adhering to all requirements of the Facility License and the Technical Specifications. Line responsibility for administrative control of the UFTR is depicted in the flow diagram of Figure 13-1.

Direct supervision over the University of Florida, its policies and affairs, is vested with the Board of Regents. All University affairs are administered by the President with the advice and assistance of the Administrative Council. The Department of Nuclear Engineering is part of the College of Engineering and is under the supervision of the Dean of the College of Engineering.

There is no further need to consider the design and construction of the UFTR because this reactor has been safely operated at the University of Florida for over 20 years and basic UFTR design and construction is addressed in other Chapters of this report.

13.1.2 Operating Organization

13.1.2.1 Director of Nuclear Facilities and Reactor Manager. The Director of Nuclear Facilities and the Reactor Manager are in complete charge of the reactor facility. They are responsible for the safe operation of the reactor, the physical protection of the facility, the scheduling and supervision of experiments using the reactor, the control of the reactor fuel, the keeping of logs and records, and the maintenance of the physical condition of the facility. They are also responsible for liaison with the NRC and other regulatory bodies, and for coordinating the teaching and research programs within the facility.

The Director of Nuclear Facilities has line responsibility over the Reactor Manager and is directly responsible for the conduct of operations of the reactor facility. The Reactor Manager reports to the Director of Nuclear Facilities and has direct supervision over the operations, maintenance and record keeping of the UFTR. The Director of Nuclear Facilities and the Reactor Manager select operator-technicians and supervise their training. The Reactor Manager enforces operating procedures and regulations and has the power to authorize operations or experiments in accordance with facility regulations.

The Reactor Manager can make changes which do not alter the original intent of a procedure and/or establish new procedures that do not have

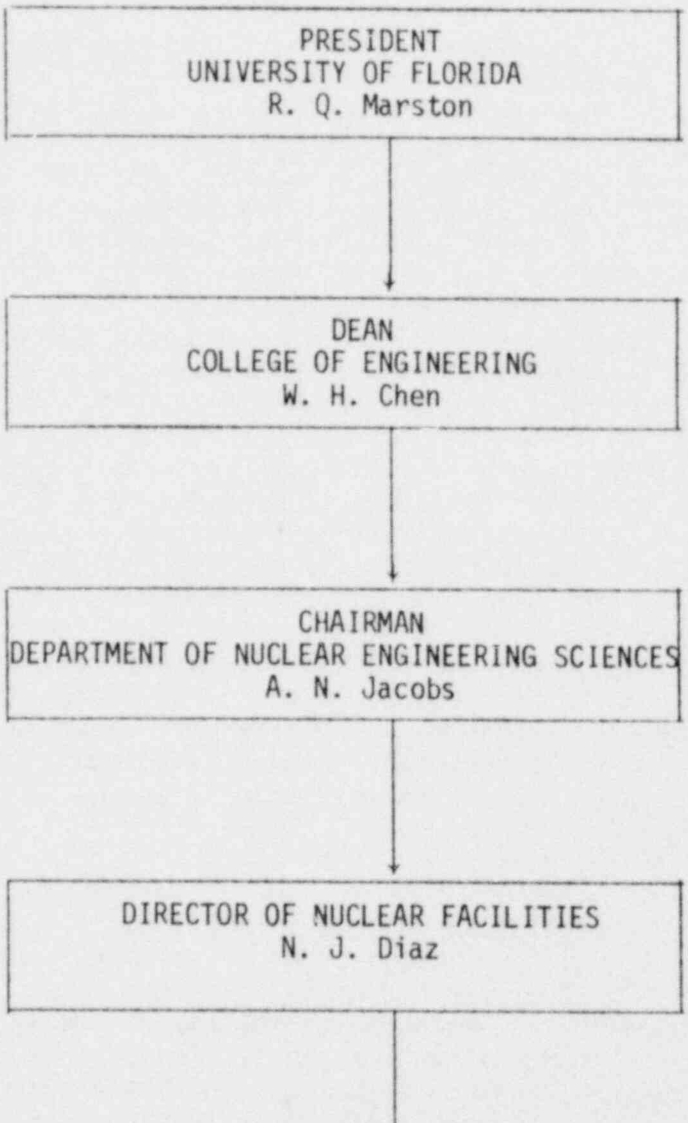


Figure 13-1. Line Responsibility Flow Diagram for Administrative Control of the UFTR.

safety significance, and submit these changes or procedures to the UFTR Reactor Safety Review Subcommittee for routine review. The Reactor Manager can also authorize repetitions of experiments previously approved by the UFTR Safety Review Subcommittee discussed in Section 13.1.2.2, and routine tests or operations which are necessary under normal operations and/or operations with no unreviewed safety implications. The Reactor Manager is advised by the Director of Nuclear Facilities, the UFTR Reactor Safety Review Subcommittee, the Radiation Control Office and the University Radiation Control Committee. The Reactor Manager is appointed by the Director of Nuclear Facilities and the Chairman of the Department of Nuclear Engineering Sciences, is formally a member of the Nuclear Engineering Sciences Faculty, is well qualified in experimental reactor physics and has qualifying experience in reactor operations.

13.1.2.2 UFTR Safety Review Subcommittee. The UFTR Reactor Safety Review Subcommittee is referred to in abbreviated form as the RSR Subcommittee or the RSRS. This Subcommittee is a part of, and answers to, the University Radiation Control Committee, which is referred to in abbreviated form as the URCC and provides its recommendations to the Director of Environmental Health and Safety. The Director of Nuclear Facilities and/or the Reactor Manager report any safety-related problems concerning the reactor to the UFTR RSR Subcommittee. After major modifications or repairs to the Safety or Control System approval of the RSRS is obtained prior to resuming operation of the UFTR facility. The RSR Subcommittee reports directly to the Chairman of the URCC. The purpose, rules and membership of the RSRS along with its basic purpose of reviewing and auditing UFTR operations for safety, are delineated in the following five paragraphs as presented in the Charter of the RSR Subcommittee included as Appendix 13A to this SAR. (43)

13.1.2.2.1 Purpose of the RSR Subcommittee. The purpose of the UFTR RSR Subcommittee is to provide an independent review and audit function of the safety aspects of reactor facility operations for the University of Florida Training Reactor.

13.1.2.2.2 Charter and Rules of the RSR Subcommittee. To assure the safety of reactor operations, the review and audit functions of the RSR Subcommittee are conducted in accordance with an established charter or directive with written rules of procedure for Subcommittee operation including provisions outlined as follows:

1. The UFTR RSR Subcommittee meets not less than once per calendar quarter, at intervals not to exceed 4 months, and more frequently as circumstances warrant, consistent with effective monitoring of facility activities. Records are kept of these meetings.

2. A quorum for RSR Subcommittee meetings consists of at least three members and at least three members must agree when voting, regardless of the number present.

3. Minutes are disseminated, reviewed, and approved in a timely manner.

13.1.2.2.3 Membership on the RSR Subcommittee. Membership requirements for the UFTR RSR Subcommittee are outlined below:

1. The UFTR RSR Subcommittee consists of five members including the Chairman of the Department of Nuclear Engineering Sciences, the Radiation Control Officer, the Reactor Manager, and two Technical Personnel (at least one from an outside department) familiar with the operation of reactors and with the design of the UFTR. These two persons are recommended for appointment to the Chairman of the URCC by the Chairman of the Department of Nuclear Engineering Sciences. Any member may designate a duly qualified representative to act in his absence.

2. The Executive Committee consists of the Reactor Manager, the Radiation Control Officer, and the Chairman of the RSR Subcommittee.

3. The Chairman of the UFTR Subcommittee is a member of the URCC and is selected by the Chairman of the URCC.

4. Appointed members to the Subcommittee are reviewed and new appointments made by October 1 of each year.

13.1.2.2.4 Review Function of the RSR Subcommittee. To meet the requirements of its review function, the UFTR RSR Subcommittee reviews the items outlined in the following paragraphs:

1. Proposed changes in equipment, systems, test, experiments, or procedures;

2. All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance;

3. All new experiments or classes of experiments that could affect the safety of the reactor or result in the release of radioactivity;

4. Proposed changes in UFTR technical specifications, UFTR license or RSR Subcommittee charter;

5. Violations of UFTR technical specifications, UFTR license or RSR Subcommittee charter and violations of internal procedures or instructions having safety significance;

6. Deficiencies having safety significance; recommendations are made for corrective actions;

7. Reportable occurrences; the RSR Subcommittee recommends corrective actions;

8. Audit reports.

13.1.2.2.5 Audit Function of the RSR Subcommittee. The audit function of the RSR Subcommittee includes selective (but comprehensive) examination of operating records, logs, and other documents. Where necessary, discussions with cognizant personnel also take place. The individual immediately responsible for an area does not perform the audit in that area. The following paragraphs describe items that are audited:

1. Facility operations are audited for conformance to the technical specifications and applicable license or charter conditions, at least once per calendar year (interval between audits not to exceed 15 months).

2. The retraining and requalification program for the operating staff is audited at least once every other calendar year (interval between audits not to exceed 30 months).

3. The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety are reviewed at least once per calendar year (interval between audits not to exceed 15 months).

4. The reactor facility Emergency Plan and the implementing procedures are reviewed at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies uncovered that affect reactor safety are immediately reported to the Chairman of the University Radiation Control Committee. A written report of the findings of the audit is submitted to the reactor management and the review and audit group members prior to March 1 of the year following the calendar year under review.

13.1.2.3 Radiation Safety Organization. The radiation safety organization at the University of Florida is directed and overseen by the University Radiation Control Committee (URCC). The Committee is appointed by the President and includes, at present, professors from the Departments of Biological Sciences, Radiological Health, Nuclear Engineering Sciences and Environmental Engineering Sciences, as well as the Radiation Control Officer (ex-officio member); the members of this committee indicated in the approved January, 1979 issue of the Radiation Control Guide are listed with their affiliations on Table 13-1 to demonstrate the breadth of interests and expertise represented on this committee. The URC Committee is responsible for advising the President on all matters related to radiation safety. The primary purpose of the Committee is to review and grant permission for, or disapprove and refuse permission for, the use of radioactive isotopes or any other sources of ionizing radiation at the University of Florida and to insure the health and safety of reactor personnel and the general public.

13.2 Training

13.2.1 Plant Staff Training Program

Training of reactor operators at the UFTR is done on an individual basis to fit the trainees' needs; schedules are arranged in a flexible manner in order to maximize the availability of the reactor as a research and teaching tool. Training procedures and requirements are determined by the Director of Nuclear Facilities/Reactor Manager and are supervised

TABLE 13-1

UNIVERSITY OF FLORIDA
RADIATION CONTROL COMMITTEE MEMBERSHIP*

Member	Affiliation
Dr. Charles E. Roessler, Chairman	Environmental Engineering Sciences
Dr. David S. Anthony	Botany Laboratory
Dr. William K. Collett	Dentistry
Dr. G. Ronald Dalton	Nuclear Engineering Sciences
Dr. F. Eugene Dunnam	Physics
Dr. Robert J. Hanrahan	Chemistry (Nuclear)
Dr. Walter Mauderli	Radiology
Dr. Richard Shaara	Director, Student Health Services
Dr. Crispin P. Spencer	Veterinary Medicine
Mr. Thomas Bauer	Radiation Control Officer

*Membership as of September 29, 1980.

by a licensed reactor operator at all times. The trainee will receive academic and operational training, to be adequately prepared for the written and practical examinations planned by the Director of Nuclear Facilities/Reactor Manager and by the NRC. These examinations are designed to fulfill the requirements established by the NRC in compliance with 10 CFR 50 and 10 CFR 55.

13.2.2 Replacement and Retraining

In the academic environment of the UFTR, reactor startups, shutdowns, normal and abnormal operations are routinely encountered by licensed senior reactor operators and reactor operators. The reactor staff routinely meets every week and discusses the reactor status quo, maintenance and tests performed or to be performed, as well as any other technical or administrative subjects considered to be pertinent to the safe operation of the UFTR. Written monthly reports summarize the reactor operations, maintenance, tests and calibration. Every licensed operator or senior operator reviews this monthly report and it is discussed in staff meetings. Changes in procedures, technical specifications and regulations are reviewed and discussed before implementation. The reactor staff participates as instructors and/or students in formal university courses involving the training of students or reactor operator training conducted for the UFTR or other facilities. A training program for the periodic requalification of UFTR operators is conducted in accordance with NRC requirements as delineated in the UFTR "Operator Requalification and Recertification Plan for July 1979 through June 1981." The requalification for the UFTR personnel meets or exceeds the requirements established by 10 CFR 55 Appendix A and draft ANSI/ANS-15.4 standards dated June, 1977 entitled "Standards for Selection and Training of Personnel for Research Reactors."

Responsibility for the administration of the program rests with the Director of Nuclear Facilities of the Department of Nuclear Engineering Sciences and his designated representative.

All licensed operators are required to participate in all phases of this program except where specifically exempted. Persons in training for an operator's license also participate in the requalification program. An operator receiving a license during a requalification period is required to complete only those portions occurring after the effective date of the license received.

The requalification training program in force at the UFTR consists of eight (8) component areas described in the following sections. The requirements that must be met in order to complete the requalification program successfully are delineated in these sections.

13.2.2.1 Requalification Schedule. The UFTR requalification program is conducted over a period not to exceed two years and is then followed by successive two-year programs. To assure that the program is most effective, the various requirements are executed according to the time schedules outlined in this guide.

13.2.2.2 Lectures, Reviews and Examinations. The requalification program is divided into the group of topics listed in Table 13-2 for which preplanned training or preparation is scheduled. The schedule is set up so that the entire program covering the topics listed in Table 13-2 is completed over the two year period.

TABLE 13-2

TOPICS FOR UFTR REQUALIFICATION TRAINING

I. Nuclear Theory and Principles of Operation

II. Design and Operating Characteristics

III. Instrumentation and Control Systems

IV. Reactor Protection System

V. Normal, Abnormal and Emergency Procedures (one per year minimum, independent of emergency drills)

VI. Radiation Control and Safety

VII. Technical Specifications and Applicable Portions of Title 10, Code of Federal Regulations

An examination is administered at the end of each segment listed in Table 13-2, no later than two weeks after the lecture or review session. For designated cases, a final examination covering all topics is substituted for individual examinations. Results of the certified individual's evaluation from the examinations and from the on-the-job training described under Section 13.2.2.6 is used to determine the operator's proficiency, weakness or deficiency.

A special training session is held prior to any refueling operation and/or fuel handling operation. The required operations are discussed/practiced and procedures are reviewed to assure proficiency of all personnel involved. Emergency actions are also reviewed.

Any changes in procedures, technical specifications, regulations, as well as any change with safety significance to the facility are reviewed by every licensed operator. Furthermore, activities in the reactor, including modification, maintenance, results of calibrations and tests, as well as any procedural changes, are summarized in a written report which is distributed to all licensed reactor operators and discussed as needed.

Various documents, letters and memos are maintained in the Required Reading List prior to permanent filing. Each operator is responsible for reviewing the list periodically in a timely manner to remain current with the information contained in the Required Reading List.

A yearly review of facility operations, maintenance, modifications, etc., is conducted by the Director of Nuclear Facilities or the Reactor Manager with the operating staff using the UFTR Annual Report as a base for the review.

13.2.2.3 Requalification Operations and Checkouts. Over the two year requalification period, each certified individual performs at least ten reactivity control manipulations in any combination of reactor startups, shutdowns, or significant reactivity changes. To insure operator proficiency over a range of ordinary operations, the following schedule of operations and checkouts is maintained by all licensed operators when the reactor is operable.

1. Each licensed operator performs at least one reactor startup quarterly at intervals not to exceed four months.

2. Each licensed operator performs at least one daily checkout quarterly at intervals not to exceed four months.

3. Each licensed operator performs at least one weekly checkout semi-annually at intervals not to exceed eight months.

It is the responsibility of each operator to insure that these requirements on performance of reactivity control manipulations are met and logged in the operator's Requalification folder. Each operator also logs monthly operating hours in the same folder.

13.2.2.4 Emergency Drills. Emergency drills are held quarterly. At least once per year these drills involve the participation of the University Police Department, the Gainesville Fire Department, and other emergency assistance teams as appropriate for the drill in question. Each operator is required to participate in two emergency drills per year at intervals not to exceed eight months. A review of the drill and applicable emergency procedures is performed with all certified individuals within seven days after completion of the drill.

13.2.2.5 Absence from Authorized Activities. An operator who has not been actively performing certified functions for a period in excess of four months is required to demonstrate to the Reactor Manager or duly authorized representative that the knowledge and understanding of the operation and administration of the facility are satisfactory before returning the operator to certified duties. An individual is required to demonstrate satisfactory knowledge and understanding of the facility operation and administration through an interview and evaluation or a written, oral or operational examination or a suitable combination thereof. Any deficiencies uncovered are corrected before the individual resumes authorized functions.

13.2.2.6. Evaluation of Operators.

13.2.2.6.1 Biennial Evaluation. An in-depth evaluation of the operating performance of each licensed operator is performed and documented biennially and/or prior to their recertification anniversary to insure that they have the knowledge, competence, and dexterity to operate the reactor safely and to take appropriate actions in response to abnormal situations that may arise.

The evaluation includes results from the examinations, the annual on-the-job evaluation of operational proficiency (as delineated under Paragraph 2 of this section)and any other available indications of the operator's capability to discharge the duties in a safe and competent manner.

13.2.2.6.2 Annual On-the-Job Training. Each licensed Reactor Operator and Senior Reactor Operator is required to demonstrate satisfactory understanding of the operation of the facility systems, operating procedures and facility procedure license changes during an annual walk-through examination administered by a designated Senior Reactor Operator.

13.2.2.6.3. Grade Requirements. All operators are required to complete each examination satisfactorily according to the following requirements:

1. A grade higher than 80% requires no additional training.

2. A grade in the range of 65% to 79% requires additional training in those areas or topics where weaknesses or deficiencies are indicated. This training is required to be completed within 60 days from the date the examination was administered.

Additional appropriate training requirements in the form of formal lectures, tutoring, self-study or on-the-job training are based on the results of the examinations conducted.

3. With a grade of less than 65% the individual is placed in an accelerated retraining program in those areas where weaknesses or deficiencies are indicated.

13.2.2.6.4. Accelerated Training. Accelerated training programs are completed within four months following the grading of the examination. Furthermore, within one month after the grading of the examination, an evaluation is made by the Reactor Manager or a designated representative to determine if the deficiencies uncovered warrant withdrawal of the individual's certification pending completion of the accelerated training program. The evaluation considers the individual's past performance record, the supervisor's evaluation and past test scores as well as current deficiencies. An oral exam may also be given to aid in the evaluation. Regardless of the score, if the individual's test indicates a deficiency in a critical area that affects safety, a training program shall be administered to promptly correct the deficiency.

13.2.2.6.5. Additional Training Requirements and Evaluations. Additional training is provided whenever needed to correct weaknesses or deficiencies uncovered. Such additional training is completed prior to the conclusions of the specific requalification program or application for renewal of operator's license, whichever occurs first.

An evaluation is made of an operator any any time his/her physical or mental condition appears impaired in a manner that his/her performance of duties as an operator appear to be affected. Any exemplary performances or additional duties performed by an operator are noted in his/her Requalification folder to aid later evaluations.

13.2.2.7. Requalification Records. Records are kept to assure that all the requirements of the UFTR "Operator Requalification and Recertification" Plan are met.

Each operator has an individual folder containing signature blocks for lectures attended, prepared or assigned self-study sessions, reactivity manipulations performed, weekly and daily checkouts performed, and quarterly drills participated in by the operator. The folder also contains copies of written examinations administered, the answers given by the operator, results of any evaluations and documentation of any additional training administered in areas in which an operator has exhibited deficiencies. The performance of, or participation in, special activities such as fuel handling by the individual operator, is also logged in the Requalification folder.

Pertinent documents and records pertaining to the Requalification program are maintained at the UFTR as part of the facility records for a period of five years.

13.2.2.8 Requalification Document Review. The individual Requalification folders are reviewed on a semi-annual basis by a designated Senior Reactor Operator as noted by the inclusion of the SROs dated signature. Any deficiencies noted during the review are brought to the attention of the Director of Nuclear Facilities or the Reactor Manager who then insures that appropriate corrective action is taken.

13.3 Emergency Planning

13.3.1 Emergency Organization

The Site Emergency Plan for the UFTR facility is described in the "Emergency Planning for the UFTR" guidelines and in the facility Standard Operating Procedures which detail the responsibilities, procedures, and actions to be taken by all personnel in the event of emergency conditions which could endanger the health and safety of the facility personnel and/or the general public.

The Director of Nuclear Facilities (or his duly authorized representative such as the Reactor Manager) has overall responsibility for the handling of emergency situations, including coordination with Law Enforcement, Disaster Preparedness, Local and State Health Agencies and the Nuclear Regulatory Commission. The Radiation Control Officer assists the Director of Nuclear Facilities and/or the Reactor Manager in all matters which concern the health and safety of the public during any emergency.

The UFTR Emergency Planning is being submitted as a separate document from the SAR to comply with all the requirements specified in the November 6, 1980 ruling amending 10 CFR 50 and 70 in connection with upgrading emergency preparedness regulations. The UFTR Emergency Planning will follow Regulatory Guide 2.6 as guidance for compliance with 10 CFR 50.54(q) and 10 CFR 50 Appendix E.

The documents outlining the UFTR Emergency Planning include all applicable procedures to effectively conduct the activities required by the plan.

13.4 Review and Audit

Review and audit functions for the UFTR facility operations are conducted to determine if the facility is being operated safely and within the terms of the license. The review and audit functions are performed by the UFTR Reactor Safety Review Subcommittee. An intensive in-depth review of facility operations is made at least annually. One of the specific concerns addressed in such reviews is the emergency planning in effect at the UFTR facility.

Review and audit of radiological safety procedures and other emergency related procedures are also performed by the University Radiation Control Committee (URCC) and the Radiation Control Officer, an ex-officio member of the UFTR Reactor Safety Review Sbucommittee.

13.4.1 Authorization of Experiments and Operations

All experiments and operations of the UFTR, including maintenance and repairs, are required to have the prior approval of the Director of Nuclear Facilities, the Reactor Manager or an authorized representative of either.

All experiments proposed for the reactor are classified into one of four categories designated by Class 1 through Class 4. The basis for the classification of experiments is the degree of novelty and potential for presenting a hazard to UFTR personnel, the public or the reactor.

Class 1 experiments include routine experiments such as gold foil irradiation. Class 1 experiments are readily approved by the Reactor Manager and the Radiation Control Officer.

Class 2 experiments include 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 hazard to the reactor, the UFTR personnel or the public. Class 2 experiments are also approved by the Reactor Manager and the Radiation Control Officer.

Class 3 experiments consist of those which pose significant questions regarding the safety of the reactor, the UFTR personnel or the public. These experiments are approved by the UFTR Reactor Safety Review Subcommittee, which recommends procedures and/or devices to minimize hazards.

Class 4 experiments comprise all experiments which have a significant potential for hazard to personnel, the public or the reactor. These experiments are approved by the UFTR Reactor Safety Review Subcommittee for performance under the direct supervision of the Reactor Manager or a duly authorized representative. A detailed description of the expected behavior of the facility during the proposed experiment along with accompanying emergency procedures specific to the experiment are required to be approved by the UFTR Reactor Safety Review Subcommittee before permission to perform the experiment is granted. Experimental procedures, criteria for evaluation and required documentation are specified in UFTR SOP-A-5. Experimental limitations are included in the Technical Specifications for the UFTR.

Startup of the UFTR after non-routine maintenance, or after repairs, an accident, or a real scram (as distinguished from operator mistake, or a scram from electrical noise or power interruption) is not permitted until first authorized by the Reactor Manager or his authorized representative.

13.5 Plant Procedures

This section describes the procedures pertinent to normal operation and administration of the UFTR facility including the performance of experiments and modifications, repairs, and tests. The Reactor Manager is responsible for ensuring compliance with the established controls.

The Reactor Manager is responsible for the preparation of detailed written procedures for normal and emergency operations. These procedures are approved by the UFTR Reactor Safety Review Subcommittee before implementation for any procedure with safety significance, and after implementation for minor changes in established procedures (that do not change the original intent) or new procedures with no safety significance (which are approved and implemented by the Reactor Manager). Procedures include a startup check list, shutdown check list, operating manual, procedures and check lists for special operations such as fuel transfers, procedures for requesting irradiations, and procedures for transfer of irradiated materials.

13.5.1 Administrative Procedures

13.5.1.1 Access and Key Control. Outside doors of the reactor facility are normally locked. The rear freight-door exit is only used for emergencies, refueling and other unusual situations. The Operator-in-Charge at the UFTR controls entry to the reactor area and can forbid entry at any time. The Operator-in-Charge requires personnel entering the Reactor Cell for work-related duties to either wear a personal dosimeter, or be escorted by a responsible person wearing a dosimeter. Visitors are escorted and must sign in and out in the log book provided. If a person dose is estimated to be less than 1/10 of the allowed occupational exposure, no badging is necessary.

When the reactor is not operating, unsupervised access is permitted only to persons holding authorization from the Reactor Administration. Entry requirements are covered under the Physical Security Plan of the Facility. Emergency instructions require that a senior member of the reactor staff or the Radiation Control Office is contacted prior to entry by police or fire personnel in case of an accident.

Key control for the Reactor Cell is summarized as follows:

1. Console keys are in the custody of a Licensed Reactor Operator.

2. Special keys are used to lock the shield plugs, the reactor crane switch and other reactor devices. These keys are kept in the Reactor Key Cabinet, which is in the Reactor Security Area and access to which is restricted to Licensed Reactor Operators.

3. Facility door keys and Security System keys are issued to persons designated by the Reactor Manager and/or the Director of Nuclear Facilities. Further details on key control are not included in this Safety Analysis Report. A detailed description of the UFTR facility key control measures are contained in the Security Plan for the UFTR submitted separately and withheld from public disclosure pursuant to 10 CFR 2.790(d).

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Routine Operation and Records. Manipulation of the UFTR reactor controls is permitted by a UFTR-Licensed Reactor Operator or by a non-licensed person under the direct observation and supervision of a licensed Reactor Operator. A Senior Reactor Operator is on call at all times that the facility is operating and a second person, duly qualified as such, is present either in the Reactor Cell or within the UFTR Facility Complex. An Operator-in-Charge is designated for each reactor operation. The Operator-in-Charge (OIC) is responsible for ensuring that the following requirements are met during the reactor operation.

1. The OIC is in a position to operate the controls of the reactor. This requirement means that the OIC cannot leave the Reactor Cell, and is normally at or within sight of the controls and instruments at the console.

2. The correct startup check and shutdown check procedures are followed and log sheets are filled out for all operations.

3. Any proposed experiment is correctly authorized and any requirements noted have been complied with.

4. The experimenter's proposed procedure conforms to University Radiation Control Committee recommended practice and the experimenter has a valid radioisotope license if the transfer of radioisotopes is outside the control of the Radiation Control Office.

5. All samples removed from the reactor are monitored, their activity levels recorded, and any necessary temporary access barriers or shielding are erected.

6. The experimenter and the Reactor Manager are informed in case of any unusual or unexpected occurrence, apparent equipment or instrument failure or other malfunction.

7. The Radiation Protection Specialists have been notified if any experiment predicted to involve high radiation levels is to be performed to assure that the Health Physicist is present as necessary.

The Operator-in-Charge normally satisfied requirements (3), (4) and (7) directly by ensuring that the Reactor Manager has correctly approved the proposed experiment and associated schedule. All members of the reactor staff are expected to be familiar with basic radiation safety procedures so that adequate safety is ensured even in the absence of Radiation Control Specialists.

The Reactor Startup Procedure (UFTR SOP-A.2) ensures that the reactor and experimental configuration are correct, and removable shielding is in place or personnel otherwise protected, the instruments are calibrated and functioning, the scram and interlock circuits are functioning and scram points properly set, and the facility is otherwise in proper condition for operation.

A daily log of UFTR operations with information related to pre-operational checks and facility usage is maintained in a bound notebook in the Reactor Control Room. The log includes the name of the operator-in-charge, the experimental configuration, special instructions, periodic readings of instruments and control rod positions, results of tests and inspections, maintenance and change records, methods and reasons for shutdown, and any other notations the Reactor Operator deems appropriate. The log book, checklists, and other pertinent records are filed and audited annually by the UFTR RSR Subcommittee.

The Operating Manual includes standard operating procedures, experimental procedures and limitations, requirements for periodic checks and maintenance, radiological safety procedures, emergency procedures, technical specifications and license limitations.

13.5.2.2 Routine Tests, Maintenance, and Monitoring. The Reactor Manager has set up a program for regular testing of all safety equipment, procedures and certain reactor components.

In addition to the startup checks of instruments, scrams and interlocks, periodic checks and maintenance are performed on a daily, weekly, quarterly, semi-annual and annual schedule.

Weekly and Daily Pre-Operational checks are required by UFTR SOP-A.1, "Pre-Operationsl Checks." The pre-operational checks are required by UFTR SOP-A.1 are divided into two (2) parts as described in this paragraph. Part I addresses Weekly Pre-Operational Checks, and Part II addresses Dialy Pre-Operational Checks and associated checklists. The Pre-Operational Checks are performed by a licensed reactor operator or trainees under his direct supervision. The results of Part I and Part II checks are filed at the UFTR Facility.

Any malfunction of the safety-related system for the UFTR is sufficient cause for stopping reactor operation until the malfunction is corrected. Written instructions for calibrations, tests and maintenance or repairs for the Reactor Safety and Control Systems are available for the UFTR as part of the SOP's. The results of all of the above periodic tests, checks, maintenance and monitoring are recorded in the Maintenance Log, the Operation Log and the Maintenance card file.

The University Radiation Control Committee performs routine announced and unannounced surveys of the reactor and the reactor area, especially during reactor operation, to check radiation levels. The results of such monitoring are recorded and maintained in the log book. The detection of any significant or abnormal radiation outside the reactor facility requires immediate investigation and subsequent corrective action including procedural changes, addition of shielding, or other action as deemed necessary to alleviate the problem.

A film badge service is provided as part of the personnel monitoring program. These are supplemented in the reactor area by pocket dosimeters which are also used for occasional visitors. The Radiation Control Officer is in charge of badging and associated records.

13.5.2.2.1 Daily Pre-Operational Checks. The Daily Pre-Operational Checks are started and satisfactorily completed within 6 hours prior to reactor startup or if the reactor has been shut down less than 6 hours and no known condition exists that would prevent successful completion of a daily check. For these purposes, reactor shutdown means that the reactor had been critical with proper functioning of all instruments and components and that a shutdown had been effected under normal conditions. The scope and detail of the Daily Pre-Operational checks required by UFTR SOP-A.1, Part II, "Daily Pre-Operational Checks" is indicated in the Daily Pre-Operational Checklist presented in Figure 13-2. The requirements of the Daily Pre-Operational Checks are summarized below:

1. The console and equipment power supply are checked to insure all items in the annunciator light panel, the radiation monitoring console, the auxiliary alarm panel, the recorders, and other systems related to all the operational equipment, are functioning correctly.

2. The proper functioning of the shield tank recirculating system, the air particulate detector, portal monitor, and primary resistivity bridge is checked.

3. The c[illegible]ion of all the nuclear instrumentation is checked.

4. The proper functioning of the control blade interlock system and the fast period interlock system is checked.

5. The proper functioning of the reactor safety system including the reactor trip systems and the annunciator alarms systems are checked.

13.5.2.2.2 Weekly Pre-Operational Checks. The Weekly Pre-Operational Checks are routinely performed on the first day of the working week when the reactor is operable. During extended shutdown periods for administrative purposes, maintenance or modifications, the weekly Pre-Operational Checks are performed each week on the operable systems. UFTR-SOP-A.1 - Part I "Weekly Pre-Operational Checks," is required to have been completed satisfactorily within seven (7) days prior to reactor startup. The scope and details of the Weekly Pre-Operational Check are summarized in the Weekly Pre-Operational Checklist presented in Figure 13-3 while the general requirements of the Weekly Pre-Operational Checks are summarized below:

1. The operability of the area radiation monitors and continuous air particulate radioactivity monitor is checked and they are internally calibrated; the high-level alarms are tested with all personnel in the vicinity notified before the alarms are tested.

UFTR S.O.P. A.1 PART II, DAILY PRE-OPERATIONAL CHECKLIST

1. CONSOLE & EQUIPMENT POWER

1.1 Elec Power Breakers (2)(on) ____
1.2 Well Pump Breaker (on) ____
1.3 Console Power (on)(grn lt) ____
1.4 Light Bulb Check:
1.4.1 Right Ann Panel (on) ____
1.4.2 Left Ann Panel (on) ____
1.4.3 Mode Switch (MANUAL) ____
1.4.4 DOWN Lights (on) ____
1.5 Recorders:
1.5.1 Log/Lin Rcdr (operating) ____
1.5.2 Area Mon (3), Stack, APD ____
1.5.3 Temp Rcdr Chart Supply ____
1.6 Radiation Monitor Console:
1.6.1 Live Zero's (4 meters) ____
1.6.2 NO FAIL Lights (4)(on) ____
1.6.3 FAILURE & TRIP 2 Sw (off), ALARM Switch (alarm) ____
1.6.4 Alarm Test, Push Button ____
1.6.5 24 vdc Power Supplies (2) ____
1.7 Auxiliary Alarm Panel:
1.7.1 Green Lights (4)(on) ____
1.8 Dump Valve:
1.8.1 Reset, (DV light out) ____
1.8.2 Secure Console Key ____
1.9 Operational Equipment:
1.9.1 Diluting Fan (>425 rpm) * ____
1.9.2 Core Vent Fan ____
1.9.3 Demineralizer Pump ____
1.9.4 PC Pump (DM 1/2 out) gpm * ____
1.9.5 Shield Water Pump ____
1.10 Shield Tank Recirc System:
1.10.1 Leaks & Noise ____
1.10.2 Flow ____
1.10.3 Valve Alignment ____
1.11 Air Particulate Det:
1.11.1 Operational ____
1.1 .1 Air Flow (litres/min) * ____
1.11.3 Range Switch at * X____
1.12 Portal Monitor (Oper) ____
1.13 Primary Resistivity Bridge:
1.13.1 Power & Grn Lights (on) ____
1.13.2 INLET, @ MΩ-cm * ____
1.13.3 OUTLET, @ MΩ-cm * ____
1.13.4 Setpoint > 1 MΩ-cm ____

2. OPERATIONAL

2.1 Key Switch to RESET, Hold: ____
2.1.1 ON Lights (out) ____
2.2 Key Switch to OPERATE:
2.2.1 SCRAMS (left panel)(out) ____
2.2.2 ON Lights (o) ____
2.2.3 Red Rotating Beacon (on) ____
2.2.4 Temperature Rcdr (oper) ____

3. BLADE INTERLOCK CHECKS

3.1 All Cal & Trip Test Switches to OPERATE or OFF ____
3.2 SOURCE Interlock Lt Out ____ or On ____
3.2.1 If "On," No Blade W/D ____
3.2.2 Source Counts (cps) * ____
3.2.3 Source INTLK (clear) ____
3.3 Cal & Trip Test Switches
3.3.1 SAF-1/LOG Switch to ZERO, INTLK (on) ____
No Blade can be W/D ____
Return Switch to OPERATE ____
3.3.2 Period Cal Sw, Hold, INTLK (on), Release Switch ____
3.3.3 SAF 2 Cal Switch to Zero, INTLK Lt (on), Release ____
3.3.4 SAF 1 Trip Test Sw, No Blade can be W/D, Rtn OFF ____
3.4 10 Second Period Interlock:
3.4.1 Pd Trip Test to 10 Sec, Fast Pd Light (on) ____
3.4.2 No Blade W/D, Rtn to OFF ____
3.5 Multiple Blade W/D Intlk:
3.5.1 All Comb, No Blade W/D ____

4. NUCLEAR INST. & CALIB CHECK

4.1 PERIOD CAL Sw, Hold, Meter to 3 Seconds, Release * ____
4.2 SAF-1/LOG Cal Switch:
4.2.1 To ZERO, Saf Ch #1 @ * ____
4.2.2 To CAL, Saf Ch #1 @ * ____
4.2.3 Pos. 1 - 6, Meter & Rcdr Agree, (EXT RANGE Lt Out @ Pos 2), Rtn Sw to OPER ____
4.3 Linear Range Selector Sw:
4.3.1 ZERO, Red Pen @ * ____
4.3.2 CALIBRATE, Red Pen @ * ____
4.3.3 Operating Range ____
4.4 Safety Ch #2 Cal Switch:
4.4.1 ZERO, Saf Ch #2 @ * ____
4.4.2 CAL, Saf Ch #2 @ * ____

5. SCRAM CHECKS & ANNUNCIATION

5.1 Secondary Pressure:
5.1.1 SAF-1/LOG Switch to Pos. #1, W/R Trip Test to 1 , SEC PRESS SCRAM (on) @ 1 ____
5.1.2 Start 2d Flow, Reset ____
5.1.3 Rtn Both Switches OP/OFF ____
5.2 High PC Temperature:
5.2.1 Rcdr to 150°F, Alarm @ * ____
5.2.2 Rcdr to 155°F, SCRAM @ * ____
5.2.3 Rcdr to Normal ____
5.3 Manual SCRAM:
5.3.1 Reset, Safety #1 to* 25 ____
5.3.2 Manual SCRAM ____
5.4 Safety Channel #1 High Pwr:
5.4.1 Reset, Safety #2 to 25 ____
5.4.2 SAF-1/LOG Switch to Pos 1 ____
5.4.3 W/R Trip Test to 125%, SAF 1 SCRAM (on) @ * ____
5.4.4 Both Switches to OP/OFF ____
5.5 Period:
5.5.1 Reset, Safety #3 to* 25 ____
5.5.2 SAF-1/LOG Sw to Pos 1 ____
5.5.3 Pd Trip Test to 3 Sec, PERIOD SCRAM @ * ____
5.5.4 Both Switches to OP/OFF ____
5.6 Safety Ch #2 High Power:
5.6.1 Reset, Reg & 1 blade to about 25 ____
5.6.2 Saf Ch #2 Trip Test to 125%, SCRAM @ * ____
DUMP VALVE Light (on) ____
PC PUMP Light (lower 1/2 on) ____
COOLANT SCRAMS (on)(PUMP, FLOW, & LEVEL) ____
ON Lights (out) ____
DOWN Lights (on) ____
5.6.3 Rtn Trip Test Sw to OFF ____
5.6.4 PC Pump Switch, Depress ____
5.6.5 2d Water, (on) or (off) * ____
5.6.6 Time - Date Charts ____
5.6.7 Operating Log Entries ____
5.6.8 Console Key Secured ____
5.6.9 Temp Rcdr (Print wheel down) ____
5.6.10 Log Lin Rcdr (pens up, chart drive off) ____

Time Begin: ____________
Time Compl: ____________
Date: ____________
Operator: ____________
Trainees: ____________

REMARKS (Use other side if needed):

SOP A.1, rev 31 May 80

Figure 13-2. UFTR Daily Pre-Operational Checklist from UFTR SOP-A.1, Part II.

TIME START: ______________

A. Stop Vent, Dilute......... ..____

B. Check Dilute.................____

C. Check RDM oil................____

D. Inclined Manometers(d/p=2)....____

E. Source Check portal monitor.. ____

F. Start "Vent," "Dilute," (Int'Lk, oper)...............____RPM

G. Start "Shield Water".........____

H. Start "Demin," Temp Recorder......................____

I. Equipment Pit

1. Check gamma level
   Pri coolant tank........____mr/hr
   Core vent filter........____mr/hr
   Pri Demin...............____mr/hr
2. Check Demin Flow........____gpm
3. Check Rupture disc......____
4. Check Dump Valve........____
5. Check PC tank level.....____in

J. Reset "Reactor ON" (4 Ext. lights ON)................____

K. Start Log/Lin N Recorder...____

L. Start "Pri Coolant"........____gpm

M. Secure Reactor Console Key ____

N. APD:

1. Filter, alarm..........____
2. Check flow............____lpm

O. In Line Resistivity, PC

1. Inlet (>.4 MΩ-cm)......____MΩ-cm
2. Outlet (>1MΩ-cm).......____MΩ-cm

P. Check Stack Alarm..........____

Q. RM: Check Trips and coincidence
   Trip 2(2.5 mr/h)..........____mr/hr
   Trip 1 (10 mr/h)..........____mr/hr
   Coincidence...............____

R. Check manual evacuation alarm (oper).................____

S. Reset A/C....................____

T. SW System: Resistivity.......____MΩ-cm

U. Core vent filters d/P
   rough...............____in
   abs.................____in
   vent................____in

V. PC: Resistivity (>.4 MΩ-cm)..____MΩ-cm

W. Sec sample

1. Sample tank...............____
2. Heat exchanger............____

X. Blade Withdrawal time checks:

Blade	Time	Final Position
S-1		
S-2		
S-3		
Reg		

Y. Stop "Vent," "Dilute," "PC," Log/Lin N, Temp.............____

Z. Secure Reactor Console Key... ____

AA. DUMP PC (cycle console power)......................____

BB. CLEAN sec strainer........... ____

COMMENTS

Records.........................____

Time Completed ______________

Date ______________

Trainee ______________

Operator ______________

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Figure 13-3. UFTR Weekly Pre-Operational Checklist from UFTR SOP-A.1, Part I.

2. Shutdown and operation of core vent and diluting fan are tested.

3. The oil level of the control rod drive mechanisms is checked.

4. The portal monitor is source checked.

5. The Shield Tank water resistivity is checked.

6. The equipment in the reactor pit is checked for proper operation, radiation levels and determination of any possible leaks in the primary cooling system.

7. The primary and secondary coolant resistivity is checked.

8. Control blade withdrawal times are checked and recorded.

9. The operation of "Reactor On" exterior lights is checked.

13.5.2.2.3 Quarterly Checks. Checks, tests and maintenance performed at the UFTR facility on a quarterly basis are summarized below:

1. The radiation monitors including the area monitors and the Reactor Vent System monitors are calibrated.

2. Evacuation drills are conducted for facility personnel, insuring their familiarity with the emergency plan.

3. The safety system operability tests are performed to check reactor scram functions in the event of:

   a. Loss of primary coolant pump power.
   b. Loss of primary coolant level.
   c. Loss of Shield Tank water level.
   d. Loss of power to ventilation and dilution fans
   e. Loss of secondary coolant flow, at power levels greater than 1 Kw.

   f. Loss of electrical power to the console.
   g. Loss of chamber high voltage.
   h. High average outlet temperatures.

13.5.2.2.4 Semi-annual Checks. Checks, tests and maintenance performed at the UFTR facility on a semi-annual basis are summarized below.

1. Verification is made to assure the minimum shutdown margin, with the most reactive blade withdrawn, is 2% $\Delta k/k$.

2. Verifications are performed to assure that the reactivity insertion rate for any single control blade does not exceed 0.06% $\Delta k/k$ second, when determined as an average over any 10 seconds of blade travel time from the characteristic experimental differential blade reactivity worth curve.

3. The control blade drop time is checked from the fully withdrawn position.

4. The control blade reactivity worth is checked.

13.5.2.2.5 Annual Checks. Routine tests, maintenance and monitoring operations carried out on an annual basis include:

1. Calibration of the log N-period channel, power level safety channel, and linear power level channel.

2. Performance of an intensive, in-depth review of UFTR facility operations.

13.6 Industrial Security

The plans for physical protection of the UFTR facility are described in the Physical Security Plan for the UFTR, already submitted to the NRC under separate cover and withheld from public disclosure pursuant to 10 CFR 2.790(d).

APPENDIX 13A

REACTOR SAFETY REVIEW SUBCOMMITTEE CHARTER

Reactor Safety Review Subcommittee

CHARTER

I. DESIGNATION

The name of the Subcommittee is: Reactor Safety Review Subcommittee. The Subcommittee may be referred to in abbreviated form as the RSRS.

II. ACCOUNTABILITY

The RSRS is a Subcommittee of and reports to the University Radiation Control Committee, which may be referred to in abbreviated form as the URCC. The URCC provides radiological safety recommendations to the Director of Environmental Health and Safety.

III. SCOPE

The RSRS shall be responsible for the review of safety-related issues pertaining to the University of Florida Training Reactor, which may be referred to in abbreviated form as the UFTR.

IV. PURPOSE

The purpose of the RSRS is to assure the safe operation of the UFTR through the discharge of the Subcommittee review and audit functions.

V. REVIEW FUNCTION

The following items shall be reviewed:

1. Determination that proposed changes in equipment, systems, test, experiments, or procedures do not involve an unreviewed safety question,

2. All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance,

3. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity,

4. Proposed changes in technical specifications, license or charter,

5. Violations of technical specifications, license or charter. Violations of internal procedures or instructions having safety significance,

6. Operating abnormalities having safety significance,

7. Reportable occurrences,

8. Audit reports and annual facility reports.

A written report or minutes of the findings and recommendations of the review group shall be submitted to RSRS members in a timely manner after the review has been completed and to the Chairman of the Radiation Control Committee whenever a finding is deemed to require review by upper level University of Florida administration.

VI. AUDIT FUNCTION

The audit function shall include selective (but comprehensive) examination of operating records, logs and other documents. Where necessary, discussions with cognizant personnel shall take place. In no case shall the individual immediately responsible for the area, audit in that area. The following items shall be audited:

1. Facility operations for conformance to the technical specifications and applicable license or charter conditions, at least once per calendar year (interval between audits not to exceed 15 months),

2. The retraining and requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months),

3. The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months),

4. The reactor facility emergency plan, and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months).

VII. MEMBERSHIP

1. The RSRS shall consist of at least five members. Membership will include the Chairman of the Nuclear Engineering Sciences Department, University Radiation Control Officer, Reactor Manager and two technical personnel familiar with the operation of reactors and with the design of the UFTR and radiological safety, at least one of whom is from outside the Department of Nuclear Engineering Sciences. Any member may designate a duly qualified representative to act in his absence from a standing URCC approved list.

2. An Executive RSRS Committee will consist of the Reactor Manager, University Radiation Control Officer and Chairman of the RSRS.

3. The Chairman of the RSRS will be appointed by the Chairman of the URCC. The Chairman of the RSRS is an ex-officio voting member of the URCC and will serve as liason between the RSRS and the URCC.

4. Members appointed to the Subcommittee shall be reviewed, and as appropriate, new appointments made by October 1 of each calendar year.

VIII. MEETINGS

1. Meeting frequency shall be quarterly at intervals not to exceed 4 months.

Meeting may be held more frequently as circumstances warrant, consistent with the effective monitoring of facility operations as determined by the RSRS Chairman

2. Review of draft minutes will be completed prior to subsequent meetings, at which time they will be submitted for approval. Responsibility to assure that this is done falls upon the RSRS Chairman. The RSRS Chairman is charged with the responsibility to assure that the minutes are submitted for approval in a timely manner.

3. A quorum shall consist of at least three members and at least three members must agree when voting, regardless of the number present.

14. INITIAL TEST PROGRAM

14.1 Specific Information to be Included in Preliminary Safety Analysis Reports

This section is not considered since this document is presented as a FSAR in support of renewal of an existing license for the already operating UFTR facility.

14.2 Specific Information to be Included in Final Safety Analysis Report

There is currently no initial test program considered for the UFTR facility. Since the UFTR is an already operating facility as presented for license renewal, an initial test program is not appropriate. A test program will be developed and included in this safety analysis report at any time that significant physical or operational safety-related changes are proposed for the UFTR reactor facility.

15. ACCIDENT ANALYSIS

This chapter addresses the evaluation of safety of the UFTR facility to include analyses of the response of the facility to postulated disturbances in process variables and to postulated malfunctions or failures equipment.

The UFTR structures, systems and components important to safety have been presented and evaluated for their susceptibility to malfunctions and failures in previous chapters. In this chapter, the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the UFTR facility to control or accommodate such failures and situations and/or to identify the limitations of expected performance.

The situations analyzed and results presented in Sections 15.1, 15.2 and 15.3 along with the Appendices to this Chapter are similar to those presented in the original UFTR Hazards Summary (2) addressing generic safety-related issues for Argonaut-type reactors and are repeated here with few but appropriate changes to account for the UFTR facility as it currently exists. This chapter concludes with Section 15.4 which contains an assessment of radiation doses applicable for a Maximum Hypothetical Accident which has an assumed fission product release whose potential hazards are not near to being exceeded by those from any accident considered credible for the Argonaut- type UFTR reactor facility. (14)

15.1 Introduction

The effects of anticipated transients, accidents and postulated component failures are presented in this chapter. The predicted consequences of such events are determined and the capability of the UFTR facility to control or accommodate such failures and related situations is evaluated. As a result of this accident analysis, the system performance characteristics and limitations are identifed for the UFTR facility.

Several accident categories are considered in this analysis to include nuclear excursions during UFTR operation, loss of coolant accident during full power operation, safety control rod system malfunctions and possible release of fission products associated with reactor malfunction. This analysis presented in Sections 15.1, 15.2 and 15.3 is based primarily upon the contents of the original UFTR Safety Analysis Report (Hazards Summary).(2) There are no substantive changes in these chapters from the original Hazards Summary Report. Finally, the predicted dose associated with analysis of a so-called Maximum Hypothetical Accident, resulting in large scale release of radioactivity is presented in Section 15.4. This last analysis is based upon more recent data and calculations.

15.1.1 Nuclear Excursions

15.1.1.1 Nuclear Excursions During Operation. It is difficult to visualize any circumstances which would result in a reactivity increase of a magnitude sufficient to cause serious degradation of the UFTR core. The design of the cooling system insures that the temperature of the reactor cannot be changed suddenly by the introduction of cold water. The maximum excursion which could occur with the normal fuel loading would result from the suddent insertion of all the available excess reactivity; with the present fuel loading, the UFTR has a total excess reactivity of ≈1.0% Δk/k available. A maximum of 2.3% excess Δk/k can be loaded. Only two (2) methods are considered possible for loading such an excess reactivity. First, the maximum excess reactivity could be reached by having the reactor temperature lowered to the freezing point of water; second, the maximum excess reactivity could be reached by having the reactor temperature lowered to the freezing point of water; second, the maximum excess reactivity could be reached by violation of the standard operating procedures.

The first method for insertion of maximum excess reactivity by reduction of reactor temperatures to the freezing point is not considered feasible or plausible, not only because of the building and climate involved but also because of the time element that would be required during which some abnormalities would be noted. As explained in the original UFTR Hazards Summary Report, the second method for insertion of maximum excess reactivity violation of the standard operating procedures is a possibility.(2)

The Hazards Summary addresses two possible violations of SOPs by which the maximum excess reactivity in the UFTR could be achieved. The first violation involves loading a sample into the reactor with sufficient absorption properties to prevent startup or reaching criticality regardless of the amount of control blade withdrawal. If the control blades were fully withdrawn in this situation and criticality were not achieved, the maximum reactivity could be added if the sample were then removed without reinserting the control blades.

The other possible, although extremely difficult, manner by which the maximum excess reactivity can be inserted would be by purposely and wantonly bypassing the Reactor Control and Safety System interlocks and trips and subsequently withdrawing the blades, in violation of the Technical Specifications and the Standard Operating Procedures.

If all the circuits of the Reactor Protection System were to fail or be incapacitated, the power level would continue to rise until the available excess reactivity were overcome by the temperature and void coefficients characteristic of the present reactor configuration.

As a result of studies made for the original Hazards Summary Report (2) concerning the effects of a large reactivity addition in the UFTR during 100 KWth operation, it was also determined that the required power excursion in order to raise the temperature of the fuel plates to the melting point of aluminum (1220°F) involves an energy generation of 32 MW-sec, as explained in Appendix 15A (22). The corresponding exponential period for this excursion is 8.3 milliseconds; therefore, the UFTR will tolerate a power excursion with a period at least as short as 8.3

milliseconds without melting any part of any fuel plate. The excess reactivity corresponding to such a period is 2.4% Δk/k; however, the UFTR has a licensed excess reactivity of only 2.3% Δk/k available. Because of strict control of all fuel not in the reactor, and the loading controls, it is considered very unlikely that any significantly larger amount of reactivity would ever be available. All fuel-plate spaces provided in the reactor are filled with fuel plates or dummy aluminum plates. Therefore, a major increase of fuel loading would require either the disassembly and reassembly of the fuel bundles or the presence of a large supply of enriched uranium in some other form. Neither possibility is considered credible.

A further comparison of the effects of a rapid reactivity insertion in research reactors versus the UFTR is contained in Appendix 15B based upon analysis in the original UFTR Hazards Summary Report (2).

15.1.1.2 <u>Nuclear Excursions During Fuel Loading</u>. A nuclear excursion during fuel loading is not considered credible. Fuel loading at the UFTR is required very infrequently, usually when the loss of reactivity from burnup makes it necessary to add additional plates. At such times, dummy fuel plates or old fuel plates in the chosen fuel bundle can be replaced. During fuel loading and unloading operations only qualified reactor operators and personnel are involved under the supervision of the Reactor Manager. The limitations and procedures to be followed are explained in detail in the UFTR SOP-C "Fuel Handling Procedures". The excess reactivity will be limited to 2.3% Δk/k by adjustment of the core loading. Any additional fuel plates are kept in a locked safe as described in Chapter 9 of this SAR and in the UFTR Security Plan submitted separately.

Additional fuel plates can be forced down into the fuel boxes between the fuel bundles due to the existing clearances necessary for the removal of fuel bundles and manufacturing tolerances. However, all fuel plates not required for the reactor loading are locked in the safe which is accessible only to the Director of Nuclear Facilities and/or the Reactor Manager. The probability of misloading the reactor by forcing extra plates into these spaces is considered to be extremely small.(2) Therefore it is concluded that a nuclear excursion during fuel loading is not very likely to occur.

15.1.2 <u>Safety-Control Blade System Malfunctions</u>

The UFTR control blade drive system consists of a two phase fractional horsepower motor that operates through a reduction gear train, and an electrically energized magnetic clutch that transmits a motor torque through the control blade shaft, allowing motion of the control blades. The blades are sustained in a raised position by means of this motor, acting through the electromagnetic clutch. Interruption of the magnet current results in a decoupling of the motor drive from the blade drive shaft, causing the blades to fall back into the core in a failsafe arrangement. In case of a loss of power, a manual scram, or any scram signal from the instrumentation system, the electromagnets are de-energized and the system fails safe by gravity dropping of the blades into the core.

The blade withdrawal inhibit system is part of the Reactor Protection System. The multiple blade withdrawal interlock prevents exceeding the reactivity addition rate of 0.06% $\Delta k/k$ per second, as specified in the UFTR Technical Specifications. The fast period blade withdrawal interlock prevents establishing a period shorter than 10 seconds by blade withdrawal.

The safety blades each control from about 1.3% to 2.3% $\Delta k/k$ in reactivity. The only way in which the rods could fail to fall into the reactor during a reactor scram would be through either failure of the circuits to de-energize the electromagnetic coupling, or the jamming of the blades in their shrouds. The operator can manually scram the reactor or turn off the power in the case of circuit malfunction. In the event of blade jamming, or combined circuit and operator failure, the reactor is shut down by the inherent shutdown mechanisms described in Chapter 4 and by the water dump trip acting as a back-up shutdown mechanism. Additonal back-up for reactor shutdown is provided by the dumping of moderator/coolant via the rupture disk.

Since the two power nuclear channels are completely independent, failure of all scram circuitry is very unlikely. A short-period scram is provided on the log channel and high-power level trips are provided in both channels. The UFTR SOPs and Technical Specifications require the testing of the instruments and scrams every operating day to insure their proper operation prior to reactor startup. The reactor key is available only to UFTR licensed reactor operators. The reactor key is used to turn the console power on and energize the magnets for control blade motion.

In the event of a mulfunction in the control drive system, the operator can initiate a reactor scram; even if the operator fails to recognize a malfunction, a scram occurs automatically whenever a power level increase above the preset limits is caused by the malfunction. This response is described in Chapter 7 and specified in the Technical Specifications. No single failure or malfunction related to the magnets, limit switches, gear reducers, motors, or instrumentation could prevent all of the blades from dropping into the core after de-energizing the magnets.

15.2 Loss of Coolant Accident

The UFTR Reactor Protection System is discussed in Chapter 7 and provides a series of interlocks and trips preventing operation in the case of primary and/or secondary cooling systems malfunction. Interlocks are provided to prevent operation when the coolant is not circulating or when the level is outside the preset limits. Reactor rod-drop trips are provided for the primary coolant pump power and flow. Redundancy is provided through the reactor core water level trip. Inherent protection is provided by the negative moderator temperature and void coefficients of reactivity. No credible circumstances are envisioned where mishandling of the cooling system can give rise to a power excursion. Studies have been performed by Wagner to analyze the effects of a Loss of Coolant Accident (LOCA) at various hypothetical power levels up to 625 KWth as addressed in Appendix 15C.(23) It should be noted that the UFTR will shut itself down due to the negative moderator void coefficient; therefore, insertion of the control blades is not a physical requirement for reactor shutdown. Wagner investigated the increase in fuel temperature following a loss of coolant and shutdown of the reactor either by the negative void coefficient of reactivity or by the insertion of the control blades into the reactor. Wagner's work summarized in Appendix 15C shows that the fuel temperature will increase about 30°F following a dump trip event. Figure 15-1 shows the calculated fuel temperature as a function of time after reactor shutdown due to decay heating effects following dump scram from equilibrium UFTR operation at 625 KWth power level.

This analysis demonstrates that even if the calculations described by Wagner are in error by as much as 200%, the maximum fuel plate temperature rise will not approach temperatures of half the melting point of aluminum for power levels much larger than present UFTR operation; therefore a LOCA is not considered to represent a hazard to the UFTR core fuel or structural integrity.

Experimental verification of the decay heat equation governing decay heat generation in the UFTR following reactor shutdown is provided in Appendix 15D, also taken from Wagner's work. (23)

15.3 Fission Products Release and Dose Assessment

The UFTR is designed to operate at a rated power of 100 KWth. The analysis discussed in Section 15.1 indicates the very low probability of fuel melting in case of an excursion resulting from the suddent insertion of as much as 2.4% $\Delta k/k$ along with failure of the reactor control and protection systems. Therefore significant releases of fission products are not considered plausible because of the inherent self-limiting characteristics of the UFTR. If a reactivity accident is assumed to occur and to cause the fuel plates to melt, a release of fission products may take place in this regard, some exposure studies were presented in the original UFTR Hazards Summary Report (2). Later analysis by Listing (22) also assumed a release of 10% of the volatile fission products from the reactor fuel plates into the building air in agreement with the Hazards Summary as indicated in Appendix 15 E.

15.4 Radiation Doses for the Maximum Hypothetical Accident

15.4.1 Methodology

The analytical methods used to predict radiation doses following a Maximum Hypothetical Accident at the UFTR facility are summarized in this section. The methodology presented includes basic equations and theory as well as basic input data and information for the calculations such as the equilibrium radioactive monitoring diffusion coefficients, release fractions from the fuel and hypothetical transport of the released nuclides.

15.4.1.1 Introduction to Basic Dose Calculations. The radioisotopes of greatest significance in case of an accident and consequently the only ones specifically addressed in the Nuclear Regulatory Commission Regula-

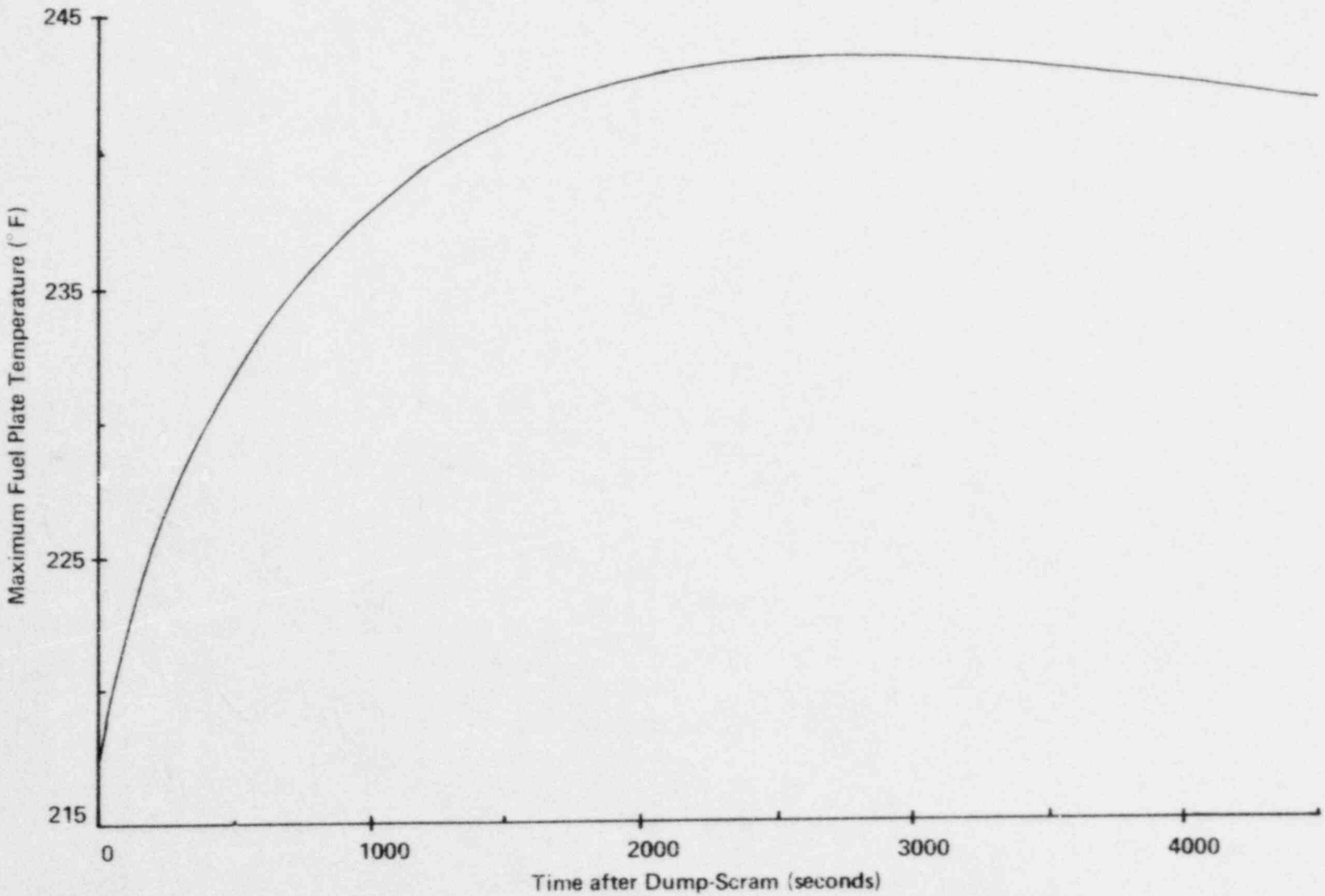


Figure 15-1. Calculated Fuel Plate Temperature Rise due to Decay Heating for Dump-Scram following UFTR Operation at 625 KWth.

tory Guides, are the noble gases and the radioiodines. The noble gases contribute solely by means of the immersion dose, and the radioiodines contribute primarily to the thyroid dose through inhalation. The applicable equations used to calculate the whole body dose and the thyroid dose are presented below:

WHOLE BODY DOSE

$$D_\gamma = \chi/Q \cdot (\sum_i Q_i \cdot DF_{\gamma i}) \quad (15\text{-}1)$$

THYROID DOSE

$$D_T = \chi/Q \cdot BR \cdot (\sum_j Q_j \cdot DFT_j) \quad (15\text{-}2)$$

where symbols utilized in these equations are defined as follows:

D_γ = whole body dose (rem);
χ/Q = atmospheric diffusion coefficient (sec/m^3);
Q_i = release to atmosphere of noble gas type "i" (Ci);
$DF_{\gamma i}$ = dose conversion factor for whole body dose from noble gas type "i" ($rem\text{-}m^3/Ci\text{-}sec$);
D_T = dose to the thyroid (rem);
BR = breathing rate (m^3/sec);
Q_j = release to atmosphere from radioiodine type "j" (Ci);
DFT_j = thyroid dose conversion factor for radioiodine type "j" (rem/Ci).

15.4.1.2 Equilibrium UFTR Radioactive Inventory. The computer code RIBD (Radio Isotope Buildup and Decay, Reference 45) was applied to calculate the isotopic concentration from a fission source, taking into consideration the beta decay, the isomeric transitions, and the (n, γ) reactions. The fission source in this case is the UFTR core operated full power to equilibrium radioactive inventory.

The input for RIBD consists of the average thermal flux in the fuel, the operation history in selected time steps of constant power, the energy released per fission and the desired times after shutdown at which the activities of the different isotopes are calculated. The output includes the activities at shutdown and at the specified times afterward for various isotopes. The primary radioisotopes of interest for this safety analysis are the iodines (especially I-131), and the noble gases represented by various krypton and xenon radioisotopes.

For the equilibrium UFTR inventory calculations at 100 KWth power with 93% enriched fuel, input to the RIBD code consisted of the average thermal neutron flux in the fuel ($1.0 \times 10^{12} n/cm^2/sec$), full rated UFTR power level (100 KWth) and total irradiation time (30 days). The average thermal flux in the fuel was obtained based upon the highly enriched fuel during full power operation at 100 KWth. (5) The equilibrium inventories calculated using the RIBD code for important radioisotopes for equilibrium operation of the 93% enriched fuel in the UFTR at 100 KWth are presented in Table 15-1.

Table 15-1

CALCULATED UFTR RADIONUCLIDE INVENTORY
FOLLOWING EQUILIBRIUM OPERATION AT 100 KWTH

RADIOACTIVE ISOTOPES	ACTIVITY OF 93% ENRICHED UFTR FUEL (curies)
IODINES	
I 131	2.207 E03
I 132	3.577 E03
I 133	5.726 E03
I 134	6.438 E03
I 135	5.356 E03
I 136M	1.682 E03
KRYPTONS	
Kr 83 M	4.482 E03
Kr 85 M	1.106 E03
Kr 87	2.146 E03
Kr 88	3.032 E03
Kr 89	3.946 E03
XENONS	
Xe 131 M	1.025 E01
Xe 133 M	8.031 E02
Xe 133	5.579 E03
Xe 135	5.571 E03
Xe 135 M	9.235 E02
Xe 138	5.274 E03

RIBD calculations were also run to verify that equilibrium radioisotope inventories are practically reached in a period of thirty days. However, the UFTR usually operates continuously for periods of only a few hours, generally due to its use for training purposes and for performing experiments with short irradiation times. Because of this limited operation, several runs of RIBD were performed using cycles defined as 8 hours of full power operation followed by 16 hours of shutdown to verify that the "equilibrium" UFTR radioactive inventory is reduced for such cyclic operation. For such cyclic operation, smaller values of radioactive inventories are expected. The expected reduced inventories arise because there is not as large a contribution from the precursors; the inventory for the cyclic operation has decayed during the shutdown period. For example, the I-131 inventory for a 4 cycle run of 8 hours on and 16 hours off is 94% of the "equivalent" continuous case, while the inventory of I-133 is only about 60% of the "equivalent" continuous case. These two iodine isotopes constitute the most important radionuclides of concern here for dose assessment and the inventory reduction associated with cyclic operation is significant.

15.4.1.3 Diffusion Coefficients for the Design Basis Accident. As presented in Chapter 2 of this Safety Analysis Report, two conservative approaches are recommended by the NRC in Regulatory Guide 1.111 for determining diffusion coefficients.(15) The more conservative model uses generic (NRC) meteorological conditions; the other method uses local meteorological conditions and is less conservative since credit for increased diffusion is possible in some regions. Both methods were used to compute diffusion coefficients for input to radiation dose calculations performed Maximum Hypothetical Accidents in the UFTR.

15.4.1.4 Fuel Release Fractions for the UFTR. In this study, it was conservatively assumed that 25% of the radioiodines and 100% of the noble gases were released from the failed fuel into the reactor cell as recommended by the NRC through the ANSI/ANS-15.7 Standard (46) and Regulatory Guide 1.111(15). The percentage of failed fuel was assumed to be 100% for the dose calculations considered here. This 100% failure is the common assumption used by the NRC in the evaluation of the radiation doses for the Loss of Coolant Accident associated with commercial Light Water Reactors as proposed in Regulatory Guide 1.4(20) and claimed in the Palo Verde PSAR(47). However, this 100% failure assumption is not made explicitly in the standard (ANSI/ANS-15.7) applicable to Test and Research Reactors for which smaller percentages are allowed. In this respect, it is worthwhile to point out, that the original "University of Florida Training Reactor Hazards Summary Report" submitted to the Atomic Energy Commission assumed a 10% release of all volatile radionuclides to the reactor cell atmosphere.(2)

15.4.1.5 Transport Model for Released Radionuclides. Assuming that the reactor cell ventilation stops after the accident has occurred, the transport model for radionuclides consists of two compartments: the reactor cell and the environs. The reactor cell radioisotope inventory, $N_i(t)$, is lost by decay and by leakage to the environs; the environs in turn lose radioisotopes only by decay. The initial number of atoms of radioisotope "i" present in the reactor cell is represented by:

$$N_{io} = f_i I_{io} \qquad (15\text{-}3)$$

where:

I_{io} = initial inventory of nuclide atoms of type "i"

f_i = fraction of type "i" atoms released from the fuel.

Given the two compartment model presented above, the total number of atoms of the nuclide "i" present in the reactor cell is governed by the following differential equation:

$$\frac{dN_i(t)}{dt} = -(\lambda_i + L)N_i(t) \qquad (15\text{-}4)$$

where

λ_i = radioactive decay constant for nuclide "i" (1/sec);

$N_i(t)$= number of atoms of nuclide "i" in the cell at time t;

L = fractional leak rate from the cell (1/sec).

The solution of Equation (15-4) is a simple exponential as follows:

$$N_i(t) = f_i I_{io} \exp[-(\lambda_i + L)t] \qquad (15\text{-}5)$$

so the number of atoms which escape to the environs per unit time, $E_i(t)$, is given by a similar exponential as follows:

$$E_i(t) = L f_i I_{io} \exp[-(\lambda_i + L)t] \qquad (15\text{-}6)$$

The total number of atoms which have escaped in the time interval running from $t = T_1$ to $t = T_2$ is then given by the following time-integrated expression:

$$E_i(t) = \frac{L f_i I_{io}}{\lambda_i + L} [\exp(-(\lambda_i + L)T_1] - \exp[-(\lambda_i + L)T_2] \qquad (15\text{-}7)$$

From this expression for the release of nuclide "i", the activity (Ci) released in the time interval between T_1 and T_2 is then given by $\lambda_i E_i(t)$ so that the activity released for nuclide "i" is directly related to the initial inventory of the radioactive nuclide "i" as expected. The total activity released can then be determined by adding up the corresponding contributions for radionuclides of interest.

15.4.2 Dose Calculation Model Code

The computer code, DORA, was written following the methodology of Section 15.4.1 and then used to calculate the radiation doses for the Maximum Hypothetical Accident based upon the model presented in Section 15.4.1.(14) DORA can be used to evaluate such doses for each time period and for each radioisotope; DORA adds all the individual contributions giving as an output the whole body dose and the thyroid dose for different time periods. The input for DORA consists of the leak rate from the reactor cell, the Design Basis Accident diffusion coefficients corresponding to four time periods, 0-2 hours, 2 hour-1 day, 1-4 days, and 4-30 days as presented in Chapter 2 of this SAR, and the core inventory of radioactive iodines and noble gases. The thyroid dose conversion factors for the radioiodines and the whole body dose conversion factors for the noble gases obtained from References 47 and 48. Both types of dose conversion factors along with the radionuclide decay constants used in DORA are presented in Table 15-2. The DORA library also contains decay constants for the various radionuclides of interest as well as the breathing rates applicable during each period; the applicable breathing rates by period following such a Maximum Hypothetical Accident for the UFTR are presented in Table 15-3.

15.4.3 Results of the Maximum Hypothetical Accident Dose Calculations

A sensitivity analysis was performed for the whole body and the thyroid doses, for the periods of 2 hours, 1 day and 30 days. The distance from the vent to the receptor as well as the leak rate were varied for the UFTR operating at 100 KWth full rated power. Cases were examined corresponding to all the possible combinations among the four different parameters listed in Table 15-4: receptor distance, time period, reactor building leak rate and type of meteorological conditions. The results calculated for these various site parameters are presented graphically in Figure 15-2 through Figure 15-11 as labeled.

The range for the leak rates from the Reactor Cell varies from the upper value of 20%/hr, which was the value used in the original UFTR Hazards Summary Report (2) to the lowest value of 0.1%/hr which is about 24 times higher than the design value for typical LWR containment leak rates. A wide body of standards and regulations exist concerning the determination of the containment leak rates for Power Reactors in the course of an accident. Usually the value incorporated in the Technical Specifications for Power Reactors depends upon the calculated containment peak pressure during the accident and has to be determined in the Preoperational Tests, in accordance with Appendix J to 10 CFR 50. Although no equivalent method has been applied for Test and Research Reactors, it is felt that the UFTR reactor building leak rate following a Maximum Hypothetical Accident should certainly be bounded by these extremes and likely will be on the lower end of this range.

15.4.4 Dose Assessment for the Maximum Hypothetical Accident

It can first be concluded that the doses calculated with the local Gainesville meteorology are almost a factor of 8 smaller than those obtained using the standard meteorological conditions recommended by the

Table 15-2

RADIONUCLIDE DECAY CONSTANTS AND DOSE CONVERSION FACTORS

RADIONUCLIDE	DECAY CONSTANT (1/sec)	THYROID DOSE CONVERSION FACTOR (rem/inhaled curie)	WHOLE BODY DOSE CONVERSION FACTOR (rem-m^3/Ci-sec)
I-131	9.975 E-07	1.48 E06	NA
I-132	8.424 E-05	5.35 E04	NA
I-133	9.255 E-06	4.00 E05	NA
I-134	2.196 E-04	2.50 E04	NA
I-136M	1.444 E-02	1.24 E05	NA
Kr-83M	1.035 E-04	NA	1.045 E-02
Kr-85M	4.297 E-05	NA	3.775 E-02
Kr-87	1.520 E-04	NA	3.437 E-02
Kr-88	9.358 E-07	NA	4.357 E-01
Kr-89	3.655 E-03	NA	5.156 E-01
Xe-131M	6.689 E-07	NA	4.189 E-02
Xe-133M	3.597 E-06	NA	8.250 E-03
Xe-133	1.512 E-06	NA	7.500 E-03
Xe-135	2.099 E-05	NA	6.150 E-02
Xe-135M	7.549 E-04	NA	1.055 E-01
Xe-138	8.133 E-04	NA	7.075 E-01

Table 15-3

APPLICABLE BREATHING RATES FOR THYROID DOSE CALCULATIONS

PERIOD	BREATHING RATE (m^3/sec)
0-2 hours	3.30 E-04
2-24 hours	2.58 E-04
1-30 days	2.64 E-04

Table 15-4

PARAMETRIC CASES EXAMINED FOR DOSE ASSESSMENT OF MAXIMUM HYPOTHETICAL ACCIDENT*

RECEPTOR DISTANCE	TIME PERIODS	REACTOR BUILDING LEAK RATES	METEROLOGICAL CONDITIONS
0.1 mi.	0-2 hrs.	20%/hr	Local Conditions
0.2 mi.	1 day	2%/hr	NCR Standard Conditions
0.3 mi.	30 days	0.2%/hr	
0.4 mi.		0.1%/hr	
0.5 mi.			

*These cases all correspond to the equilibrium UFTR inventory for the current design using 93% enriched fuel at 100 KWth.

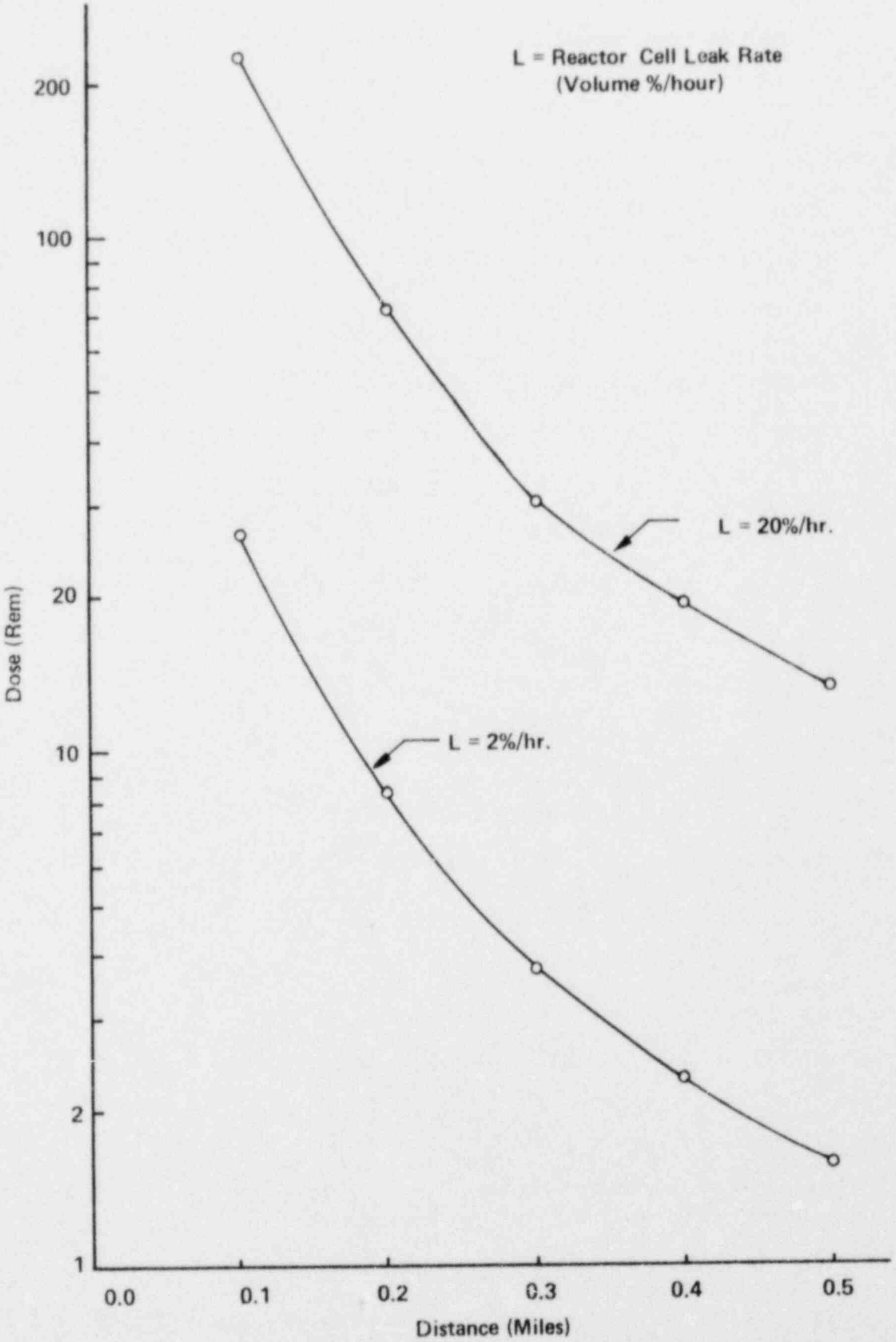


Figure 15-2. Two-Hour Thyroid Dose for UFTR Operation at 100 KWth Using Local Meterology.

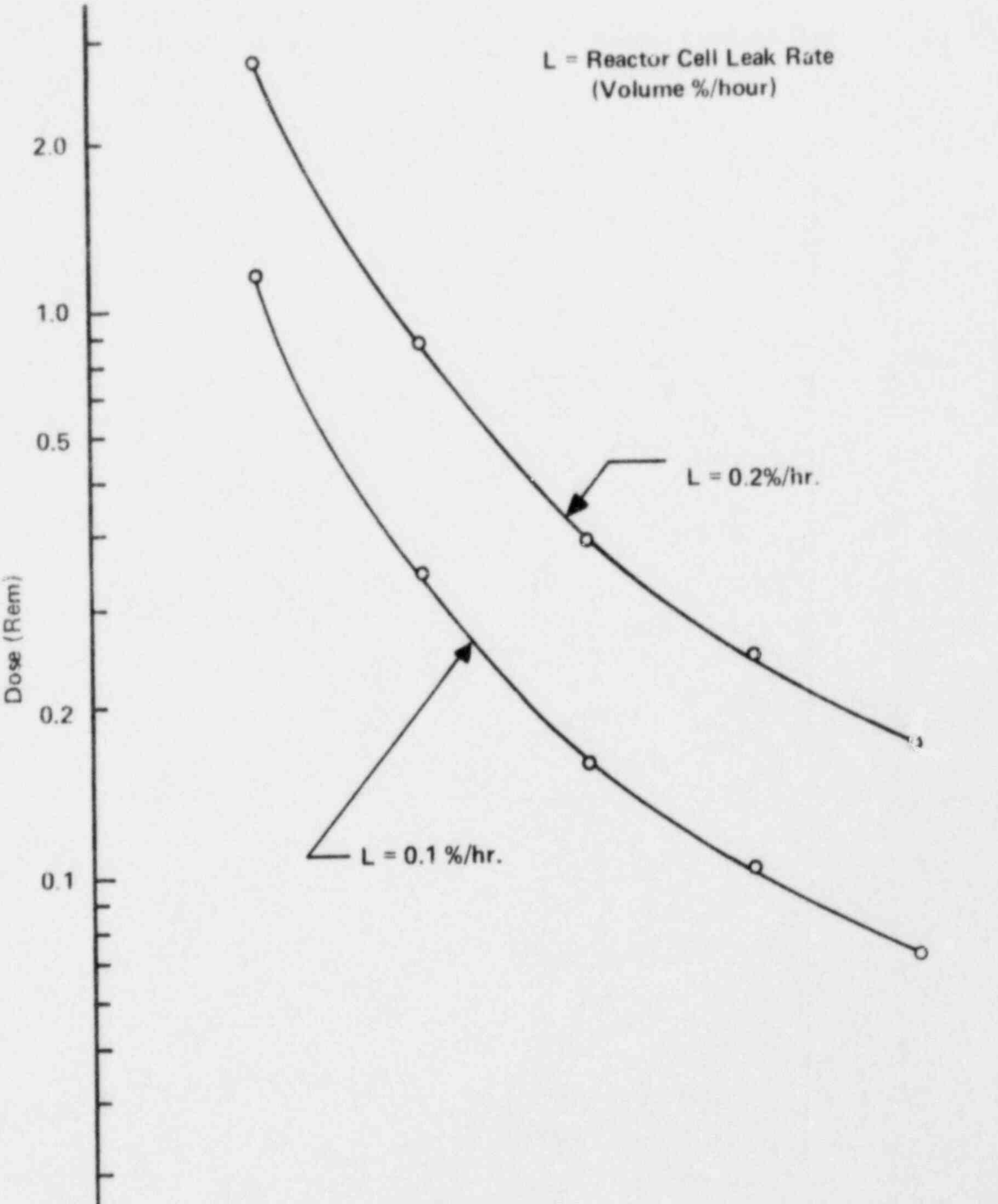


Figure 15-3. Two-Hour Thyroid Dose for UFTR Operation at 200 KW th Using Local Meteorology.

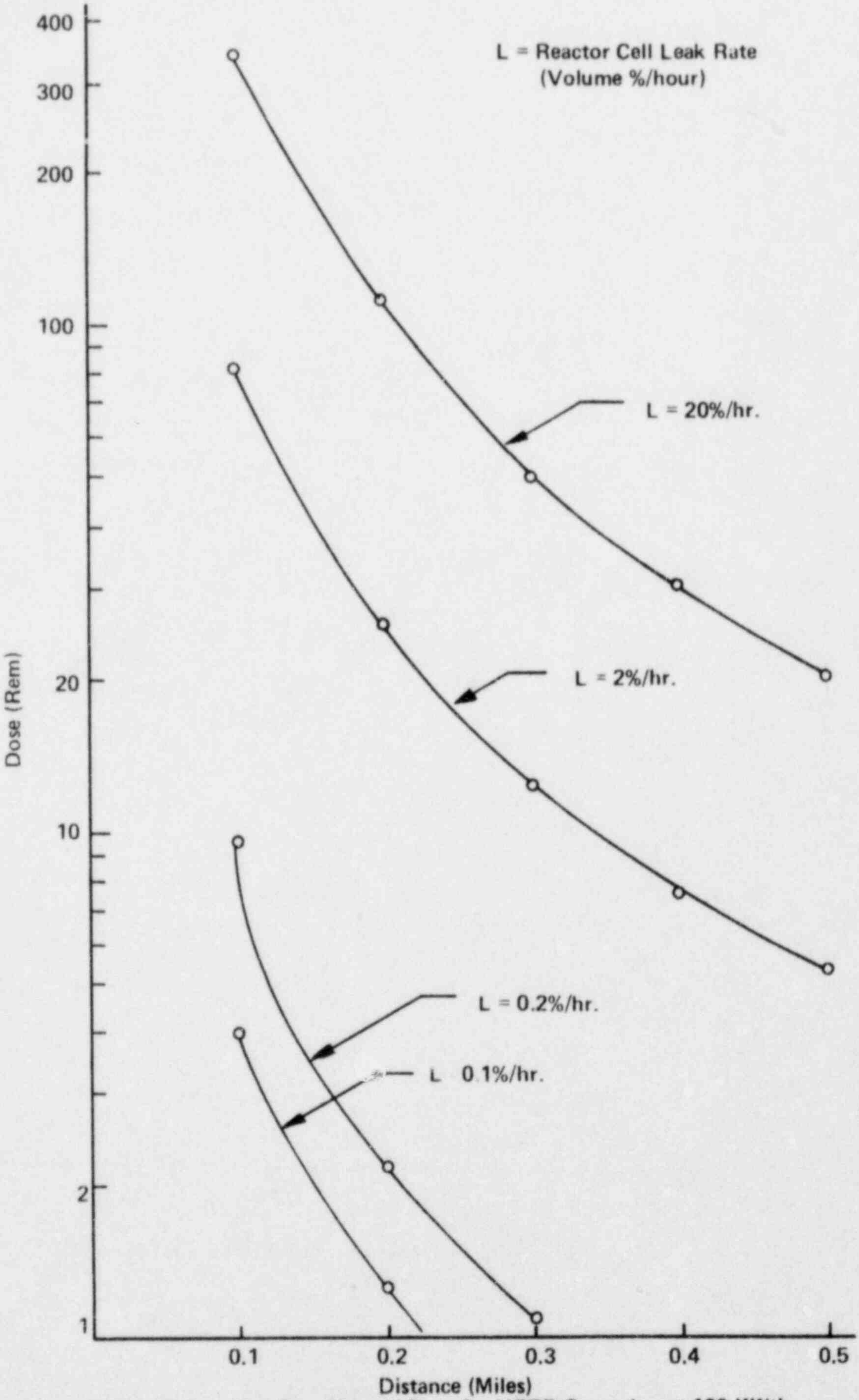


Figure 15-4. One Day Thyroid Dose for UFTR Operation at 100 KWth Using Local Meteorology.

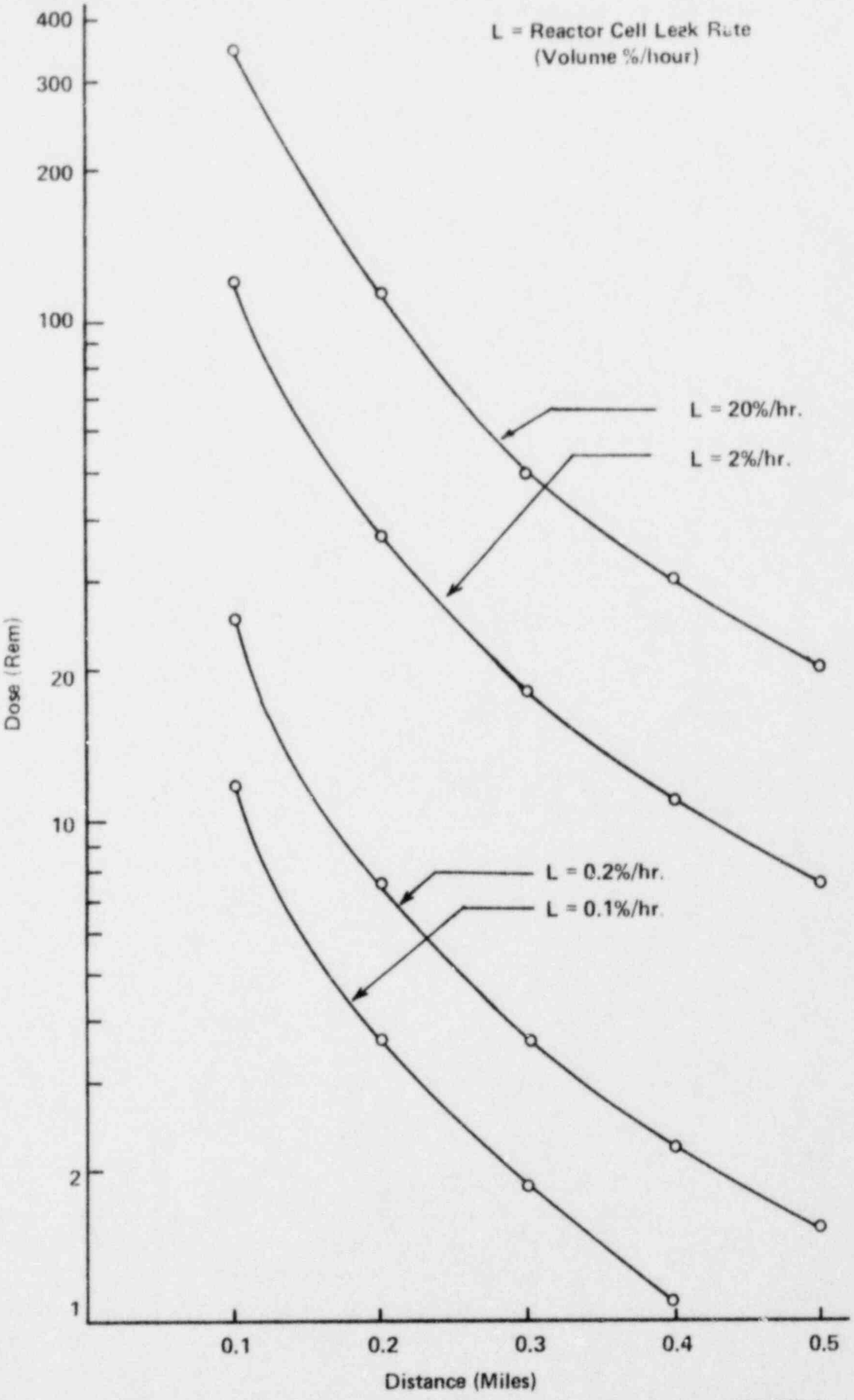


Figure 15-5. Thirty Days Thyroid Dose for UFTR Operation at 100 KWth Using Local Meteorlogy.

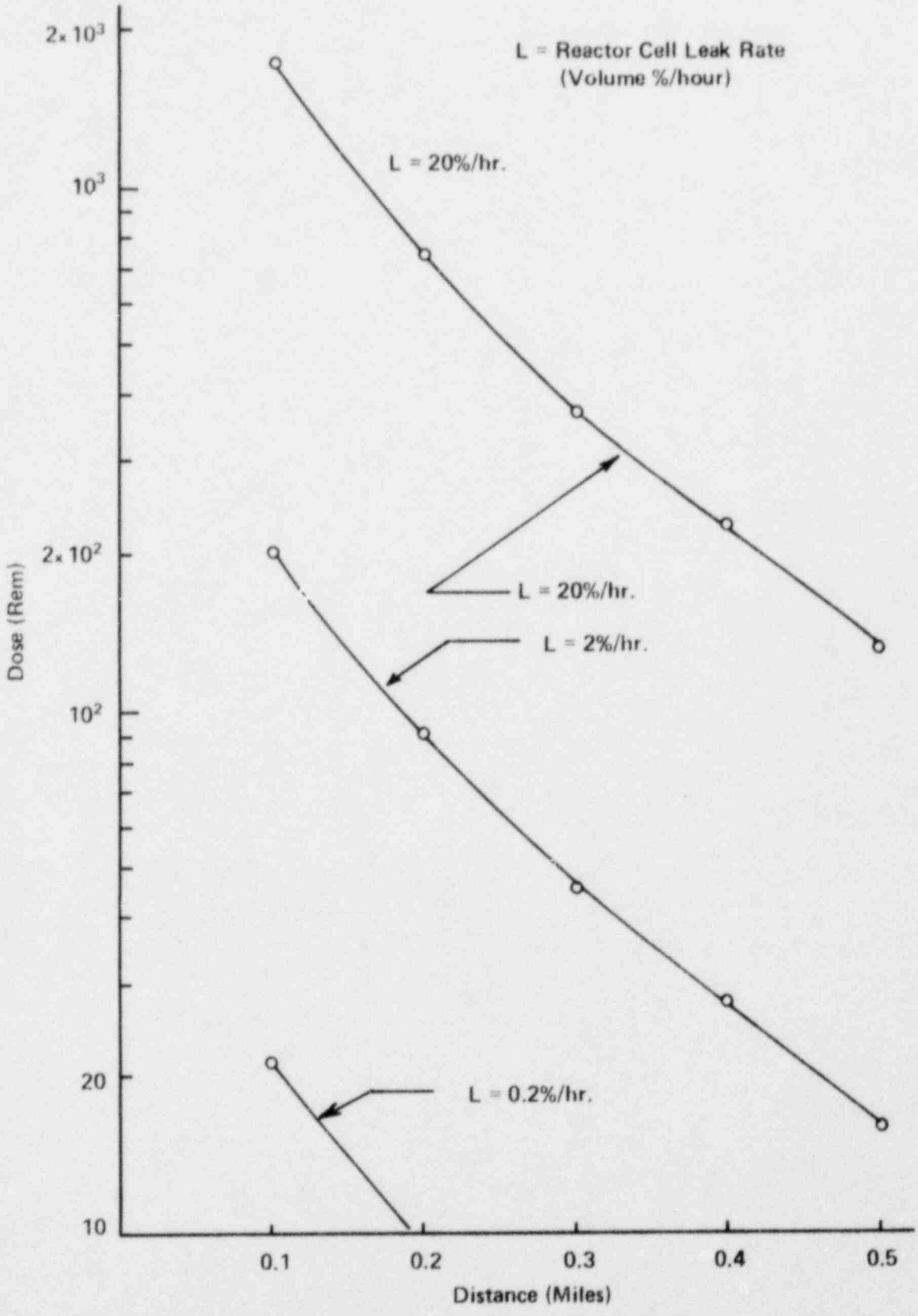


Figure 15-6. Two-Hour Thyroid Dose for UFTR Operation at 100 KWth Using NRC Standard Meteorology.

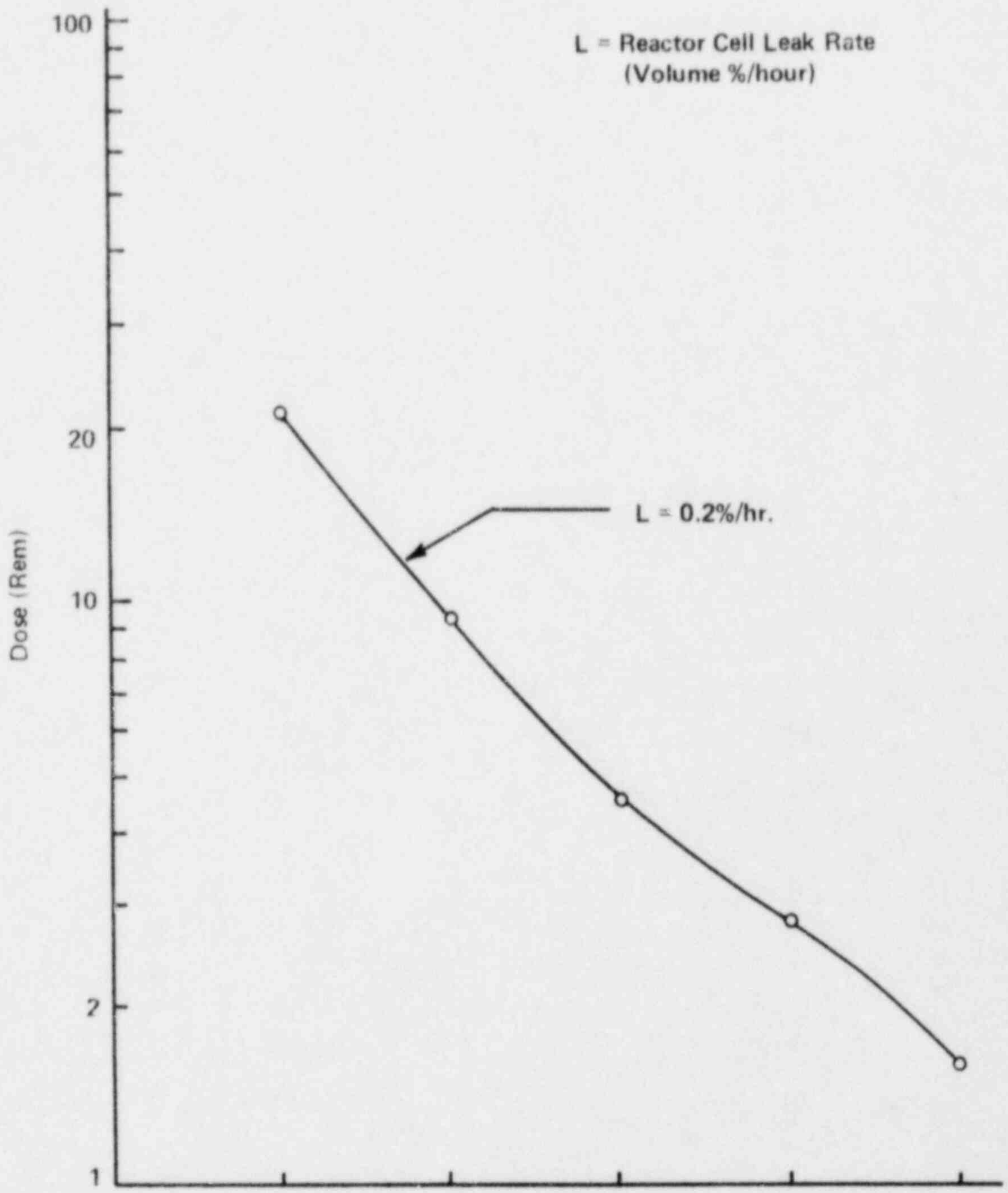


Figure 15-7. Two-Hour Thyroid Dose for UFTR Operation at 100 KWth Using NRC Standard Meteorology.

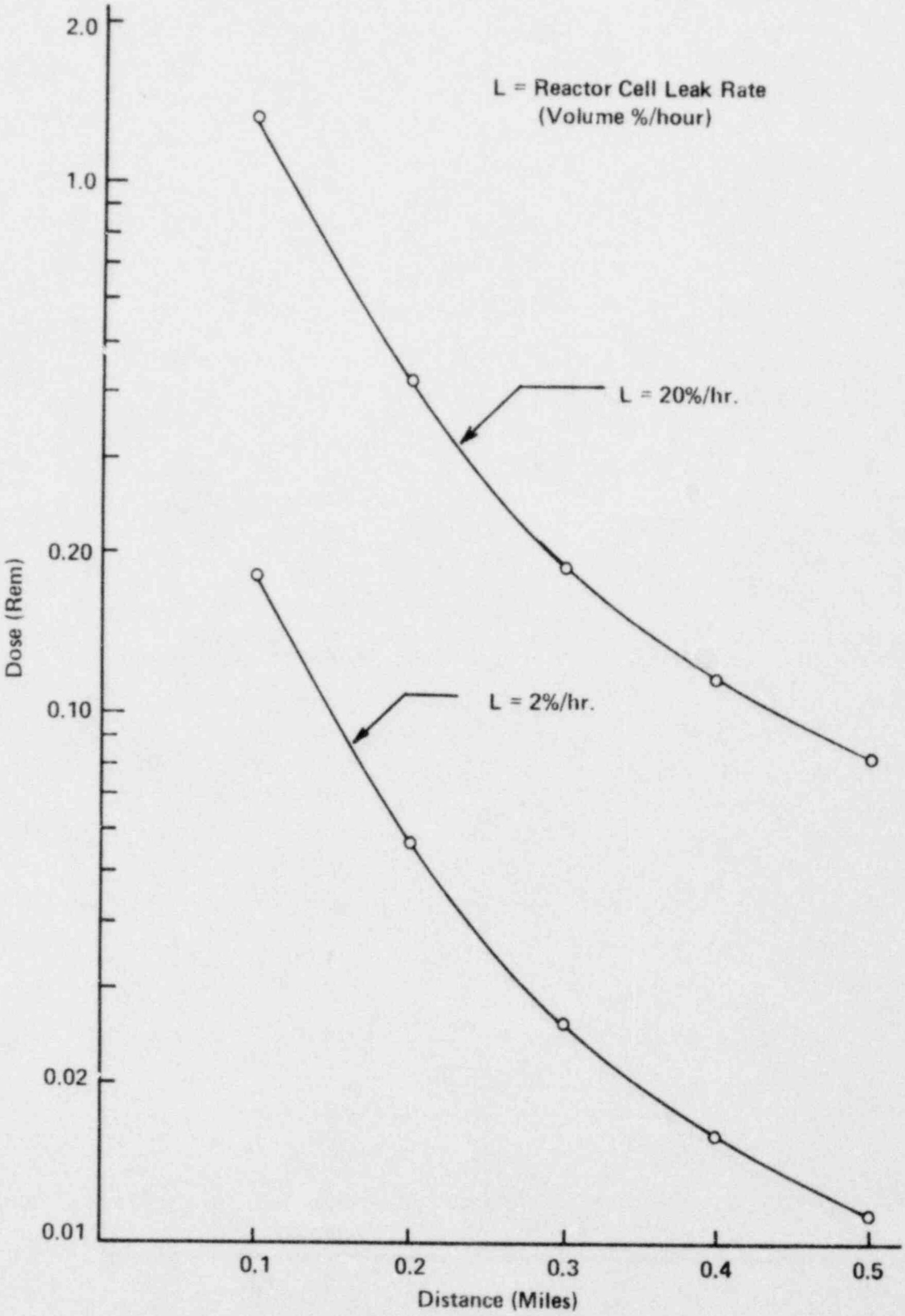


Figure 15-8. Two-Hour Whole Body Dose for UFTR Operation at 100 KWth Using Local Metorology.

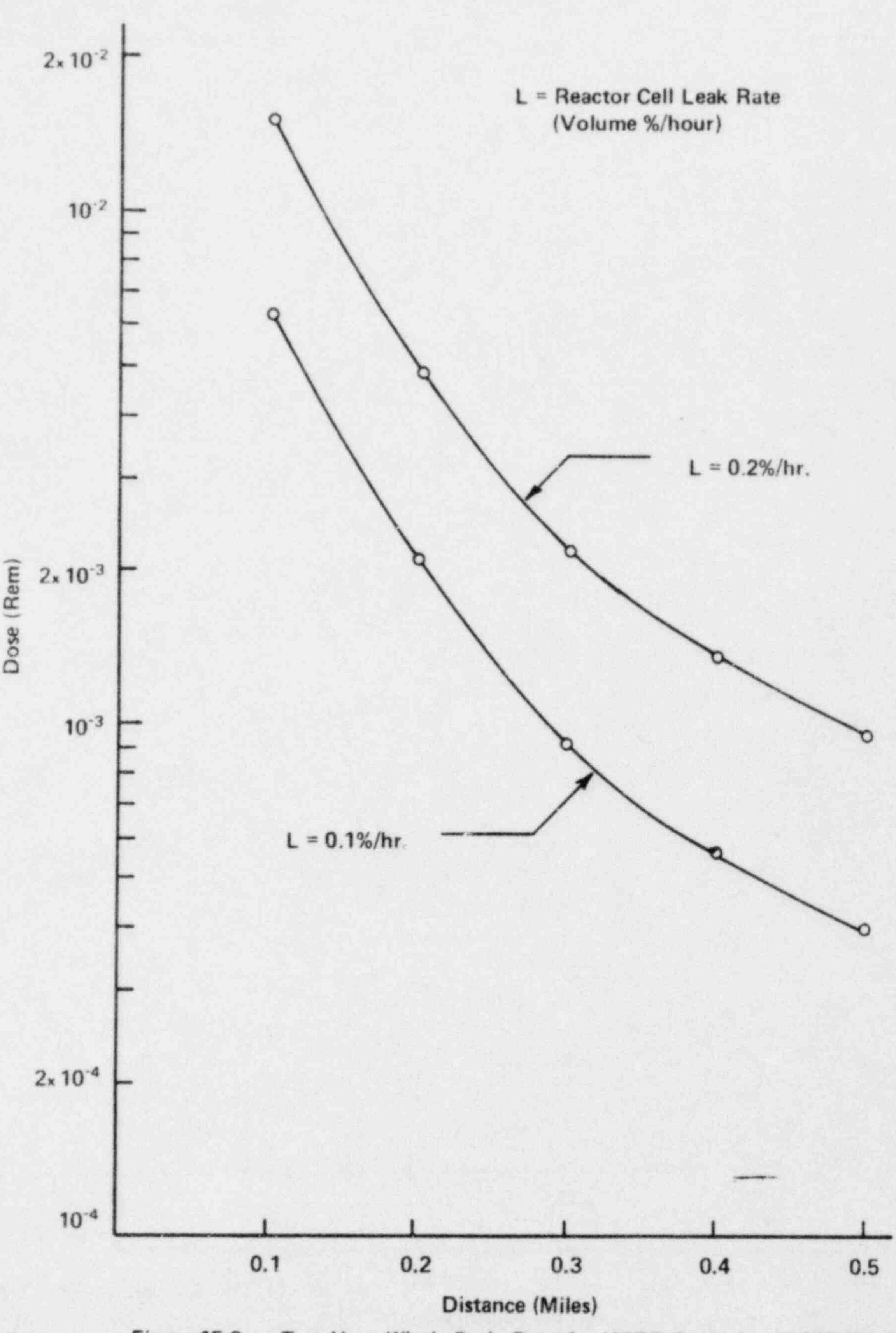


Figure 15-9. Two-Hour Whole Body Dose for UFTR Operation at 100 KWth Using Local Meteorology.

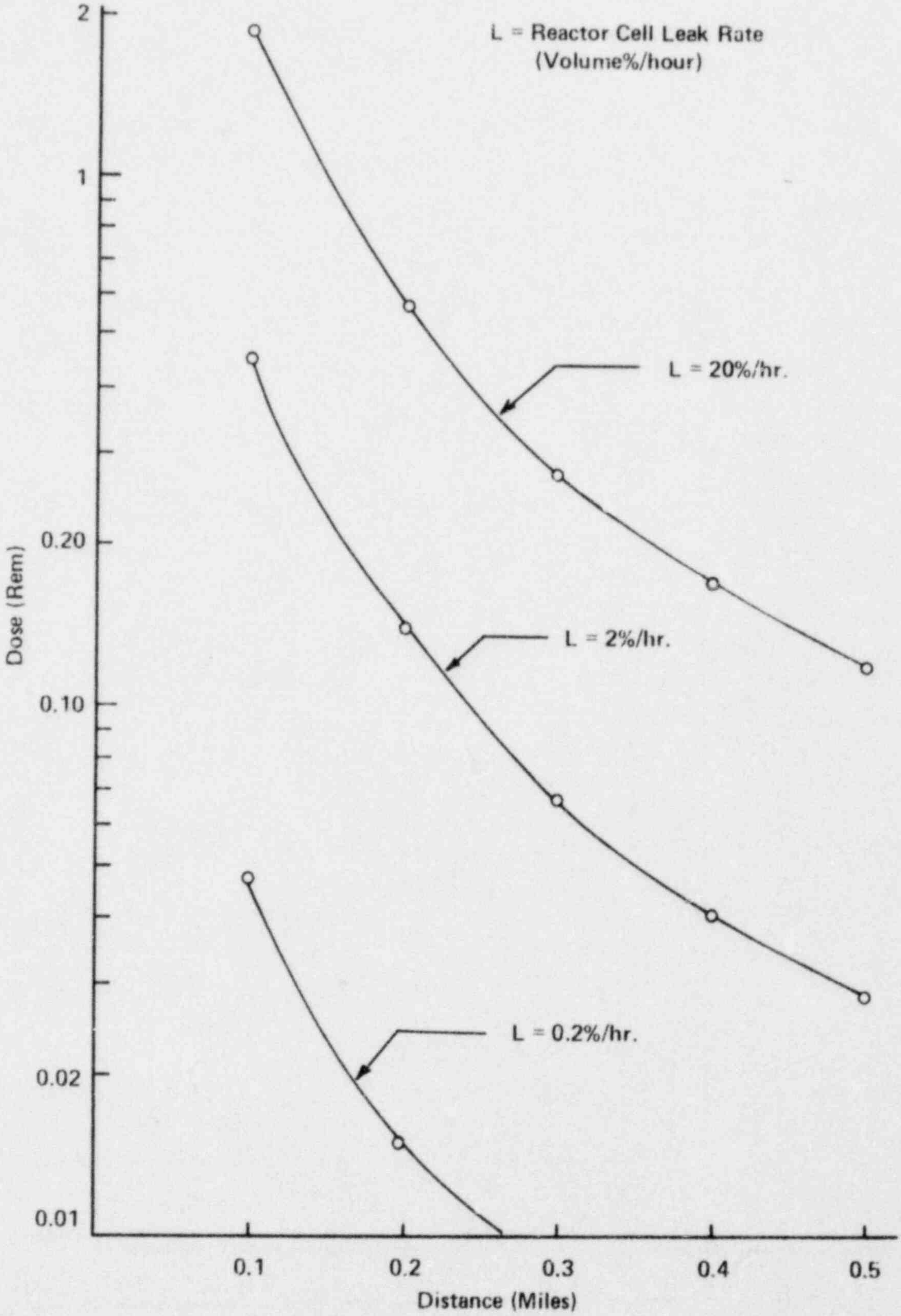


Figure 15-10. One Day Whole Body Dose for UFTR Operation at 100 kWth Using Local Meterology.

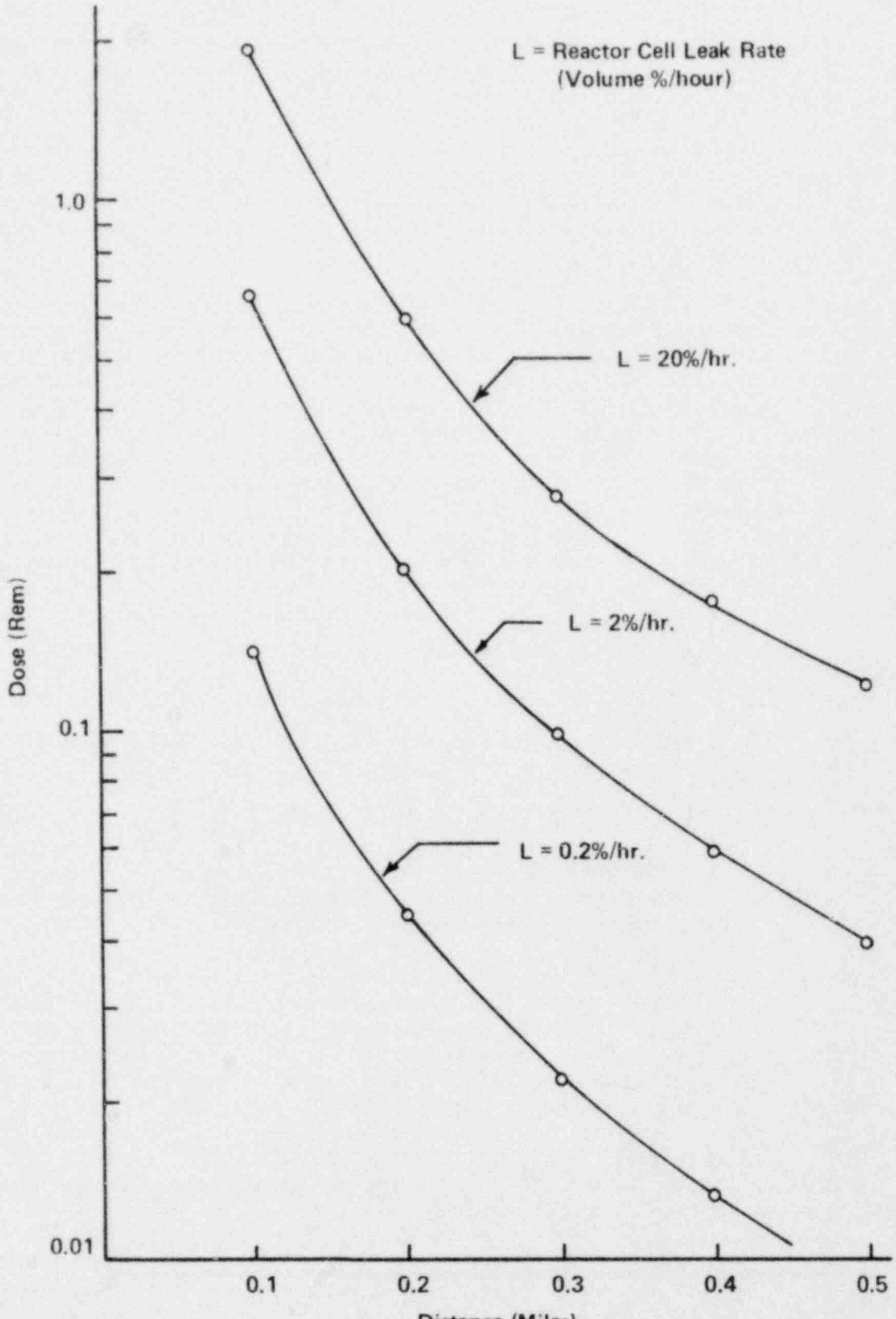


Figure 15-11. Thirty-Days Whole Body Dose for UFTR Operation at 100 KWth Using Local Meteorology.

NRC (20) which is easily demonstrated by comparing Figure 5-4 with Figure 5-7. This difference is expected since the NRC Standard Conditions are extremely conservative.

The second conclusion drawn from the MHA dose calculation is that there is a considerable reduction, about a factor of 15, when the distance from the receptor to the reactor vent varies from 0.1 to 0.5 miles. Therefore, from this basis, the urban boundary distance placed at 0.5 miles is reasonable.

Finally, analysis of the computer output results used to produce the final dose results shows that I-131 is the critical radioisotope which contributes most to the thyroid dose through inhalation. The relative significance of the I-131 contribution increases with time due to its longer half life. As shown in the figures for the MHA dose results, when the leak rate decreases, the time period at which the highest dose is received moves from the first period to later periods. For example, for a 20%/hr leak rate, the highest thyroid dose is received during the first period from 0-2 hours; for a 2%/hr leak rate the highest thyroid dose is received during the second period from 2 hours to 1 day; finally, for a 0.2% leak rate, the highest thyroid dose is received from 1 to 4 days. Therefore, for more likely leak rates, longer periods are required for peak doses and evacuation would be very possible for such longer periods.

15.4.5 Selection of Site Parameters Based on MHA Dose Results

In regard to Research Reactor Site Evaluation, the ANSI/ANS 15.7-1977 standard presents the following definitions: (46)

a) "Site boundary. The site boundary is that boundary, not necessarily having restrictive barriers, surrounding the operations boundary wherein the reactor administrator may directly initiate emergency activities. The area within the site boundary may be frequented by people unacquainted with the reactor operation."

b) "Urban boundary. The urban boundary means the nearest boundary of a densely populated area or neighborhood containing population of such number or in such a location that a complete rapid evacuation is difficult or cannot be accomplished within two hours using available resources."

The dose commitment for the "site boundary" is 5 rem to the whole body or 15 rem to any other organ for a two hour period. The dose commitment for the "urban boundary" is 0.5 rem to the whole body or 1.5 rem to any other organ for a one day period. For the UFTR case, the dose results represented in Figures 15-2 to 15-11 indicate that the thyroid dose limit is the critical dose as expected. The distances which would comply with the above dose limits for the site and urban boundaries, as a function of the assumed reactor cell leak rate for the UFTR operated at 100 KWth are presented in Table 15-5.

Due to actual site conditions for the UFTR, the urban boundary distance, according to its definition, can be estimated to be about 0.5 miles which is approximately the distance from the reactor building to Shands Teaching Hospital as indicated on the map presented in Chapter 5 of this SAR, Figure 5-1. This hospital is also the closest facility whose evacuation would be time-consuming in the event of a major radiological emergency at the UFTR. For this urban boundary to meet the dose limits specified in ANSI/ANS 15.7 would require a reactor cell leak rate of 0.5%/hr for UFTR equilibrium operation at 100 KWth. These leak rate are based on

Table 15-5

CALCULATED SITE AND URBAN BOUNDARY DISTANCES TO COMPLY WITH ANSI/ANS 15.7 - 1977 DOSE LIMITS FOR UFTR OPERATION AT 100 KWTH

HYPOTHETICAL REACTOR CELL LEAK RATE (%/hr)	SITE BOUNDARY DISTANCE REQUIRED BY ANSI/ANS 15.7 (MILES)	URBAN BOUNDARD DISTANCE REQUIRED BY ANSI/ANS 15.7 (MILES)
20	0.5	>0.5
2	0.15	>0.5
0.5	-	≅0.5
0.2	<0.1	0.35
0.1	<0.1	0.18

the assumption of <u>total</u> <u>fuel</u> <u>meltdown</u> with the associated assumed fractional release of radioiodines and noble gases to the reactor cell atmosphere indicated in Section 15.4.1.4. However, the assumption of total fuel meltdown is unrealistic for the UFTR under any credible accident conditions with current safety requirements. In addition, the UFTR is not likely to operate to equilibrium radionuclide concentrations at 100 KWth power levels. As previously discussed, calculations with the RIBD code indicate that, if the operation is assumed to run in cycles with 8 hours of operation followed by 16 hours of shutdown in each cycle, the equilibrium I-131 inventory is reduced to less than 50% of the equilibrium inventory for continuous operation. Therefore, it is concluded that the high doses resulting from the calculations presented here are extremely conservative and unlikely to occur under any circumstances; and specifying the distance to Shands Hospital complex as the urban boundary is therefore considered both reasonable and very conservative.

APPENDIX 15A

ENERGETIC EFFECTS OF A LARGE REACTIVITY ADDITION

15A. ENERGETIC EFFECTS OF A LARGE REACTIVITY ADDITION(2)

In Appendix 15B of the UFTR Hazards Summary Report, a comparision between the Boral I and Borax II reactors and the UFTR with respect to the effects of a rapid reactivity insertion is studied. The characteristics of the UFTR which determine its behavior during power transients resulting from large reactivity additions are similar to, but not identical with, those of the Borax I reactor.

In the Borax I experiment, it was found that a 41 MW-sec power excursion due to a rapid reactivity insertion was required to raise the temperature of the fuel plates to the melting point of aluminum, 1220 °F. The characteristics of the reactor which determine the maximum tolerable reactivity insertion rate before fuel melting occurs is shown to be:

1. The coolant channel thickness.

2. The void coefficient of reactivity.

A comparision between coolant channel thickness and void coefficients for the UFTR and the Borax I reactor is made, resulting in a conservative estimate of the limiting non-melting power excursion rate of 35 MW-sec for the UFTR. This would be an accurate estimate of the maximum excursion rate for the UFTR if the fuel plate thickness for both reactors was the same. The fuel plates in the UFTR are thicker which allows for more energy storage during a transient, however. The increased ability to absorb energy depends on the following ratio:

$$\frac{\left(\frac{\text{Heat Flux}}{\Delta T_{c-s}}\right)_{UFTR}}{\left(\frac{\text{Heat Flux}}{\Delta T_{c-s}}\right)_{Borax}} = 0.82 \qquad (15A-1)$$

where ΔT_{c-s} is the temperature difference between the center and surface of the fuel plate. The ratio of peak to average flux for the two reactors

$$\frac{\left(\frac{\text{Peak}}{\text{Avg}}\right)_{Borax}}{\left(\frac{\text{Max}}{\text{Avg}}\right)_{UFTR}} = \frac{1.82}{1.63} = 1.12 \qquad (15A-2)$$

is also taken into consideration when calculating the permissible excursion rate for the UFTR as 35 MW-sec x 0.82 x 1.12 = 32 MW-sec. The corresponding exponential period for a 32 MW-sec excursion is 8.3 milliseconds. It is therefore concluded that the UFTR will tolerate a power excursion of period at least as short as 8.3 millisec, without the melting of any part of any fuel plate. The excess reactivity corresponding to this period is 2.4% $\Delta k/k$. However, the UFTR has a total excess reactivity of only 2.3% $\Delta k/k$ available.

APPENDIX 15B

ESTIMATION OF EFFECTS OF ASSUMED LARGE REACTIVITY ADDITIONS

15B. ESTIMATION OF EFFECTS OF ASSUMED LARGE REACTIVITY ADDITIONS(2)

15B.1 Introduction

It has been demonstrated repeatedly in the Borax and SPERT reactors that water-cooled, water-moderated reactors of suitable design may have a very substantial self-protection against the effects of reactivity accidents, even in the absence of corrective action by the reactor control system. This self-protection is provided by the negative steam-void coefficient of reactivity and the corresponding negative reductions in reactivity as the reactor power rises. The UFTR has been designed with a high degree of self-protection of this negative reactivity feedback type. In this appendix estimates are presented of the behavior of the UFTR under various hypothetical conditions of excess reactivity addition with no corrective action by the control sytem.

The characteristics of the UFTR which determine its behavior during power transients resulting from large reactivity additions are quite similar to, but not identical with, those of he Borax I reactor. UFTR behavior is predicted most reliably by utilizing the Borax I data with simple correction factors to convert them to the UFTR conditions.

The significant quantitative characteristics of the UFTR and the Borax 1 reactor are compared in Table 15B-1.

The U-238 in the 93 percent enriched fuel of the UFTR introduces a small negative Doppler coefficient of reactivity estimated to be of the order of 4×10^{-7} $\Delta k/k/$, equivalent to .004 percent reduction in k per 100 °C rise in fuel temperature. Although the Doppler coefficient acts nearly instantaneously to assure shutdown of the reactor in case of a reactivity accident, its shutdown effect is not expected to be important because expulsion of the water moderator terminates an excursion before the fuel temperature has risen appreciably. Mostly, the water will be ejected though the rupture disk in the primary coolant dumpline.

In addition to the quantitative differences, the UFTR differs from Borax 1 in that the maximum coolant water level is only a few inches above the upper ends of the fuel plates (instead of about 4 ft) and the coolant water, once it has been ejected forcibly from the core by a power excursion, cannot fall or flow back into the core.

15B.2 Effect of 0.6 Percent Excess Reactivity

An excess reactivity of 0.6% $\Delta k/k$ may possibly be inserted in the UFTR reactor through removal of a non-secured or movable experiment. The addition of this reactivity would cause the reactor to operate at a power such that the reactivity losses associated with the temperature increase and the voids formed will equal the excess reactivity. If the reactivity is added slowly, after the reactor is critical, the power approaches such an equilibrium level slowly as the reactivity is added. If the reactivity is added suddenly when the reactor is initially subcritical or at very low power, the power will at first rise exponentially with a period not shorter than 0.8 sec which is the asymptotic period corresponding to the full excess reactivity of 0.6% $\Delta k/k$. Many

Table 15B-1

COMPARISION OF UFTR AND BORAX 1 CHARACTERISTICS

Characteristic	UFTR	Borax 1
Fuel plate "meat"	46 w/o U-Al alloy (93 percent enriched)	18 w/o U-Al alloy (fully-enriched)
Fuel plate cladding	1100 aluminum	1100 aluminum
"Meat" thickness	0.040 in.	0.020 in.
Cladding thickness	0.015 in.	0.020 in.
Coolant-channel thickness	0.137 in.	0.117 in.
Core volume (approx.)	71 liters	106 liters
Void coefficient of reactivity (calculated)	-0.21% $\Delta k/k/\%$ coolant void	-0.24% $\Delta k/k/\%$ coolant void
Temperature coefficient of reactivity (room temperature)	-0.01% $\Delta k/k/$ °C (estimated)	-0.01 %$\Delta k/k/$ °C
Effective prompt-neutron lifetime	1.4×10^{-4} sec (calculated)	0.65×10^{-4} sec
Power Peaking Factor in core (maximum/average)	1.63	1.82

experiments with the Borax reactors have demonstrated that for periods of this order of magnitude, the transition from the exponential power rise to the equilibrium power level (in which excess reactivity is balanced by temperature and steam void coefficients) is a smooth one involving little or no power overshoot. On the basis of this experience, it is concluded tha the magnitude of the power excursion which would result from the 0.6 percent reactivity addition, as related to experiments, does not depend greatly on whether the reactivity is added suddenly or relatively slowly in neither case will it approach a level which can cause a fuel plate to burn out or melt.

In order to compute the power level at which the reactor will operate after the addition of the 0.6 percent excess reactivity discussed in the foregoing, it is necessary to know the water-temperature coefficient of reactivity. The relative importance of the two moderators, graphite and water, in determining the effective neutron temperature introduces uncertainties in the theoretical computation of this coefficient. The coefficient cannot, however, have an absolute magnitude less than that of the water-density coefficent of reactivity referred to a temperature scale, i.e., the coefficient computed on the assumption that:

$$\frac{dk_{eff}}{dT} = \frac{\delta k_{eff}}{\delta \rho} \cdot \frac{\delta \rho}{\delta T} \qquad (15B\text{-}1)$$

where ρ is the water density and T is the temperature and $\delta\rho/\delta T$ is the negative of the void coefficient of reactivity. On the assumption that this minimum value is correct, a rise of water temperature from near 0°C to 80°C would reduce reactivity by 0.6% $\Delta k/k$.

The capacity of the reactor-coolant system is such that if the outside air temperature were 0°C and the average water temperature in the reactor were 80°C, energy would be removed at the rate of 365,000 BTU/hr or 107 KWth. Under these conditions the reactor water-inlet temperature would be 60 °C and the exit temperature coincidentally, would be 100 °C. It is, therefore, concluded that if the full available excess reactivity of 0.6% $\Delta k/k$ were added to the reactor on a cold day with the coolant system operating, 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. Before reaching the equilibrum power, when the water in the coolant system would be heated to the equilibrium value, the reactor would operate at a somewhat higher power level and some net steam production might occur. If the coolant were not flowing during the time of excess reactivity addition, the equilibrium power level would be quite low and equal to the heat losses. In no case would the power level approach a value high enough to justify any fear of fuel-plate burnout.

15B.3 Maximum Tolerable Sudden Reactivity Addition

In order to assess the safety factor which exists between the normal excess reactivity available in the reactor and the excess reactivity necessary for a serious power excursion, it is useful to estimate the value of excess reactivity which, if suddenly inserted and not removed by the control system, would raise the maximum temperature in the hottest fuel plate to the melting point. Such an excursion would damage the reactor core but would not result in any substantial release of fission products.

The first step in the procedure is the estimation of the exponential period corresponding to the excess reactivity which would have characterized a power excursion of similar effect in Borax I. The estimate requires that (1) a relationship be established between the maximum temperature of the fuel plate and the energy release of the excursion and (2) the energy release be related to the period of the excursion.

For the case of power excursions of short period, with reactor water at saturation temperature, it is shown in the original Hazards Summary that the maximum fuel-plate temperature rise is, within experimental error, proportional was determined to be constant 24.4°F per MW-sec. Listing's measurements of the same type with cold reactor water (the case directly applicable to the UFTR) showed a similar relationship but with a proportionality constant of only about 10°F per MW-sec. (22)

The difference is not an unreasonable one since the subcooled water represents a more effective heat sink than the saturated water. However, the experiments with the saturated water were carried to short periods in the range of interest whereas the subcooled experiments were limited to longer periods. Therefore, more conservative saturated water data will be used. To raise the maximum temperature of the fuel plate from the temperature of boiling water to the melting point of aluminum, a temperature change of approximately 1000°F, would require a power excursion with a total energy release of 1000 °F/24.4°F/MW-sec or 41 MW-sec.

According to Listing's data, a "subcooled" power excursion of reciprocal period 150sec^{-1} would give an energy release of 41 MW-sec in addition to the energy necessary to raise the fuel plate temperature to the saturation temperature of water. (22) It is therefore concluded that a power excursion of period at least as short as 1/150 sec (6.7 millisec) could have been tolerated by Borax I with subcooled water without melting at the hottest point in the fuel plates. Actually, the energy data of the original Hazard's Summary Report were revised in Reference 49 because of later and better calibrations of the instrumentation. The numbers above are taken from the later (more conservative) data.

Experiments of the Borax and SPERT types have not been made with reactors having widely different neutron lifetimes. The general evidence of the experiments however supports the supposition that of the three related variables--neutron lifetime, excess reactivity, and exponential period--which characterize the neutron physics of a power excursion, it is the exponential period which determines the total energy release and the temperatures attained. The excess reactivity and neutron lifetime have large effects only as the jointly determine the period. This supposition is consistent, for example, with the observations that the total energy transferred to the coolant water during a power excursion is many times the amount which would vaporize enough water to compensate for the excess reactivity, and that the actual reactivity reduction which occurs during the excursion is much larger than the initial excess reactivity. The extension of the Borax results to the UFTR is made on the basis of this evidence.

It is convenient, first, to treat only the effects of the slightly greater fuel-plate spacing and the slightly lower void coefficient of reactivity of the UFTR relative to the Borax I. Information wil also be drawn from the Borax II experiments. The Borax II reactor differed from Borax I in that the coolant channel thickness was greater in the ratio $\frac{0.264 \text{ in.}}{0.117 \text{ in.}} = 2.26$ and that the calculated void coefficient of reactivity was lower by the following ratio:

$$\frac{0.10\%\ k_{eff}/\%\ \text{void}}{0.24\%\ k_{eff}/\%\ \text{void}} = \frac{1}{2.4} = 0.416 \qquad (15B\text{-}2)$$

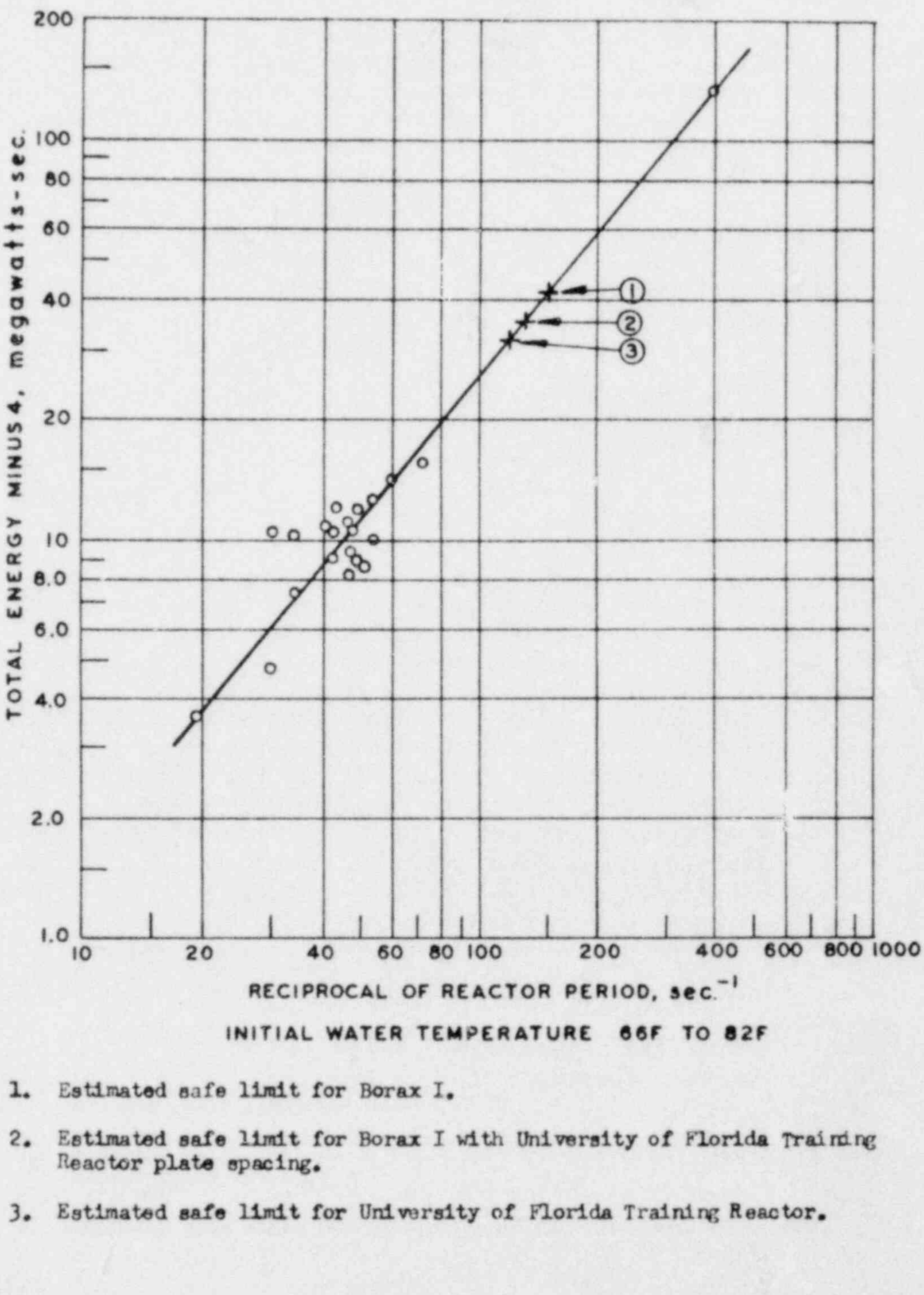
Both of these differences would be expected to cause a higher energy release per fuel plate in Borax II than in Borax I for a power excursion of given period. The measurements made with subcooled water at periods down to 23 millisec showed that the energy release per fuel plate in Borax II was between 1.7 and 2.0 times that of Borax I, with the smaller ratio applying to the shorter periods.(49) Therefore, it seems quite conservative to assume, in the case of any two reactors, (1) and (2), of the Borax type having a ratio of fuel plate spacings, S_1/S_2, and a ratio of void coefficients of reactivity, C_1/C_2, that the ratio of energy release per fuel plate for a subcooled power excursion of given period will be no greater than $E_2/E_1 = S_2/S_1$ or $E_2/E_1 = C_1/C_2$ whichever is larger. For the UFTR and Borax I the ratios are given as follows:

$$\frac{S_{UF}}{S_{Bp}} = \frac{0.137}{0.117} = 1.17 \text{ and } \frac{C_{Bo}}{C_{UF}} = \frac{0.24}{0.21} = 1.14 \qquad (15B\text{-}3)$$

It is concluded, therefore, that a Borax reactor having a coolant-channel thickness and a void coefficient of reactivity equal to those of the UFTR would release not more than 1.17 times as much energy per fuel plate as Borax I. The limiting nonmelting period for such a reactor would be that which in Borax I gave an energy release of 41/1.17=35 MW-sec. The period obtained from Figure 15B-1, corresponding to a total energy release of 35 MW-sec, is 7.7 millisec.

The remaining difference between Borax I and the UFTR is in the composition of the fuel plates. The UFTR plates are thicker; their uranium-aluminum alloy has a somewhat lower conductivity because of the higher uranium concentration, and their aluminum cladding is thinner.

In comparing the behavior of different fuel plates, it must be recognized that the total energy release of the power excursion can no longer be considered as a definitive variable because a large fraction of the total energy released is stored in the fuel plate during the



1. Estimated safe limit for Borax I.

2. Estimated safe limit for Borax I with University of Florida Training Reactor plate spacing.

3. Estimated safe limit for University of Florida Training Reactor.

Total Energy of Excursion Minus Energy Required (4 mw-sec) to Raise Temperature of Center of Average Plate to Boiling Point. (from Fig. 41 AECD-3668, Appendix F. Ref. F-3)

Figure 15B-1. Excursion Energy for Reactor Transients important stage of the reactor shutdown. For example, a reactor composed of fuel plates of high heat capacity undoubtedly will experience a larger total energy release, but not necessarily a higher maximum temperature, during a power excursion of given period, than a reactor having plates of low heat capacity.

From examination of the Borax results, it seems clear that two distinct phases of the reactor shutdown process occur consecutively and that both may be important in determining the maximum center temperature of a fuel plate. The first phase covers the interval before an important amount of boiling occurs at the fuel-plate surface. During this interval the heat loss to the water is small and the important consideration is evidently the ratio of fuel-plate surface temperature (which determines the start of boiling) to center temperature. For periods in the range under consideration, this temperature ratio is theoretically not far from unity (0.76 minimum for a 10-millisec period in Borax I). Experimentally the temperature ratio was unity for periods down to 5 millisec in the Borax I measurements. Since the total effect is small and the temperature ratio for Borax and UFTR fuel plates should not be much different, the thinner cladding will tend to balance the effect of the poorer "meat" conductivity. It is concluded, therefore, that there will be no important difference in fuel plate performance during this initial phase of the excursion.

The second phase of the power excursion begins when a significant rate of boiling is established at the plate surface. Reactivity and consequently generation are reduced at a rate which must be a function of the rate at which heat can be transferred into the boiling water. At the same time the transfer of heat into the water removes heat from the fuel plate and limits its temperature rise. The important characteristic of the plate during this phase of the excursion is the heat flux which it can supply to the water for a given temperature difference between theplate center and surface. A figure assumed to be roughly indicative of the relative performance or merit of fuel plates during this phase is the ratio of heat flux to temperature difference under stead-state conditions. This ratio (figure of merit) will overemphasize the difference between fuel plates since the temperature distribution in the plate will be more peaked during a steady-state conduction than during conduction when the general temperature level is rising. The ratio of these figures of merit for Borax I and the UFTR is

$$\frac{\left(\frac{\text{Heat Flux}}{\Delta T_{c-s}}\right)_{UFTR}}{\left(\frac{\text{Heat Flux}}{\Delta T_{c-s}}\right)_{Borax}} = 0.82 \qquad (15B\text{-}4)$$

A conservative procedure would be to apply the above factor to the permissible total energy of excursion on the Borax I curve. At the same time, however, the difference in gross maximum to average power ratio for the two reactors should be taken into account since it is the temperature of the hottest point in the hottest fuel plate which is being considered. The power ratio for the two reactors is

$$\frac{\left(\frac{\text{Max}}{\text{Avg}}_{\text{Borax}}\right)}{\left(\frac{\text{Max}}{\text{Avg}}\right)_{\text{UFTR}}} = \frac{1.82}{1.63} = 1.12 \qquad (15B\text{-}5)$$

The combination of these two factors reduces the permissible equivalent energy of the Borax-type excursion to 32 MW-sec based upon multiplicative combination of these factors: 35 x 0.82 x 1.12 = 32 MW-sec. The corresponding exponential period from Figure 15B-1 is 8.3 millisec. It is, therefore, concluded that the UFTR will tolerate a power excursion of period at least as short as 8.3 millisec without the melting of any part of any fuel plate. The excess reactivity corresponding to this period is 2.4% $\Delta k/k$.

15B.4 Successive Power Excursions

It is typical of the Borax and SPERT reactors that, unless the excess reactivity is removed by external means, an initial power excursion which terminates itself by expelling water from the reactor core will be followed by subsequent excursions as the water falls and flows back into the core. An exception to this behavior occurs when the initial excursion is violent enough to cause a permanent loss of reactivity by throwing a large amount of water completely out of the reactor tank. In the UFTR the total quantity of water in the core is small, the submergence of the core is small, and the water will be forcefully expelled from the core through the rupture disk in the dumpline. Consequently, even a relatively mild power excursion (e.g., one having an exponential period of from 20 to 30 millisec) in the UFTR should result in permanent self- induced shutdown of the reactor. By this same design feature, the possibility of large successive power excursions, such as those studied in the SPERT project, resulting from the ramp addition of excess reactivity, is eliminated. It can be anticipated that the UFTR is safe against quite large ramp additions (larger than 2.4% $\Delta k/k$) provided only that the ramp rate is not so rapid as to add an excess reactivity of more than 2.4% $\Delta k/k$ before the reactor power reaches a high level. To exceed this limit the ramp rate would need to be of the order of 1.0% $\Delta k/k$ per second or larger. Applicable references from the original Hazards Summary are 50, 51 and 52.

APPENDIX 15C

LOSS OF COOLANT ACCIDENT

15C. LOSS OF COOLANT ACCIDENT(22)

An increase in the UFTR operating power necessitates an investigation of fuel temperatures following a loss of coolant and shut-down of the reactor either by the negative void coefficient of reactivity or by mechanical insertion of the blades. It should be emphasized here that insertion of the control blades is not necessary, and the reactor will shut itself down simply due to the moderator void effect.

Wagner (Appendix 15D) has investigated the problem described above using a rather pessimistic heat transfer model and has concluded that fuel plate temperatures will increase only about 30 °F following a dump scram event.(23)

To represent the decay heat generation following a reactor shutdown, the empirical Equation (15C-1) of Shure and Dudziak is used as follows: (53)

$$\frac{P}{P_{irr.}} = 0.005\ [a_1 \tau_c^{-a_2} - a\ (\tau_1 + \tau\ )_c^{-a_2}\ ] \qquad (15C\text{-}1)$$

where

- τ = irradiation time,
- p = power after decay time, τc
- $P_{irr.}$ = reactor operating power after irradiation and a_1 and a_2 are constants which are given in the original UFTR Hazards Summary.

Wagner (23) further assumed a heat transfer model which ignored radiative and convective heat transfer mechanisms, the only heat loss mechanism being conduction by partial contact of the fuel plates with the fuel box wall. Figure 15-1 shows the fuel plate temperature as a function of time after reactor shutdown due to decay heating effects. Even if the calculation described here is in error by 100 or 200%, which is unlikely considering the conservative approach, the fuel plate temperature rise will not approach temperatures of even half the melting point of alumnium. It must be concluded then, that a loss of cooling flow accident in no way represents a hazard to core structural integrity of the UFTR.

APPENDIX 15D

DECAY HEAT EFFECTS

15D. DECAY HEAT EFFECTS(23)

15D.1 Experimental Verification of the Decay Heat Equation

To represent the decay heat level, P, following a reactor shutdown, the empirical equation of Shure and Dudziak is used, and is given by Equation (15D-1):(53)

$$\frac{P}{P_{irr}} = 0.005\ [a_1(\tau_c)^{-a_2} - a_1(\tau + \tau_c)^{-a_2}] \qquad (15D\text{-}1)$$

where

τ = irradiation time,
τ_c = decay time following shutdown,
P = power after decay time, τ_c
Pirr = reactor opearting power during irradiation,
a_1, a_2 = experimentally determined empirical constants.

The applicability of Equation (15D-1) to the UFTR problem was checked by the following experimental procedure. The reactor was

TABLE 15D-1

CONSTANTS FOR TOTAL FISSION PRODUCT ENERGY RELEASE (53)

Applicable Time Interval (Sec)	a_1	a_2	Maximum Positive Deviation (%)	Maximum Negative Deviation (%)
0.0 to 10	12.05	0.0639	4	3
10.0 to 150	15.31	0.1807	3	1
150.0 to 8×10^8	27.43	0.2962	5	6

operated at 100 KWth for 13 hours and then shutdown, with the secondary coolant water shut off. The primary coolant ΔT_c across the core was then observed and compared to the ΔT_c calculated from a primary coolant heat balance, using the decay heat source given in Equation (15D-1) and also including the heat added by the primary coolant pumps. Results indicate that, at approximately 600 seconds after shutdown, the primary coolant inlet and outlet temperatures are stablized at 130.1°F and 130.5°F respectively, corresponding to ΔT_c=0.4°F. Assuming 90 percent of the 0.72 KW electrical power applied to the primary pump is dissipated as heat to the primary coolant, and considering the decay heat as given by Equation (15D-1) with τ = 43,800 seconds and τ_c = 600 seconds, the primary coolant ΔT_c is computed to be ΔT_c = 0.48°F. This result for the temperature rise across the core agrees relatively well with the experimentally measured values of ΔT_c despite possible recording errors in the coolant temperature readings, and provides some degree of verification for the applicability of Equation (15D-1) to this situation.

15D.2 Simulation of Heat Flow Following a Dump Trip

An increase in the UFTR operating power necessitates an investigation of decay heating of the fuel following a loss of coolant, that is, a dump-scram situation. The magnitude of such an effect is estimated by using a conservative model which assumes that the UFTR has been operated to equilibrium at 625 KWth (over 6 times rated power) and then an instantaneous dump-trip occurs. The fuel plate temperature as a function of time after the dump-scram is then calculated by simulating the heat flow in the core region.

Equation (15D-1) gives the rate of emission of beta and gamma energy; however, not all gamma rays are absorbed in the fuel boxes. The fraction of gamma rays absorbed in a fuel box (a center fuel box of dimensions: 5¼" x 6¼" x 48", and weighing 13.9 lb., is considered for all calculations) is arrived at by treating the box as a homogenized cylinder, allowing volume weighted mass absorption coefficients for seven gamma energy groups to be determined. Using equations 6.4 -32 and 6.4 -30 of Reference 34 (p. 382), the gamma fluxes are respectively calculated at the center of the cylindricized fuel box and at the exterior midplane point on the side of the box. With these values and the equilibrium fission product decay gamma spectrum as given by Reference 54, 63.6% of the gamma energy is found to be absorbed in the center fuel box. Since about 53 percent of the total decay energy is in the form of gamma rays (55), and estimating the radiation from the adjacent two fuel boxes, 88.1% of the total fission product decay energy is computed to be absorbed in the center fuel box.

Within each fuel box the fuel plates are aligned along the east-west direction. A center pin is forced in the center of the four fuel bundles which pushes the fuel plates in contact with the east and west side of the fuel box. The center fuel box itself touches graphite which lies on the north and south sides. Thus the heat produced within a fuel plate flows through its edges (25.625" x 0.07" side) to the fuel box (QPLAT) and from the box to graphite (QBOX), which is assumed at a constant temperature. Convective and radiative heat losses are negligible. The center pin which forces the fuel plates to the sides of the fuel box is applied only at the top of the bundles; therefore, contact at points towards the bottom of the fuel box may be small and possibly nonexistent. The thermal resistance for QPLAT is thus determined by assuming 50% Al-Al contact and 50% separation by a 1/32" thick air wall. The average unit thermal conductance for an Al-Al contact at a contact pressure of 14.223 psi is 634.25 Btu/hr-°F (56) which gives,

$$\text{QPLAT} = 1.109 \times 10^{-3}\ (\text{TPLAT} - \text{TBOX})\ \text{Btu/sec-}^{\circ}\text{F} \qquad (15\text{D-}2)$$

where

TPLAT = temperature of the fuel plate (°F), and
TBOX= temperature of the fuel box (°F).

The thermal resistance for QBOX is determined by also considering a 1/32" thick wall separating the fuel box and the graphite, with 50% of the area comprised of air and 50% of graphite. This gives,

$$QBOX = 1.799 \times 10^{1} (TBOX - TGRAP) \text{ Btu/sec -}^{\circ}F \qquad (15D\text{-}3)$$

where TGRAP = temperature of the graphite (°F)

There are a total of 230 fuel plates in the UFTR, each "ideally" weighing 0.52771 lb. This simulation conservatively assumes that each of the 44 fuel plates in the center fuel box was operating at a hot-channel factor of 1.5 which gives a power output of 3.864 Btu/sec/plate for a 625 KWth power level. Using equations (15D-1) and (15D-3), the heat production rate within a plate and the heat flow rates in the core can be found, allowing the determination of the fuel plate and fuel box temperatures.

The fuel plate temperature versus time after dump-trip, following infinite reactor operation at 625 KWth is shown in Figure 15-1. The fuel plate temperature rise is 26°F and occurs 2800 seconds after the dump-trip initiation. It can be seen that decay heating of the fuel is minimal. The 26°F tempeature rise represents an upper bound for the decay heat effects due to the conservatism of the model.

The actual extent of this conservatism is difficult to establish since experimental support cannot be obtained from the UFTR due to a lack of temperature detection devices on the fuel plates. Thermocouples do exist, however, on the coolant outlet pipes of each fuel box. Measurements at these locations would thus reflect the temperature of the fuel box, rather than the fuel plate, and would be quite low because of heat conduction losses of the box to the graphite. This problem does not exist for the UCLA Argonaut reactor. A short duration, high power dump- trip experiment was investigated for this reactor.(7) It was found that the surface temperature rise of the midpoint of the hottest fuel plate following a dump-trip at 500 KWth was only 14°F. Since the maximum temperature was achieved within one minute after the dump-scram and since prior steady reactor operation at this power level occurred for only 8 minutes, it is assumed that decay heating for this case is negligible. The temperature rise is mainly due to the thermocouple on the surface of the fuel plate climbing to the actual fuel plate temperature rather than remaining at the "average" temperature of the fuel and coolant water. As predicted by the heat flow model, a fuel temperature rise of about 0.04°F, due to decay heating, would result for the UCLA experiment where it is assumed that TPLAT = 197°F, TGRAP = TBOX = 190°F, and τ = 600 seconds.

APPENDIX 15E

RADIATION RELEASES RESULTING FROM
RELEASE OF FISSION PRODUCTS INTO THE
ATMOSPHERE

(Reference 2)

15E. RADIATION RELEASES RESULTING FROM RELEASE OF FISSION PRODUCTS INTO THE ATMOSPHERE(2)

Although a significant fission product release is not considered plausible because of the inherent self-limiting characteristics of the UFTR, it is informative to calculate exposure rates due to the release of fission products following a fuel meltdown. Calculations in the UFTR Hazards Report (See Appendix 15F) are based on reactor operation at 10 KWth long enough to have attained equilibrium concentrations of the relatively short-lived fission products, i.e., the iodine, bromine, and krypton isotopes. The incident is assumed to result in the transfer of 10% of the volatile fission products from the reactor fuel plates to the building air. It is assumed further that none of the non-volatile fission products are transferred to the building air although they may be released to the reactor coolant water and retained within the reactor building. The major avenues of leakage of fission products from the reactor room are through the doors.

Three arbitrary exposures were calculated and presented in the original Hazards Report as follows:

1) The I-131 dose to the thyroid of a person standing at a distance of 61 meters downwind of the leak for 8 hours resulting in a total integrated dose of 155 rem.

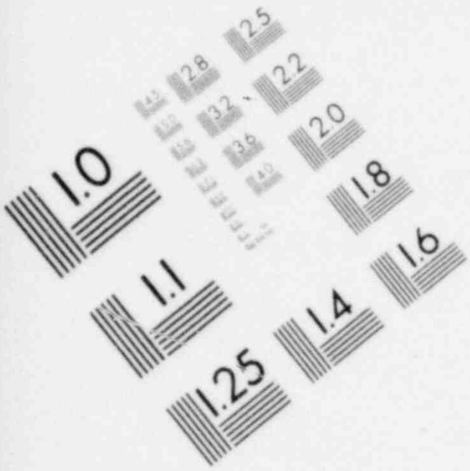
2) The external Beta-dose from the I-131 isotope for a person standing at a distance of 61 meters downwind of the leak for 8 hours resulting in a total integrated dose of 19 mr.

3) The total gamma dose to a person standing 61 meters downwind of the leak for 8 hours resulting in a total integrated dose of 19.2 mr.

Since the exposures calculated are proportional to activity of the fission products, and the concentration of fission products is proportional to reactor flux, the total integrated dose in each case for 100 KWth operation would simply be 10 times the dose at 10 KWth.

APPENDIX 15F

RADIATION DOSES RESULTING FROM RELEASE OF FISSION PRODUCTS INTO THE ATMOSPHERE



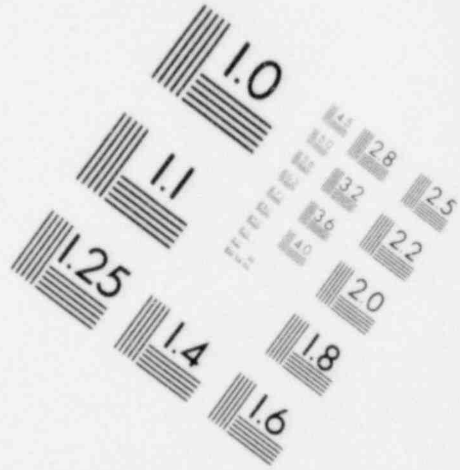
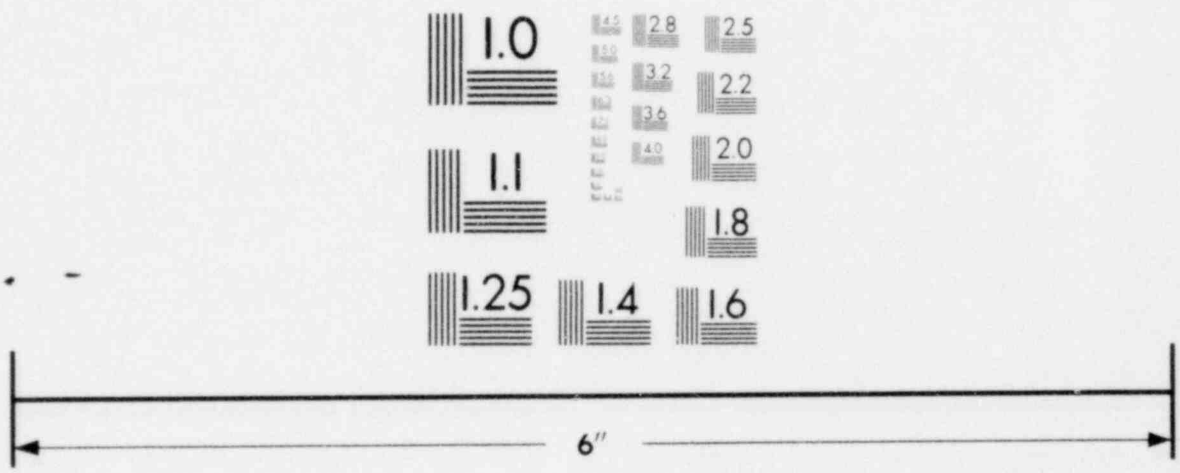
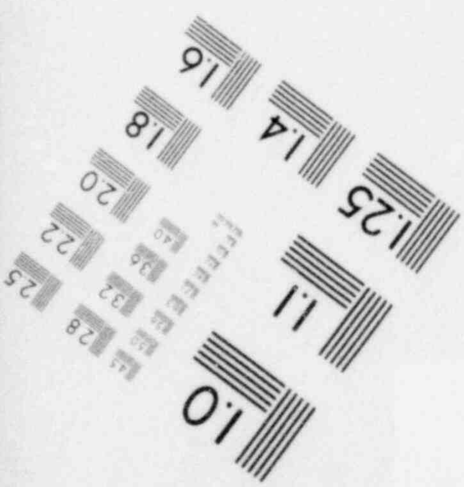
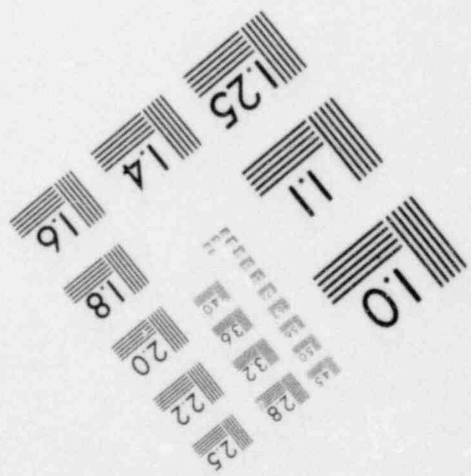


IMAGE EVALUATION TEST TARGET (MT-3)







15F. RADIATION DOSES RESULTING FROM RELEASE OF FISSION PRODUCTS INTO THE ATMOSPHERE (2)

Estimates have been made of the radiation doses which would be received by persons outside the reactor building should there be a release of reactor fission products into the reactor building and leakage of the building air to the outside. The radiation exposures considered here are those which would result from the passage of the airborne cloud of radioactive contaminants over the ground. These include the external beta and gamma radiation exposures and the internal exposure of critical body organs resulting from inhalation of the airborne contaminants. The most important of the internal exposures are the iodine dose to the thyroid and the strontium dose to the bones.

The radiation exposure received by a person standing a given distance from the reactor building obviously depends on such factors as (a) curies of fission products stored within the core at the time of release, (b) fraction of the core fission products escaping into the building air, (c) building out-leakage rate, and (d) atmospheric dispersive properties. Hence, in the analysis, certain basic assumptions are required as to the circumstances surrounding the release of the fission products, as to atmospheric conditions and, as to the tightness of the buiding at the time of release. The results obtained here are based on assumptions which, except for the arbitrary one that a release has occurred, are considered reasonable for the reactor and building design. The calculation method is described and illustrated in sufficient detail that additional calculations based on other assumptions can be made if desired as shown in th original UFTR Hazards Summary.(2)

The material presented in the Hazard's Summary is divided into three sections. The first section describes the model assumed for for the release and spread of radioactivity and gives the necessary references and formulae used in calculating the radiation doses. The second section illustrates the calculational procedure. The third section presents the results obtained for the radiation exposure hazards with the assumed model. The details on the original dose calculations can be found on pages 114-119 of the original Hazards Summary while the results of the radiation exposure calculations are summarized in Table 15F-1.

Table 15F-1

TOTAL INTEGRATED DOSE (rep) FROM AN 8-HR EXPOSURE
AT VARIOUS DISTANCES DOWNWIND FROM REACTOR BUILDING LEAK (2)

x, meters	External Beta Dose	Gamma Dose	Thyroid Dose	Bone Dose
		Severe Inversion		
15	14	.080	1800	.006
61	1.6	.019	220	.0007
152	.4	.010	59	.0002
305	.15	.005	20	---
		Mild Lapse		
15	2.2	.040	290	.001
61	.19	.007	26	.0001
152	.04	.004	6	----
305	.012	.002	2	----

16. TECHNICAL SPECIFICATIONS

Section 50.36 of 10 CFR Part 50 requires that each operating license issued by the Nuclear Regulatory Commission contain Technical Specifications that set forth the limits, operating conditions, and other requirements imposed on facility operation for the protection of the health and safety of the public and other reasons. The UFTR established Technical Specifications on July 22, 1970, according to Amendment 10 to Facility License R-56.

The new Technical Specifications (TS) for the UFTR included in this FSAR are an upgrading of the first set of TS to better satisfy NRC requirements, the ANS-15.1 Standard for the Development of Technical Specifications for Research Reactors and to better describe and establish limits for the facility safety-related and overall capabilities.

Many of the design bases for setpoints and trip-points are historical for the UFTR and other Argonauts. Revisions of these specifications have been made to match and upgrade previous modifications and present capabilities of the UFTR systems. For example, the primary coolant flow trip-point has been historically set at 18 gpm and proven sufficiently conservative safety-wise; however, the new UFTR setpoint for primary coolant flow has been 30 gpm for the past five years (after improvements to the reactor coolant system) and is therefore established as the proposed TS for coolant flow.

The selection of the UFTR trip-points follows a conservative and practical approach to the operational safety of the UFTR, often without the liberalization of margin/setpoints resulting from detailed analysis and measurements. Operating experience demonstrates that the historical and upgraded setpoints will maintain fuel and coolant temperatures well within stringent safety limits.

The definition of 1.1.5 of Abnormal Occurrences in the new TS deserves an explanation. This definition is intended to address specifically those occurrences which have safety significance, or could lead the reactor to be operated in violation of a Limiting Safety System Setting or in violation of a Limiting Condition for Operation. In this regard, occurrences affecting the reactivity of the reactor which are due to the expected and proper functioning of the Control and Safety System are not considered to be "an uncontrolled or unanticipated change in reactivity." Therefore, the following situations are accepted as normal regarding reactivity insertions:

- Reactor trips caused by loss of power to the reactor console or to any component of the Control and Safety System when the systems respond as specified.

- Reactor trips caused by operator or student operator-in-training or induced by fail-safe components when the Reactor Safety System is performing its intended function.

- The controlling actions of an operator or student operator-in-training and occurrences which do not result in Safety System actuation and do not violate Limiting Conditions for Operation.

The new UFTR Technical Specifications are improved in the areas of Limiting Conditions for Operation, surveillance and reporting requirements.

Technical Specifications

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TECHNICAL SPECIFICIATIONS FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR

1.0 DEFINITIONS

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

1.1 Abnormal Occurrences--An abnormal occurrence is any one of the following:

1.1.1 Operating the reactor with a safety system setting less conservative than specified in the Limiting Safety System Setting's section of the Technical Specifications.

1.1.2 Operating the reactor in violation of a Limiting Condition for Operation.

1.1.3 A malfunction of a Safety System component or other component or system malfunction which could, or threaten to, render the system incapable of performing its intended safety function.

1.1.4 A release of fission products from the reactor fuel of a magnitude to indicate a failure of the fuel cladding.

1.1.5 An uncontrolled or unanticipated change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded.

1.1.6 An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor.

1.1.7 An uncontrolled or unanticipated release of radioactivity to the environment.

1.2 Channel Calibration--A channel calibration is an adjustment of the channel components such that its output responds, within specified range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including readouts, alarm or trip.

1.3 Channel Check--A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with other independent channels or methods of measuring the same variable.

1.4 Channel Test--A channel test is the introduction of an input signal into the channel to verify that it is operable.

1.5 Independent Experiments--Experiments not connected by a mechanical, chemical or electrical link.

1.6 Inhibit--An inhibit is a device which prevents the withdrawal of control blades under a potentially unsafe condition.

1.7 Measured Value--The measured value of a parameter is the value as it appears at the output of a measuring channel.

1.8 Measuring Channel--A measuring channel is the combination of sensor, lines, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable.

1.9 Movable Experiment--A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating, or have moving in-core components during operation.

1.10 Non-secured Experiments--Experiments where it is intended that the experiment should not move while the reactor is operating, but is held in place with less restraint than a secured experiment.

1.11 Operable--A system or component is operable when it is capable of performing its intended function in a normal manner.

1.12 Operating--A system or component is operating when it is performing its intended function in a normal manner.

1.13 Reactor Operating--The reactor is considered to be operating whenever it is not secured or shut down.

1.14 Reactor Safety System--The Reactor Safety System is that combination of measuring channels and associated circuitry which forms the automatic protective action to be initiated or provides information which requires the initiation of manual protective action.

1.15 Reactor Secured--The reactor is secure when:

1.15.1 It contains sufficient fissile material or moderator present in the reactor, adjacent experiments or control rods, to attain criticality under optimum available conditions of moderation and reflection,

or

1.15.2 A. The reactor is shutdown, and
B. Electrical power to the control blade circuits is switched off and switch key is in propoer custody, and
C. No work is in progress involving core fuel, core structure, installed control rods or control rod drives unless they are physically decoupled from the control rods, and
D. No experiments are being moved or serviced that have, on movement a reactivity worth exceeding the maximum value allowed for a single experiment or one dollar whichever is smaller.

1.16 Reactor Startup--A series of operator manipulations of reactor controls - in accordance with approved procedures - intended to bring the reactor to a k_{eff} of 0.99 or greater. It does not include control blade manipulations made for purposes of testing equipment or component operability within a k_{eff} of 0.99 or less.

1.17 Reactor Shutdown--The reactor is shutdown when all control blades are inserted and the reactor is subcritical by a margin greater than 2% $\Delta k/k$. When calculating the subctitical margin, no credit shall be taken for experiments, temperature effects or xenon poisoning.

1.18 Reactor Trip--A reactor trip is deemed to occur whenever one of the two following actions take place:

1.18.1 Rod-Drop Trip: a gravity drop of all control blades into the reactor core as a result of terminating electrical power to the blade drive magnetic clutches.

1.18.2 Full Trip: the water is dumped from the reactor core by the safety-actuation of the dump valve in addition to the Rod Drop Trip.

1.19 Reportable Occurrence--A reportable occurrence is any of the conditions described in Section 6.5.2 of this specification.

1.20 Research Reactor--A device designed to support a self-sustaining neutron chain reaction to supply neutrons or ionizing radiation for research, developmental, educational, training or experimental purposes, and which may have provisions for the production of non-fissile radioisotopes.

1.21 Rod Drop Time--Rod Drop Time is the elapsed time between the instant a limiting safety system set point is reached or a manual scram is initiated and the instant that the rod is fully inserted.

1.22 Safety Channel--A safety channel is a measuring channel in the reactor safety system.

1.23 Secured Experiment--A secured experiment is a stationary experiment held firmly in place by a mechanical device secured to the reactor structure or by gravity providing that the weight of the experiment is such that it cannot be moved by a force of less than sixty (60) pounds.

1.24 Secured Experiment with Movable Parts--A secured experiment with movable parts is one that contains parts that are intended to be moved while the reactor is operating.

1.25 Shutdown Margin--The minimum shutdown reactivity necessary to provide confidence that the system can be made subcritical by means of the control and safety systems, starting from any permissible operating condition, and that the reactor will remain subcritical, without further operator action.

1.26 Unscheduled Shutdown--An unscheduled shutdown is any unplanned shutdown of the reactor, after startup has been initiated.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Safety Limits for nuclear reactors are limits upon important process variables whcih are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. The principle physical barrier shall be the fuel cladding.

2.1.1 Applicability

These specifications apply to the variables that affect thermal, hydraulic and materials performance of the core.

2.1.2 Objectives

To assure fuel cladding integrity.

2.1.3 Specifications

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

2.1.4 Bases

Operating experiences and detailed calculations at Argonaut reactors have demonstrated that Specifications A and B suffice to maintain the maximum fuel temperature below 200°F, which is well below the temperature where fuel degradation would occur. For the readily available flow rate of up to 65 gpm, it has been shown that the fuel temperature will be well below 200°F for steady state power operation of up to 500 KWth. No fuel damage is known to occur from transient operation up to 500% full power at the present 40 gpm primary flow rate. Specification C is included to prevent boiling of the primary coolant at any fuel box. Specification D suffices to maintain adequate water quality conditions to prevent deterioration of the fuel cladding and still allow for expected transient changes in the water resistivity.

2.2 Limiting Safety System Settings

Limiting Safety System Settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions.

2.2.1 Applicability

These specifications are applicable to the Reactor Safety System setpoints.

2.2.2 Objectives

To insure that automatic protective action is initiated prior to exceeding a Safety Limit or prior to creating a radioactive hazard which is not considered under Safety Limits.

2.2.3 Specifications

The Limiting Safety System Settings shall be as follows:

A. Power level at any flow rate shall not exceed 125 KW,
B. The primary coolant flow rate shall be greater than 30 gpm at all power levels greater than one watt,
C. The average primary coolant outlet temperature shall not exceed 155°F when measured at any fuel box outlet,
D. The reactor period shall not be faster than 3 seconds,
E. The high voltage applied to Safety Channels 1 and 2 neutron chambers shall be 90% or more of the established normal value;
F. The primary coolant pump shall be energized during reactor operations,
G. The primary coolant flow rate shall be monitored at the return line,
H. The primary coolant core level shall be at least 2" above the fuel boxes,
I. The secondary coolant flow shall satisfy the following conditions when the reactor is being operated at power levels equal to or larger than 1 KW:

   (a) Power shall be provided to the well pump and the well water flow rate shall be larger than 60 gpm when using the Well System for secondary cooling

   or

   (b) The water flow rate shall be larger than 8 gpm when using the City Water System for secondary cooling,

J. The reactor shall be shutdown when the Main Alternating Current (A.C.) power is not operating,
K. The Reactor Vent System shall be operating during reactor operations,
L. The water level in the Shield Tank shall not be reduced 6" below the established normal level.

2.2.4 Bases

The UFTR Limiting Safety System Settings (LSSS) arise from operating experience and safety considerations. The LSSS 2.2.3 A through J are established for the protection of the fuel, the fuel cladding and the reactor core integrity. The primary and secondary bulk coolant temperatures as well as the outlet temperatures for the six fuel boxes are monitored and recorded in the Control Room. LSSS 2.2.3 K is established for the protection of reactor personnel in relation to accumulation of Argon-41 in the Reactor Cell and for the control of radioactive gaseous effluents from the Cell. LSSS 2.2.3 L is established for the protection of reactor personnel from potential external radiation hazards caused by loss of biological shielding.

3.0 LIMITING CONDITIONS FOR OPERATION

Limiting conditions for operation are the lowest function capability or performance levels of equipment required for safe operation of the facility.

3.1 Reactivity Limitations

3.1.1 Shutdown Margin--The minimum shutdown margin, with the most reactive control blade fully withdrawn shall not be less than 2% $\Delta k/k$.

3.1.2 Excess Reactivity--The core excess reactivity at cold critical, without xenon poisoning, shall not exceed 2.3% $\Delta k/k$.

3.1.3 Coefficients of Reactivity--The primary coolant void and temperature coefficients of reactivity shall be negative.

3.1.4 Maximum Single Blade Reactivity Insertion Rate--The reactivity insertion rate for a single control blade shall not exceed 0.06% $\Delta k/k$/sec, when determined as an average over any ten seconds of blade travel time from the characteristic experimental integral blade reactivity worth curve.

3.1.5 Experimental Limitations--The reactivity limitations associated with experiments are specified in 3.5 below.

3.1.6 Bases

These specifications are provided to limit the amount of excess reactivity to within limits known to be within the self-protection capabilities of the fuel; to assure that a reactor shutdown can be established with the most reactive blade out of the core; to assure a negative overall coefficient of reactivity and to limit the reactivity insertion rate to levels commensurable with efficient and safe reactor operation.

3.2 Reactor Control and Safety Systems

3.2.1 Reactor Control System

A. Four cadmium-tipped, semaphore-type blades shall be used for reactor control. The control blades shall be protected by shrouds to assure freedom of motion.
B. Only one control blade can be raised by the manual reactor controls at any one time. The Safety Blades shall not be used to raise reactor power simultaneously with the Regulating Blade when the Reactor Control System is in the Automatic Mode of Operation.
C. The reactor shall not be started unless the Reactor Control System is operable.
D. The Control Blade drop time shall not exceed 1 second from initiation of blade drop to full insertion (rod drop time), as determined according to Surveillance Requirements.
E. The following Control Blade Withdrawal Inhibit Interlocks shall

be operable for reactor operation for the following conditions:

(a) a source (startup) count rate of less than 2 cps (as measured by the Wide Range Drawer operating on Extended Range),
(b) a reactor period less than 10 seconds,
(c) Safety Channel #1 and #2, and Wide Range Drawer Calibration Switches not in "OPERATE" condition,
(d) attempt to raise any two or more blades simultaneously when the reactor is in Manual Mode, or two or more Safety Blades simultaneously when the reactor is in Automatic Mode,
(e) power is raised in the Automatic Mode at a period faster than 30 seconds. The Automatic Controller action is to inhibit further Regulating Blade withdrawal or drive the Regulating Blade down until the period is ≥ 30 seconds.

F. Following maintenance or modification to the Reactor Control System, an operability test and calibration of the affected portion of the system, including verification of control blade drive speed, shall be performed before the system is considered operable.

3.2.2 Reactor Safety System

A. The reactor shall not be started unless the Reactor Safety System is operable in accordance with Table I.
B. Tests for operability shall be made in accordance with Table II.

3.2.3 Reactor Control and Safety Systems Measuring Channels

The minimum number and type of measuring channels operable and providing information to the control room operator required for reactor operation are given as follows:

Channel	No. Operable
Safety 1[a] and 2 Power Channel	2
Linear with Auto Controller	1
Log N and Period Channel [a]	1
Startup Channel [a]	1
Rod Position Indicator	4
Coolant Flow Indicator	1
Coolant Temperature Indicator	
Primary	6
Secondary	1
Core Level	1
Ventilation System	
Core Vent Annunciator	1
Exhaust Fan Annunciator	1
Exhaust Fan rpm	1

(a) Subsystems of the Wide Range Drawer

Table I

SPECIFICATIONS FOR REACTOR SAFETY SYSTEM TRIPS

Automatic Trips

Specification	Type of Safety System Trip
Period less than 3 seconds	FULL
Power at 125% of full power	FULL
Loss of Chamber High Voltage ($\geq$ 10%)	FULL
Loss of Electrical Power to Control Console	FULL
Primary Cooling System Loss of pump power Low water level in core (<42.5") No outlet flow Low inlet water flow (< 30 gpm)	Rod-Drop
Secondary Cooling System (at power levels above 1 kw) Loss of flow (well water < 60 gpm, city water < 8 gpm) Loss of pump power	Rod-Drop
High Primary Coolant Average Outlet Temperature ($\geq$ 155°F)	Rod-Drop
Shield Tank Low water level	Rod-Drop
Ventilation System Loss of power to dilution fan Loss of power to Core Vent System	Rod-Drop

Manual Trips

Specification	Type of Safety System Trip
Manual Scram Bar	Rod-Drop
Console Key-Switch OFF (two blades off bottom)	FULL

Table II

SAFETY SYSTEM OPERABILITY TESTS

Component or Scram Function	Frequency
Log-N Period Channel Power Level Safety Channels	Prior to each reactor start-up following a shutdown in excess of 6 hours, and after repair or de-energization 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	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	With Daily Checkout

3.2.4 Bases

The Reactor Control System provides the operator with reactivity control devices to control the reactor within the specified range of reactivity insertion rate and power level. The operator has available digital blade position indicators for the three Safety Blades and the Regulating Blades. The three Safety Blades can only be manipulated by the UP-DOWN Blade Switches (manual); the Regulating Blade can be manually controlled or placed under Automatic Control, which uses the Linear Channel as the measuring channel, and a % power setting control. The two independent Reactor Safety Channels provide redundant protection and information on reactor power in the range 1% - 150% of full power. The Linear Power Channel is the most accurate neutron instrumentation channel, and provides a signal for reactor control in Automatic Mode. The % power information is displayed by the linear channel 2 pen recorder. It does not provide a protective function. The log Wide Range Drawer provides a series of information, inhibit and protection function from extended source range to full power. The Safety Channel 1 signal and the Period Protection signal are derived from the Wide Range Drawer. The Wide Range Drawer provides protection during startup through the Source Count Rate Interlock (2 cps), 10 second period inhibit and the 3 second period trip. The primary and secondary coolant flow rate, temperature and level sensing instrumentation provides information and protection over the entire range of reactor operations and is proven to be conservative from a safety viewpoint. The key switch prevents unauthorized operation of the reactor and is an additional full trip (manual scram) control available to the operator. The core level trip provides redundant protection to the primary flow trip. The level trip acts as an inhibit during startup until the minimum core water level is reached.

3.3 Reactor Vent System

These specifications apply to the equipment required for controlled release of gaseous radioactive effluent to the environment via the stack or its confinement within the reactor cell.

3.3.1 Specification

A. The Reactor Vent System shall be capable of maintaining an air flow rate between 1 and 400 cfm from the reactor cavity whenever the reactor is operating and as specified in these Technical Specifications. The reactor air cavity flow shall be periodically calibrated to minimize Argon-41 releases to the environment while maintaining a negative pressure within the reactor cavity to minimize radioactive hazards to reactor personnel.

B. The diluting fan shall be operated whenever the reactor is in operation and as otherwise specified in these Technical Specifications, at an exhaust flow rate larger than 10,000 cfm.

C. The Reactor Vent System is interlocked to shut off automatically when the Air Conditioning/Ventilation System is shut-off. The AC/Ventilation System is automatically shut-off whenever the reactor building evacuation alarm is automatically or manually actuated.

D. All doors to the Reactor Cell shall normally be closed while the reactor is operating. Transit is not prohibited through Air Lock and Control Room doors.

3.3.2 Bases

Under normal conditions, to effect controlled release of gaseous activity through the Reactor Vent System, a negative cell pressure is required so that any building leakage will be inward. Under emergency conditions, the Reactor Vent System will be shut down and the damper closed, thus minimizing leakage of radioactivity from the Reactor Cell.

3.4 Radiation Monitoring Systems and Radioactive Effluents

3.4.1 The reactor cell shall be monitored by at least three area radiation monitors, two of which shall be capable of audibly warning personnel of high radiation levels. The output of at least two of the monitors shall be indicated and recorded in the Control Room. The setpoints for the radiation monitors shall be in accordance with Table III.

3.4.2 Argon-41 Discharge

The following operational limits are specified for the discharge of Argon-41 to the environment:

A. The concentration of Argon-41 in the gaseous effluent discharge of the UFTR is determined by averaging it over a consecutive 30 day period.

B. The dilution resulting from the operation of the Stack Dilution Fan (flow rate of 10,000 cfm or more) and atmospheric dilution of the stack plume (a factor of 200) may be taken into account when calculating this concentration.

C. When calculated as above, discharge concentration of Argon-41 shall not exceed MPC ($4.0 \times 10^{-8} \mu c/ml$). Operation of the UFTR shall be such that this maximum permissible concentration (averaged over a month) is not exceeded.

3.4.3 The Reactor Vent System shall be operated at all times during reactor operation. In addition, the Vent System shall be operated until the stack monitor indicates less than 10 counts per second. Whenever the Reactor Vent System is operating, air drawn through the Reactor Vent System shall be continuously monitored for gross concentration of radioactive gases. The output of the monitor shall be indicated and recorded in the Control Room. The Reactor Vent System shall be immediately secured upon detection of: a failure in the monitoring system, a failure of the absolute filter or an unanticipated high stack count rate.

3.4.4 The Reactor Cell environment shall be monitored by at least one Air Particulate Monitor, capable of audibly warning personnel of radioactive particulate airborne contamination in the Cell atmosphere.

Table III

RADIATION MONITORING SYSTEM SETTINGS

Type	No. of Required Operable Functions	Alarm(s) Setting	Purpose
Area Radiation Monitors	3 detecting 2 audio alarming 2 recording	5 mr/hr low level 25 mr/hr high level	Detect/alarm/record low and high level external radiation
Air Particulate Monitors	1 detecting 1 audio alarming 1 recording	Adjusted according to APD type (fixed or movable filter type)	Detect/alarm/record airborne radioactivity in the reactor cell
Stack Radiation Monitor	1 detecting 1 audio alarming 1 recording	1) Fixed alarm at 4000 cps 2) Adjustable alarm as per power level	Detect/alarm/record release of gaseous radioactive effluents in the reactor vent duct to the environs.

NOTE: For maintenance or repair, the required radiation monitors may be replaced by suitable portable instruments provided the intended function is being accomplished.

Service, calibration and testing interruptions for brief period are permissible when the reactor is not in operation.

3.4.5 Liquid Effluents Discharge

A. The liquid waste from the radioactive liquid waste holding tanks shall be sampled and the activity measured before release to the sanitary sewage system.
B. Releases of radioactive liquid waste from the holding tanks/ campus sanitary sewage system shall be in compliance with the limits specified in 10 CFR 20 Appendix B, Table 1, Column 2, as specified in 10 CFR 20.303.

3.4.6 Solid Radioactive Waste Disposal

Solid radioactive waste disposal shall be accomplished in compliance with applicable regulations and under the control of the Radiation Control Office of the University of Florida.

3.5 Limitations on Experiments

3.5.1 Applicability

These specifications apply to all those experiments or experimental devices installed in the reactor core or its experimental facilities.

3.5.2 Objective

The objective is to maintain 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.

3.5.3 Specifications

A. General

The Reactor Manager and the Radiation Control Officer (or their duly appointed representative) shall review and approve in writing all proposed experiments prior to their performance. The Reactor Manager shall refer to the Reactor Safety Review Subcommittee (RSRS) the evaluation of the safety aspects of new experiment and all changes to the facility which may be necessitated by the requirements of the experiments and which may have safety significance. When experiments contain substances which irradiation in the reactor can convert into a material with significant potential hazards, a determination will be made of the acceptable reactor power level and length of irradiation, taking into account such factors as: isotope identity and chemical and physical form and containment, toxicity, potential for contamination of facility or environment, problems in removal or handling after irradiation including containment, transfer and eventual disposition. Guidance should be obtained from the ANS 15.1 Standard. Experimental apparatus, material or equipment to be inserted in the reactor shall be reviewed to insure non-interference with the safe operation of the reactor.

B. Classification of Experiments

Class I--Routine experiments, such as gold foil irradiation. This class shall be approved by the Reactor Manager; the Radiation Control Officer may be informed if deemed necessary.

Class II--Relatively routine experiments which need to be documented for each new group of experimenters carrying them out or whenever the experiment has not been carried out for one calendar year or more by the original experimenter, and which pose no hazard to the reactor, the personnel or the public. This class shall be approved by the Reactor Manager and the Radiation Control Officer.

Class III--Experiments which pose significant questions regarding the safety of the reactor, the personnel, or the public. This class shall be approved by the Reactor Manager and the Radiation Control Officer, after review and approval by the Reactor Safety Review Subcommittee (RSRS).

Class IV--Experiments which have a significant potential for hazard to the reactor, the personnel or the public. This class shall be approved by the Reactor Manager and Radiation Control Officer after review and approval by the RSRS and specific Emergency Operating Instructions shall be established for conducting the experiments.

C. Reactivity Limitations on Experiments

(a) The absolute reactivity worth of any single movable or non-secured experiment shall not exceed 0.6% $\Delta k/k$.
(b) The total absolute reactivity worth of all experiments shall not exceed 2.3% $\Delta k/k$.
(c) When determining the absolute reactivity worth of an experiment, no credit shall be taken for temperature effects.
(d) An experiment shall not be inserted or removed unless all the control blades are fully inserted or its absolute reactivity worth is less than that which would cause a positive 20 second stable period.

D. Explosive Materials

Explosive materials shall not be irradiated

E. Thermal-Hydraulic Effects

The experiment shall be designed such that during normal operation, or failure, the thermal hydraulic parameters of the core do not exceed the Safety Limits.

F. Chemical Effects

The experiment shall be designed such that during normal operation, or failure, the physical barrier described in paragraph 2.1 will not be compromised by either chemical or blast effects from the experiment.

G. Fueled Experiments

A limit should be established on the inventory of fission products in experiments containing fissile material, according to its potential hazard and as determined by the RSRS

H. Radioactive Releases from Experiments

Class III and Class IV experiments shall be evaluated for their potential release of airborne radioactivity and limits shall be established for the permissible concentration of radioisotopes in the experiments, according to the 10 CFR 20 limitations for exposure of individuals in restricted and unrestricted areas.

3.5.4 Bases

The general specifications assure that an adequate review process is followed to determine the safety, conditions and procedures for all experiments. The classification of experiments clearly delineates the responsibility for approving experiments according to their potential hazards; to assure that potentially hazardous experiments are analyzed for their safety implications; and that appropriate procedures are established for their execution. The reactivity limitations on experiments are established to prevent prompt criticality by limiting the worth of movable or non-secured experiments; to prevent a reactivity insertion larger than the stipulated maximum step reactivity insertion in the Accident Analysis, and to allow for reactivity control of experiments within the Reactor Control System capabilities (20 second positive period limitation). These specifications prohibit the irradiation of explosive materials and limit the amount of fissile materials that can be irradiated in the reactor according to its potential hazard and the reactor system's capability to handle a potential release to the cell environment. Explosive materials are defined as those materials normally used to produce explosive or detonating effects, materials which can chemically combine to produce explosion or detonations or any materials which can undergo explosive decomposition under influence of neutron, gamma or heat flux of the reactor or as defined by applicable standards.

3.6 Reactor Building Evacuation Alarm

These specifications apply to the equipment required for the evacuation of the Reactor Cell and the Reactor Building (including the Reactor Annex).

3.6.1 Specification

The Reactor Cell and the Reactor Building shall be evacuated when any of the following conditions exist:

A. The Evacuation Alarm is actuated automatically when two area radiation monitors alarm high ($\geq$ 25 mrem/hr) in coincidence.
B. The Evacuation Alarm is actuated manually when an Air Particulate Monitor is in a valid alarm condition.
C. The Evacuation Alarm is actuated manually when a reactor operator detects a potentially hazardous radiological condition and

preventive actions are required to protect the health and safety of operating personnel and the general public.

3.6.2 Bases

To provide early and orderly evacuation of the Reactor Cell and the Reactor Building and to minimize radioactive hazards to the operating personnel and reactor building occupants.

3.7 Fuel and Fuel Handling

3.7.1 Applicability

These specifications apply to the arrangement of fuel elements in core and in storage, as well as the handling of fuel elements.

3.7.2 Objective

To establish the maximum core loading for reactivity control purposes, to establish the fuel storage conditions, and to establish fuel performance and fuel handling specifications with regard to radiological safety considerations.

3.7.3 Specifications

A. The maximum fuel loading shall consist of 24 full fuel elements consisting of 11 plates each containing enriched uranium and cladded with high purity aluminum.
B. Fuel element loading and distribution in the core shall comply with the fuel handling procedures.
C. Fuel elements exhibiting release of fission products due to cladding rupture shall, upon positive identification, be removed from the core. Fission product contamination of the primary water shall be treated as evidence of fuel element failure.
D. The reactor shall not be operated if there is evidence of fuel element failure.
E. All fuel shall be moved and handled in accordance with approved procedures.
F. Fuel elements or fueled devices shall be stored and handled out of core in a geometry such that the k_{eff} is less than 0.8 under optimum conditions of moderation and reflection.
G. Irradiated fuel elements or fueled devices shall be stored so that temperatures do not exceed design values.

3.7.4 Bases

The fuel loading is based on the present fuel configuraton. The reactor systems do not have adequate engineering safeguards to continue operating with a detectable release of fission products into the primary coolant. The fuel is to be stored in a safe configuration and shall be handled according to approved written procedures for radiological safety purposes.

3.8 Primary Water Quality

3.8.1 Applicability

These specifications apply to primary cooling system water in contact with fuel elements.

3.8.2 Objective

To minimize corrosion of the aluminum cladding of fuel plates and activation of dissolved materials.

3.8.3 Specifications

A. Primary water temperature shall not exceed 155°F.
B. Primary 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 six (6) hours.
C. Primary water shall be sampled, evaporatively concentrated and the gross radioactivity of the residue measured with an adequate measuring channel. This specification procedure shall prevail:

(a) During the Weekly Checkout,
(b) Upon the appearance of any unusual radioactivity in the primary water or the primary water demineralizers and,
(c) Prior to the release of any primary water from the site.

D. Primary Equipment Pit water level sensor shall alarm in the Control Room whenever a detectable amount of water (1" above floor level) exists in the equipment pit.

3.8.4 Bases

Specification 3.8.3 A and 3.8.3 B are designed to protect the fuel element integrity and are based upon operating experience. At the specified quality, the activation products (of trace minerals) do not exceed acceptable limits. Specification 3.8.3 C is designed to detect and identify fission products resulting from fuel failure and to fulfill reportability requirements pertaining to liquid wastes. Specification 3.8.3 D is designed to alert the operator to potential loss of primary coolant, to prevent reactor operations with a reduced water inventory and to minimize the possibility of an uncontrolled release of primary coolant to the environs.

3.9 Radiological Environmental Monitoring Program

3.9.1 General

The UFTR Radiological Environmental Monitoring Program is conducted to assure that the radiological environmental impact of reactor operations is as low as reasonably achievable; it is conducted in addition to the radiation monitoring and effluents control specified under Section 3.8 of these Technical Specifications.

The Radiological Environmental Monitoring Program shall be conducted as specified below and under the supervision of the Radiation Control Office.

3.9.2 Radiological Environmental Monitoring

A. Monthly environmental radioactivity surveillance outside the restricted area shall be conducted by measuring the gamma doses at selected fixed locations surrounding the UFTR complex with acceptable personnel monitoring devices. A minimum of six (6) independent locations shall be used. A review of potential causes shall be conducted whenever a measured dose of over 40 mrem/month at two or more locations is determined and a report shall be submitted to the RSRS for review.

B. Radioactivity surveillance of the restricted area (Reactor Cell) shall be conducted as follows:

(a) Surface contamination in the restricted area shall be measured by taking random swipes in the Reactor Cell during the Weekly Checkout. Measured surface contamination greater than 100 dpm/cm^2 Beta-Gamma or greater than 50 dpm/100cm^2 Alpha are limiting conditions for operation requiring review and possible radiological safety control actions.

(b) Airborne particulate contamination shall be measured using a high volume air sampler. Measured radioactive airborne contamination 25% above mean normal levels are limiting conditions for operation requiring review and possible radiological safety control actions.

C. Radioactivity Surveys

The following radioactivity surveys, using portable radiation monitors are limiting conditions for operation:

(a) Surveys measuring the radiation doses in the restricted area shall be conducted quarterly, at intervals not to exceed four (4) months, and at any time a change in the normal radiation levels is noticed or expected. Radiation exposures shall be within 10 CFR 20 limits for radiation workers.

(b) Surveys measuring the radiation levels in the unrestricted areas surrounding the UFTR complex shall be conducted quarterly, at intervals not to exceed four (4) months, and at any time a change in the normal radiation levels is noticed or expected. Doses shall be within 10 CFR 20 limits for the general public.

3.9.3 Bases

The bases for establishing the Radialogical Environmental Surveillance Program are the established limits for internal and external radiation exposure and requirements that radiation doses be maintained "as-low-as-reasonably-achievable" (ALARA).

4.0 SURVEILLANCE REQUIREMENTS

Surveillance requirements relate to testing, calibration or inspection to assure that the necessary quality of systems and components is maintained; that facility operation will be within safety limits; and that the limiting conditions for operation will be met. Tests not performed within the specified frequency because of physical or administrative limitations shall be performed prior to resuming normal operations.

4.1 Surveillance Pertaining to Safety Limits and Limiting Safety Settings

4.1.1 Whenever an unscheduled shutdown occurs, an evaluation shall be conducted to determine whether a safety limit was exceeded.

4.1.2 Safety System operability tests shall be performed in accordance with Table II.

4.2 Surveillance Pertaining to Limiting Conditions for Operation

4.2.1 Reactivity Surveillance

A. The reactivity worth and reactivity insertion rate of each control blade, the shutdown reactivity and excess reactivity shall be measured annually (at intervals less than 14 months) or whenever physical or operational changes create a condition requiring re-evaluation of core physics parameters.
B. The temperature coefficient of reactivity shall be measured annually at intervals not to exceed 14 months.
C. The void coefficient of reactivity shall be checked biennially to assure that it is negative, at intervals not to exceed 30 months.

4.2.2 Reactor Control and Safety System Surveillance

A. The control blades drop time, from the fully withdrawn position, shall be measured semi-annually but at intervals not to exceed 8 months. If maintenance is performed on a blade, the drive mechanism or associated electronics, the rod drop time shall be measured before the system is considered operable.
B. The control blade full withdrawal and controlled insertion time shall be measured semi-annually, at intervals not to exceed 8 months.
C. Tests, limits, and frequencies of tests for the Control Blade Withdrawal Inhibit Interlocks operability tests shall be performed as listed in Table IV.
D. The mechanical integrity of the control blades and drive system shall be inspected during each in-core inspection but shall be fully checked at least once every 5 years.
E. Following maintenance or modification to the control blade system, an operability test and calibration of the affected portion of the system, including verification of control blade drive speed, shall be performed before the system is to be considered operable.

Table IV

CONTROL BLADE WITHDRAWAL INHIBIT INTERLOCKS OPERABILITY TESTS

Inhibit	Limit	Frequency
Reactor Period	$\leq$10 sec	Daily Checkout
Safety Channels and Wide Range Drawer not in OPERATE Position	-	Daily Checkout
Multiple Blade Withdrawal	Any 2 or more blades simultaneously in Manual	Daily Checkout
	Any 2 safeties in Automatic	Daily Checkout
Source Count Rate	<2 cps	Verification only when count rate <2 cps during Daily Checkout

F. The reactor shall not be started unless:

(a) The Weekly Checkout has been satisfactorily completed within 7 days prior to startup and,
(b) A Daily Checkout is satisfactorily completed within 8 hours prior to startup and,
(c) No known condition exists that would prevent successful completion of a Weekly or Daily Check.

G. The limitations established under Paragraph 4.2.2 F (a) and (b) can be deleted if a reactor startup is made within 6 hours of a normal reactor shutdown on any one calendar day.

H. The following channels shall be calibrated annually, at intervals not to exceed 13 months, and any time a significant change in channel performance is noted:

(a) log N - period channel
(b) power level safety channels (2)
(c) linear power level channel
(d) primary coolant flow measuring system
(e) primary coolant temperature measuring system

I. Following maintenance or modification to the reactor safety system, a channel test and calibration of the affected channel shall be performed before the reactor safety system is considered operable.

4.2.3 Reactor Vent System Surveillance

A. The Reactor Vent System flow rates shall be measured annually at intervals not to exceed 14 months, as follows:

(a) Reactor Cavity Exhaust Duct flow (1 cfm < flow rate < 400 cfm)
(b) Stack flow rate > 10000 cfm.

B. The following interlocks shall be tested quarterly at intervals not to exceed 8 months:

(a) Core Vent System damper closed if Diluting Fan is not operating,
(b) Reactor Vent System shut off when the Air Conditioning System is shut off due to actuation of the evacuation alarm.

4.2.4 Radiation Monitoring Systems and Radioactive Effluents Surveillance

A. The Area Radiation Monitor Channels, the Stack Monitor and the Air Particulate Monitor shall be verified to be operable prior to each reactor startup, as required by the Daily Checkout. Calibration of radiation monitoring channels shall be performed quarterly but at intervals not to exceed 4 months.

B. The Ar-41 concentration in the stack effluents shall be measured semi-annually at intervals not to exceed 8 months.

C. Releases of radioactive liquid waste from the holdup tanks shall be monitored before discharging to the sanitary sewage system to assure compliance with 10 CFR 20 regulations.

D. The reactor shall be placed in a Reactor Shutdown condition whenever Specification 4.2.4 A is not met.
E. The Reactor Vent System shall be immediately secured upon detection of failure of the Stack Monitoring System.

4.2.5 Surveillance of Experimental Limits

A. Surveillance to assure that experiments meet the requirements of Section 3.5 shall be conducted prior to inserting each experiment into the reactor.
B. The reactivity worth of an experiment shall be determined at approximately 1 watt power level or as appropriate within limiting conditions for operation, before continuing reactor operation with said experiment.

4.2.6 Reactor Building Evacuation Alarm Surveillance

A. The coincidence automatic actuation of the two area monitors and the manual actuation of the Evacuation Alarm shall be tested as part of the Weekly Checkout
B. The automatic shutoff of the Air Conditioning System and the Reactor Vent System shall be tested as part of the Weekly Checkout.
C. Evacuation drills for facility personnel shall be conducted quarterly, at intefvals not to exceed four (4) months, to assure that facility personnel are familiar with the emergency plan.

4.2.7 Surveillance Pertaining to Fuel

A. The in-core reactor fuel elements shall be inspected biennially at intervals not to exceed 30 months, in a randomly chosen pattern, as deemed necessary. At least two (2) elements will be inspected.
B. Fuel handling tools and procedures shall be reviewed for adequacy prior to fuel loading operations. The assignment of responsibilities and training of the fuel handling crew shall be performed according to written procedures.

4.2.8 Primary and Secondary Water Quality Surveillance

A. The primary water resistivity shall be determined as follows:

(a) Primary water resistivity shall be measured during the Weekly Checkout by a portable Solu Bridge using approved procedures. The measured value shall be larger than 0.4 megohm.
(b) Primary water resistivity shall be measured during the Daily Checkout at both the inlet and outlet of the demineralizers (DM). The measured value, determined by an on-line Solu Bridge alarming in the Control Room, shall be larger than 0.5 megohm-cm at the outlet of the DM.

B. The primary water radioactivity shall be measured during the Weekly Checkout for gross β-γ and gross α activity.

(a) The measured α activity shall not exceed 50 dpm above background level.

(b) The measured β-γ activity shall not exceed 25% above mean normal activity level.

(c) The secondary water system shall be tested for radioactive contamination during the Weekly Checkout according to written procedures.

5.0 DESIGN FEATURES

Design features are specified in order to assure that items important to safety are not changed without appropriate review. The items of concern are design features and parameters which were considered as limiting values (or significant for the protection of the reactor personnel and the general public) for the purpose of establishing Safety Limits, Limiting Safety System Settings, or Limiting Conditions for Operation.

5.1 Site

5.1.1 The UFTR is located on the University of Florida campus, at Gainesville Florida in the immediate vicinity of the buildings housing the College of Engineering and the College of Journalism. The Nuclear Sciences Center, which houses the Department of Nuclear Engineering, is annexed to the Reactor Building.

5.2 Reactor Cell

5.2.1 The reactor shall be housed in a reinforced concrete cell in the Reactor Building. The Reactor Building is a "vault-type" building as defined in 10 CFR 73.2(o). The Reactor Building is divided into two distinct parts based upon the difference in utilization and their structure. The overall Reactor Building measures approximately 60 ft. by 80 ft. inside. The Reactor Cell area is 30 ft. by 60 ft. with 29 ft. of head room, located at the north end of the building. The rest of the building is used for research laboratories, faculty offices and graduate study areas.

5.2.2 The Reactor Cell shall have an independent ventilation and air-conditioning system. The Reactor Vent effluents shall be discharged through the Reactor Stack, some 30 ft. above ground level.

5.2.3 All gases which may cause a hazard through neutron activation shall be exhausted from the Reactor Cell, Reactor Cavity, experiments or experimental facilities installed in or adjacent to the core or surrounding graphite and discharged to the environment via the Reactor Vent System and appropriately monitored for radioactivity, as specified under Chapter 3 of these Technical Specifications.

5.2.4 The 3-ton bridge crane shall not be used during reactor operation in a manner that could damage the control system and prevent it from performing its intended function. No load above 500 pounds shall be lifted over the control blade drive units unless the control blades are fully inserted. The crane shall be operated during reactor operations only by a licensed reactor operator.

5.2.5 Doors penetrating the Reactor Cell are the following:

A. An airlock passageway from the cell to the UFTR building lower hallway,
B. A door from the Control Room to the UFTR building lower hallway.
C. A freight door (10' x 12') leading to the environs. A panel in the freight door serves as an emergency personnel exit from the Reactor Cell. The freight door and panel shall be locked to

prevent entrance during reactor operation. The freight door or its personnel door shall not be used for general access to or egress from the Reactor Cell. This is not meant to preclude use of these doors in connection with authorized activities when the reactor is not in operation.

5.3 Reactor Fuel

5.3.1 Fuel elements shall be of the general MTR type, with thin fuel plates clad with aluminum and containing uranium fuel enriched to no more than about 93% U-235. The fuel matrix may be fabricated by alloying high purity aluminum-uranium alloy or by the powder metallurgy method where the starting ingredients (uranium-aluminum) are in fine powder form. The fuel matrix may also be fabricated from uranium oxide-aluminum (U_3O_8-Al) using the powder metallurgy process. There shall be nominally 14.5 grams of U-235 per fuel plate.

5.3.2 The UFTR Facility license authorizes the receiving,possession and use of:

A. Up to 4.82 kilograms of contained Uranium-235,
B. A 1 curie sealed plutonium-beryllium neutron source,
C. An up to 25 curie antimony-beryllium neutron source,
D. Other neutron and gamma sources may be utilized if their use does not constitute an unreviewed safety question pursuant to 10 CFR 50.59 and the sources meet the criteria established by the Technical Specifications.

5.4 Reactor Core

5.4.1 The core shall contain up to 24 fuel assemblies of 11 plates each. Up to six (6) of these assemblies may be replaced with pairs of partial assemblies. Each partial assembly shall be composed of either all dummy or all fueled plates. A full assembly shall be replaced with no fewer than ten plates in a pair of partial assemblies.

5.4.2 Fuel elements shall conform to these nominal specifications:

A. Overall size (bundle): 2.845 in. x 2.14 in. x 25.625 in.
B. Clad thickness: 0.015 in.
C. Plate thickness: 0.070 in.
D. Water channel width: 0.137 in.
E. No. of plates: standard full element - 11 fueled plates
partial element - 5 fueled plates
F. Plate attachment: bolted with spacers
G. Fuel content per plate: 14.5 g U-235 nominal

5.4.3 The reactor core shall be loaded so that all fuel assembly positions are occupied.

5.4.4 The fuel assemblies are contained in six (6) aluminum boxes arranged in two parallel rows of three boxes each, separated by about 30 centimeters of graphite. The fuel boxes are surrounded by a 5'x5'x5' reactor grade graphite assembly.

5.4.5 The top of the fuel boxes are covered during operations at power above

1 KW, by the use of the shield plugs and/or gasketed aluminum covers secured to the top of the fuel boxes. The devices function to prevent physical damage to the fuel, to minimize evaporation/leakage of water from the top of the fuel boxes and to minimize entrapment of Argon in the coolant water for radiological protection purposes.

5.5 Reactor Control and Safety Systems

Design feature of the components of the Reactor Control and Safety Systems that are important to safety, as specified under Section 3.2 of these Technical Specifications, are given below.

5.5.1 Reactor Control System

Reactivity control of the UFTR is provided by four (4) control blades, three (3) Safety and one (1) Regulating Blade. The control blades are of the swing-arm type consisting of four (4) aluminum vanes tipped with aluminum, protected by magnesium shrouds. They operate in a vertical arc within the spaces between the fuel boxes. Blade motion is limited to a removal time of at least 100 sec and the insertion time under trip conditions is stipulated to be less than 1 sec. The reactor blade withdrawal interlock system prevents blade motion which will exceed the reactivity addition rate of 0.06%Δk/k per second, as specified in these Technical Specifications. The control blade drive system consists of a two phase fractional horsepower motor that operates through a reduction gear train, and an electrically energized magnetic clutch that transmits a motor torque through the control blade shaft, allowing motion of the control blades. The blades are sustained in a raised position by means of this motor, acting through the electromagnetic clutch. Interruption of the magnet current results in a decoupling of the motor drive from the blade drive shaft, causing the blades to fall back into the core. Position indicators, mechanically geared to the rod drives transmit rod position information to the operator control console. Reactor shutdown can also be accomplished by voiding the moderator/coolant from the core. Two independent means of voiding the moderator/coolant from the core are provided:

A. Water dump via the Primary Coolant System Dump Valve opening under Full Trip conditions,
B. Water dump via the rupture disk breaking under pressure conditions above design value.

The integral worths of the individual safety blades vary from about 1.3 to 2.3%Δk/k depending on position in the core and individual characteristics. The Regulating Blade worth is about 1%Δk/k. The rod worths, drive speeds and drop-time values are sufficiently conservative to ensure compliance with the specified reactivity limitations. Additional reactivity and power related features are obtained from the Control Blade Withdrawal Inhibits. The Regulating Blade may be engaged by a servo-mechanism controlled by the Linear Channel for automatic reactor power control.

5.5.2 Reactor Safety System

A. Power Level Channels

Two independent measuring channels are provided for power level limits; both are required for the reactor to be operable. Each channel covers reliably the range from about 1% to 150% of full power (of 100 KW). One channel (Safety 1) is part of the Wide Range Drawer, and receives its main signal from a fission chamber. The Safety 2 channel uses an uncompensated ion chamber for neutron detection. Each channel drops all control rods and the moderator coolant from the core by actuating bi-stable trips in the Safety System in a one-out-of-one trip logic. Visual indication of the power measured by each chamber, as well as annunciator of channel status is available to the operator in the Control Room.

B. Wide Range Logarithmic Power Level and Period Channel

The Logarithmic Power Channel covers the wide range from reactor startup to full power in 10 decades. It uses a fission chamber for this entire range and uses a B^{10} proportional counter only in the startup (source) range. Signals from the fission chamber and the B^{10} counter are amplified by a preamplifier before going to the Log Channel. The preamplifier also processes test signals from the console controls and de-energizes the B^{10} proportional counter at about 400 cps. Power level information is displayed on a meter and on a 2 pen recorder. The channel provides the following blade withdrawal inhibits or blade trips: minimum source count inhibit of 2 cps, fast period inhibit of 10 seconds, fast period trip of 3 seconds, and inhibit limiting power escalation in the automatic mode to no faster than 30 seconds, and a trip at or above 1% power when secondary coolant flow is below the trip setting. Because this is a wide range channel, a separate startup channel is not used. These control or limiting actions prevent startup or operation of the reactor unless it is properly monitored or if operational restrictions are not met. Period is displayed on a meter and is effective for control over the entire range of operation.

C. Startup (Neutron) Source(s)

A permanent, regenerable, Antimony-Beryllium source of up to 25 curies and/or a removable Plutonium-Beryllium source of 1 curie may be used for reactor startup to monitor the approach to criticality. The use of a neutron source insures that behavior of the reactor is being monitored by the reactor instrumentation during subcritical control blade manipulations.

D. Linear Neutron Channel and Automatic Flux Control System

The Linear Channel is required to be operable when the reactor is to be operated in the Automatic Mode. The Linear Channel uses a compensated ion chamber for neutron detection; its signal is transmitted by a multirange pico-ammeter. The pico-ammeter sends a signal to one channel of the 2 pen recorder to display power level from source level to full power. It also sends a signal to the automatic flux controller which, in comparison with a signal from a % power setting control acts to establish and/or hold power level at a desired value. The rate of power increase is controlled by the action of a limiter in the Linear Channel/Automatic Control System which maintains the reactor period at or slower than 30 seco

5.6 Cooling Systems

5.6.1 Primary Cooling System

The primary coolant is demineralized light water, which is normally circulated in a closed loop. The flow is from the 200 gallon storage (dump) tank to the primary coolant pump; water is then pumped through the primary side of the heat exchanger and to the bottom of the fuel boxes, upward past the fuel plates to overflow pipes located about six (6) inches above the fuel, and into a header for return to the storage tank. A purification loop is used to maintain primary water quality. The purification loop pump circulates about 1 gpm of primary water, drawn from the discharge side of the heat exchanger, through mixed bed ion-exchange resins and a ceramic filter. The purification loop pump automatically shuts off when the primary coolant pump is operating since flow through the purification system is maintained. Primary Coolant may be dumped from the reactor fuel boxes by opening an electrically operated solenoid dump valve, which routes the water to the dump tank. A pressure surge of about 2 pounds above normal in the system will also result in a water dump by breaking a graphite rupture disk in the dump line. This drains the water to the primary equipment pit floor actuating an alarm in the Control Room. The Primary Coolant System is instrumented as follows:

A. Thermocouples at each fuel box and the main inlet and outlet header (8 total), alarming and recording in the Control Room,
B. A flow sensing device in main inlet line, alarming and displayed in the Control Room,
C. A flow sensing device (no flow condition) in the outlet line, alarming in the Control Room,
D. Resistivity probes monitoring the inlet and outlet reactor coolant flow, alarming and displayed in the Control Room,
E. An equipment pit water level monitor, alarming in the Control Room.

The reactor power is calibrated annually by the use of the coolant flow and temperature measuring channels.

5.6.2 Secondary Cooling System

Two secondary cooling systems are normally operable in the UFTR: a Well Secondary Cooling System and a City Water Secondary Cooling System. The Well Secondary Cooling System is the main system used for removal of reactor generated heat to the environment. A deep well furnishes about 200 gallons per minute of cooling water to the shell side of the heat exchanger, removing primary heat and rejecting it to the storm sewer. Weekly samples monitor the activity of this water. Flow indications in the Control Room are 140 gpm as a warning and 60 gpm to initiate a trip at or above 1 KW after a 10 second warning. The City Water Secondary Cooling System can be used for back-up cooling or for specific operations requiring reactor coolant temperatures hotter than those obtained with the Well Cooling System. The secondary flow by the City Water System is about 40 gpm, with a reactor trip set at 8 gpm (as measured by a flow switch) for power levels at or above 1 KW. A Back Flow Preventer in the city water line insures compliance with the requirements of the National Plumbing Code to prevent contamination of a Potable Water Supply. The secondary coolant system inlet and outlet temperatures are monitored by thermocouples, with alarm and record functions in the Control Room.

5.7 Radiological Safety Design Features

5.7.1 Physical Features

The containment structure consists of the Reactor Cell, with a free air volume of about 1600 m^3. This building houses the reactor, reactor control room, the primary cooling system (including the dump tank heat exchanger and purification loop), secondary coolant piping and Reactor Vent System. Access to the Reactor Cell, which is the designated restricted and security area, is controlled by the specifications established by the Physical Security Plan of the UFTR (WITHHELD FROM PUBLIC DISCLOSURE PURSUANT TO 10 CFR 2.790(d)). Ventilation is through the independent Air Conditioning Ventilation and Reactor Vent System. The Reactor Vent System can be secured to prevent uncontrolled discharg of radioactivity to the environment or releases in excess of permissibl levels (per 10 CFR 20). Rough and absolute filters are used to eliminate or minimize radioactive air particulate contamination from the exhaust air. The electrically actuated damper in the core exhaust line is fail-safe and closes upon de-energization.

5.7.2 Monitoring System

Area and Stack Monitors are used for radioactivity monitoring, as delineated in Sections 3.3, 3.4 and 3.6 of these Technical Specifications The cell air is monitored by an Air Particulate Detector. Exhaust air drawing from the reactor cavity, reactor cell or experiments is continously monitored for gross concentrations of radioactive gases.

5.7.3 Evacuation Sequence

The emergency evacuation sequence is initiated either automatically by two area monitors alarming high in coincidence or manually by the console operator. The sequence is that the reactor room AC Ventilation System and the Reactor Vent System are shut down and the core vent damper is closed.

5.8 Fuel Storage

5.8.1 New Fuel

Unirradiated new fuel elements are stored in a vault-type room security area equipped with intrusion alarms in accordance with the Security Plan. Elements are stored in a steel, fireproof safe in which a cadmium plate separates each layer of bundles to assure subcriticality under optimum conditions of moderation and reflection.

5.8.2 Irradiated Fuel

Irradiated fuel is stored upright in dry storage pits within the reactor building in criticality-safe holes.

6.0 ADMINISTRATIVE CONTROLS

6.1 Definitions

6.1.1 Certified Operators--An individual authorized by the Nuclear Regulatory Commission to carry out the duties and responsibilities associated with a position requiring the certification.

6.1.2 Class A Reactor Operator--Any individual who is certified to direct the activities of Class B reactor operators; such an individual is also a reactor operator. Such an individual is commonly referred to as a Senior Reactor Operator.

6.1.3 Class B Reactor Operator--Any individual who is certified to manipulate the controls of the reactor. Such an individual is commonly referred to as Reactor Operator.

6.2 Organization

6.2.1 Structure

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

A. Level 1 - Individuals responsible for the reactor facility's licenses, charter and site administration.
B. Level 2 - Individual responsible for reactor facility management.
C. Level 3 - Individual responsible for reactor operations and supervision of day-to-day facility activities.
D. Level 4 - Reactor Operating Staff (Class A and B Reactor Operators and trainees).
E. The Reactor Safety Review Subcommittee is appointed by, and shall report to,the Chairman of the Radiation Control Committee. The Chairman of the Radiation Control Committee reports to the Director of Environmental Health and Safety, who reports to the Vice-President for Administrative Affairs. Radiation safety personnel shall report to Level 2 or higher.

6.2.2 Responsibility

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

6.2.3 Staffing

A. The minimum staffing when the reactor is not secured shall be:

(a) A certified reactor operator in the Control Room.

(b) A second person present at the facility complex able to carry out prescribed written instructions including instructions to initiate the first stages of the emergency plan, including evacuation and initial notification procedures. Unexpected absence for two hours is acceptable provided immediate action is taken to obtain a replacement.

(c) A designated Class A Reactor Operator shall be readily available on call. "Readily Available on Call" means an individual who (1) has been specifically designated and the designation known to the operator on duty, (2) keeps the operator on duty informed of where he may be rapidly contacted and the phone number or other means of communication available, and (3) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g. 30 minutes or within a 15 mile radius)

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

(a) Management personnel,
(b) Radiation safety personnel,
(c) Other operations personnel.

C. Events requiring the direction of Class A Reactor Operator:

(a) All fuel or control-rod relocations within the reactor core region,
(b) Relocation of any in-core experiment with a reactivity worth greater than one dollar,
(c) Recovery from unplanned or unscheduled shutdown (in this insta documented verbal concurrence from Class A Operator is require

6.2.4 Selection and Training of Personnel

The selection, training and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1977, Sections 4-6.

6.2.5 Review and Audit

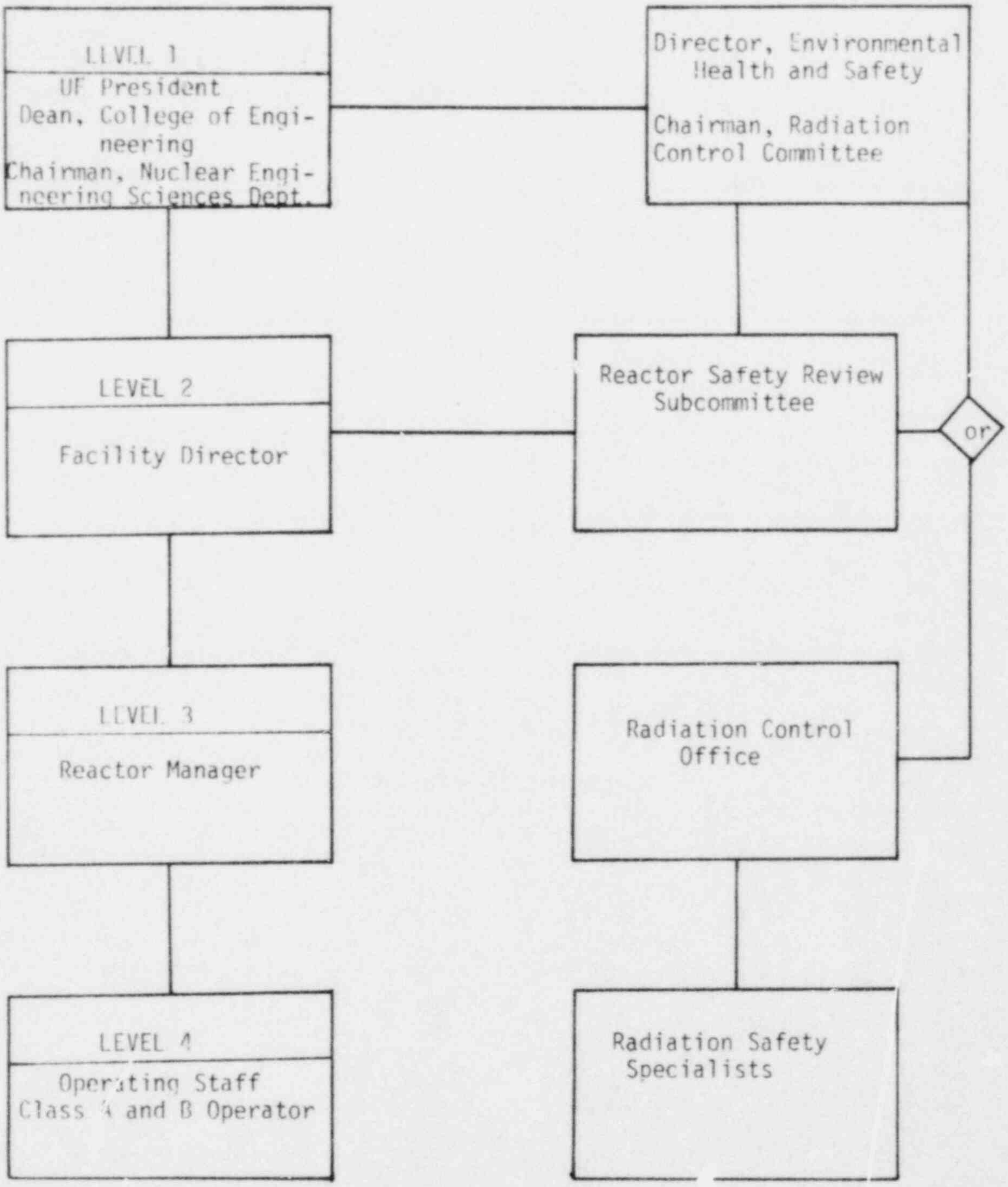
A method for the independent review and audit of the safety aspects of reactor facility operations shall be established to advise management. The review and audit functions of the UFTR operations are conducted by the Reactor Safety Review Subcommittee (RSRS).

A. Composition and Qualifications

The RSRS shall be composed of a minimum of five members, including the Reactor Manager and Radiation Control Officer (both ex-officio voting members), the Chairman of the Nuclear Engineering Sciences Department and two other members having expertise in reactor technology and/or radiological safety.

Figure 1

UFTR ORGANIZATION CHART



B. Charter and Rules

The review and audit functions shall be conducted in accordance with the following established charter:

(a) Designation - The name of the Subcommittee is: Reactor Safety Review Subcommittee. The Subcommittee may be referred to in abbreviated form as the RSRS.
(b) Accountability - The RSRS is a Subcommittee of and reports to the University Radiation Control Committee, which may be referred to in abbreviated form as the URCC. The URCC provides radiological safety recommendations to the Director of Environmental Health and Safety.
(c) Scope - the RSRS shall be responsible for the review of safety related issues pertianing to the University of Florida Training Reactor, which may be referred to in abbreviated form as the UFTR.
(d) Purpose - The purpose of the RSRS is to assure the safe operation of the UFTR through the discharge of the Subcommittee review and audit functions
(e) Membership -

(1) The RSRS shall consist of at least five members. Membership will include the Chairman of the Nuclear Engineering Sciences Department, University Radiation Control Officer, Reactor Manager and two technical personnel familiar with the operation of reactors and with the design of the UFTR and radiological safety, at least one of whom is from outside the Department of Nuclear Engineering Sciences. The two technical personnel will be recommended to the Chairman of the URCC by the Chairman of the Department of Nuclear Engineering Sciences. Any member may designate a duly qualified representative to act in his absence from standing URCC approved list.
(2) An Executive RSRS Committee will consist of the Reactor Manager, University Radiation Control Officer and Chairman of the RSRS.
(3) The Chairman of the RSRS will be appointed by the Chairman of the URCC. The Chairman of the RSRS is an ex-officio voting member of the URCC and will serve as liason between the RSRS and the URCC.
(4) Members appointed to the Subcommittee shall be reviewed, and as appropriate, new appointments made by October 1 of each calendar year.

(f) Meetings -

(1) Meeting frequency shall be quarterly at intervals not to exceed 4 months. Meetings may be held more frequently as circumstances warrant, consistent with the effective monitoring of facility operations as determined by the RSRS Chairman.
(2) Review of draft minutes will be completed prior to subsequent meetings, at which time they will be submitted for approval. Responsibility to assure that this is done fa

upon the RSRS Chairman. The RSRS Chairman is charged with the responsibility to assure that the minutes are submitted for approval in a timely manner.

(3) A quorum shall consist of at least three members and at least three members must agree when voting, regardless of the number present.

C. Review Function

The following items shall be reviewed:

(a) Determination that proposed changes in equipment, systems, test, experiments, or procedures do not involve an unreviewed safety question,
(b) All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance,
(c) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity,
(d) Proposed changes in technical specifications, license or charter.
(e) Violations of technical specifications, license or charter, Violations of internal procedures or instructions having safety significance,
(f) Operating abnormalities having safety significance,
(g) Reportable occurrences,
(h) Audit reports and annual facility reports.

A written report or minutes of the findings and recommendations of the review group shall be submitted to RSRS members in a timely manner after the review has been completed and to the Chairman of the Radiation Control Committee whenever a finding is deemed to require review by Level 1.

D. Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs and other documents. Where necessary, discussions with cognizant personnel shall take place. In no case shall the individual immediately reponsible for the area, audit in that area. The following items shall be audited:

(a) Facility operations for conformance to the technical specifications and applicable license or charter conditions, at least once per calendar year (interval between audits not to exceed 15 months).
(b) The retraining and requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months).
(c) The results of action taken to correct those deficiencites that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months)
(d) The reactor facility emergency plan, and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Radiation Control Committee and the Dean of the College of Engineering. A written report of the findings of the audit shall be submitted to the Dean of the College and the review audit group members within 3 months after the audit has been completed.

6.3 Procedures

The facility shall be operated and maintained in accordance with approved writ procedures. All procedures and major revisions thereto shall be reviewed and approved by the Director of Nuclear Facilities prior to being effective.

6.3.1 The following types of written procedures shall be maintained:

A. Normal startup, operation and shutdown procedures for the reactor. These procedures shall include applicable checkoff lists and instru tions,
B. Fuel loading, unloading and movement within the reactor,
C. Procedures for handling irradiated and unirradiated fuel elements,
D. Routine maintenance of major components of systems that could have an effect on reactor safety,
E. Surveillance tests and calibrations required by the technical specifications or those that may have effect on reactor safety,
F. Personnel radiation protection, consistent with applicable regulations,
G. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity,
H. Implementation of the Emergency Plan,
I. Procedures which delineate the operator action required in the event of specific malfunctions and emergencies,
J. Procedures for flooding conditions in the reactor facility, including guidance as to when the procedure is to be initiated and guidance on reactivity control.

6.3.2 Substantive changes to the above procedures shall be made effective only after documented review by the RSRS and approval by the Facility Director (Level 2) or designated alternatives. Minor modifications to the original procedures which do not change their original intent may be made by the Reactor Manager (Level 3) or higher but modifications must be approved by Level 2 or designated alternates within 14 days. Temporary deviations from the procedrues may be made by the senior operating individual present, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documen and reported to Level 2 or designated alternates.

6.4 Experiments Review and Approval

6.4.1 Experiments review and approval shall be conducted as specified under Section 3.5 - Limitations on Experiments - of these Technical Specifications.

6.4.2 The experiments review and approve shall assure compliance with the requirements of the license, technical specifications, and applicable regulations and shall be documented.

6.4.3 Substantive changes to previously approved experiments with safety significance shall be made only after review by the RSRS,approval in writing by Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

6.4.4 Approved experiments shall be carried out in accordance with established approved procedures.

6.5 Required Actions

6.5.1 Action to be Taken in Case of Safety Limit Violation

A. The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.
B. The safety limit violation shall be promptly reported to Level 2 or designated alternates.
C. The safety limit violation shall be reported to the Nuclear Regulatory Commission.
D. A safety limit violation report shall be prepared. The report shall describe the following:

(a) Applicable circumstances leading to the violation including when known the cause and contributing factors,
(b) Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public,
(c) Corrective action to be taken to prevent recurrence.

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

6.5.2 Action to be Taken in the Event of an Occurrence of the Type Identified in Section 6.6.2

A. Reactor conditions shall be returned to normal or the reactor shall be shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operations shall not be resumed unless authorized by Level 2 or designated alternates.
B. Occurrence shall be reported to Level 2 or designated alternates and to the Commission as required.
C. Occurrence shall be reviewed by the review group at their next scheduled meeting.

6.6 Reports

In addition to the requirements of applicable regulations, reports shall be made to the Commission as follows:

6.6.1 Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted to the Commission within three months following the end of each prescribed year. The prescribed year ends August 31 for the UFTR. Each annual operating report shall include the following information:

A. A narrative summary of reactor operating experience including the energy produced by the reactor and the hours the reactor was critical,
B. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence,
C. Tabulation of major preventive and corrective maintenance operation having safety significance.
D. Tabulation of major chang in the reactor facility and procedures, and tabulation of new tests or experiments, that are significantly different from those performed previously and are not described in the Safety Analysis Report, including conclusions that no unreviewed safety questions were involved,
E. A summary of the nature and amount of radioactive effluents release or discharged to the environs beyond the effective control of the facility operators as determined at or before the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient,
F. A summarized result of environmental surveys performed outside the facility,
G. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed.

The annual report shall be submitted to the Director, Division of Licensing, U.S. NRC, Washington, DC 20555 and to the Director, NRC Region II, Inspection and Enforcement Office, Atlanta, Georgia.

6.6.2 <u>Special Reports</u>

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

(a) Release of radioactivity from the site above allowed limits,
(b) Violation of safety limits (see 6.5.1),
(c) Any of the following:

(1) Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the technical specifications,
(2) Operation in violation of limiting conditions for operati established in the technical specifications unless prompt remedial action is taken,
(3) A reactor safety system component malfunction which rend the reactor safety system incapable of performing its intended safety function unless the malfunction or condit is discovered during maintenance tests or periods of reac shutdowns. (Note: Where components or systems are provided in addition to those required by the technical specifications, ... ure of the extra components or system

is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.),

(4) An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded.

(5) Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment or both,

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

(7) A violation of the Technical Specifications or the facility license.

6.7 Records

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single, or multiple records, or a combination thereof. Recorder charts showing operating parameters of the reactor (i.e., power level, temperature, etc.) for unscheduled shutdown and significant unplanned transients shall be maintained for a minimum period of two years.

6.7.1 Records To Be Retained for a Period of at Least Five Years

A. Normal reactor facility operation (supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least one year).
B. Principal maintenance operations.
C. Reportable occurrences.
D. Surveillance activities required by the Technical Specifications.
E. Reactor facility radiation and contamination surveys where required by applicable regulations.
F. Experiments performed with the reactor.
G. Fuel inventories, receipts, and shipments.
H. Approved changes in operating procedures.
I. Records of meeting and audit reports of the RSRS.

6.7.2 Records To Be Retained for at Least One Training Cycle

A. Retraining and requalification of certified operations personnel. Records of the most recent complete cycle shall be maintained at all times the individual is employed.

6.7.3 Records To Be Retained for the Lifetime of the Reactor Facility

(Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

A. Gaseous and liquid radioactive effluents released to the environs.
B. Off-site environmental monitoring surveys required by the Technical Specifications.

C. Radiation exposure for all personnel monitored.
D. Updated drawings of the reactor facility.

7.0 ALARA (10 CFR 50.36a)

The principal routine emission from the UFTR facility complex is Argon-41 discharged by the Reactor Vent System. There is no known biological uptake of Argon-41 and exposure limits are based upon external, total body irradiation.

The concentration of Argon-41 in the stack effluent is continuously monitored when the reactor is operating, and is normally less than 1 x 10^{-5} μCi/ml after several hours of full power operation. The annual release is related to the number of equivalent hours of 100 KW operation (KWh per year). Reactor operations are limited by prior agreement, and by these Technical Specifications, to the limit the Argon-41 discharges to MPC when averaged over a month and using the established atmospheric dilution factor of 200.

The off-site environmental radioactive surveillance program has proven that exposure to the general public from the reactor radioactive effluents approaches consistently the non-detectable level and certainly are always well below the 500 mrem/year federal limit.

The ALARA program at the UFTR minimizes unnecessary production of radioactive effluents by selectivity of operations. The potential reduction of Argon-41 releases is frequently reviewed, and is a major item of consideration during the upcoming reviews to upgrade facility operations to 500 KWth. A reduction of the vent flow as well as of the Argon dissolving in the primary coolant are being proposed as well as the possibility of utilizing storage tanks.

Radioactive liquid effluents and personnel radioactive exposure are well within ALARA guidelines.

17. QUALITY ASSURANCE

Paragraph (a) (7) of Part 50.34 "Contents of Applications; Technical Information" of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each applicant for a construction permit to build a utilization facility include in its Preliminary Safety Analysis Report a description of the Quality Assurance Program to be applied to the design and construction of the structures, systems, and components of the facility. Since the UFTR is an already operating facility presented for license renewal, these preliminary requirements are not applicable here.

At the time of construction and installation of the UFTR facility, and at other points whenever significant physical modifications were made to the UFTR reactor facility as documented in license amendments and other records of system changes, the necessary assurances of quality in the design, procurement, construction, installation and operation of the facility were obtained and records kept and stored by those responsible for assuring UFTR safety. At any future time that significant physical modifications are considered for the UFTR, established quality assurances are required. Since the UFTR is a small installation, there is no separate QA program division. However, the various requirements for QA programs are and will continue to be met by an active and effective system of overviews. In addition, adequate records to assure UFTR quality have been kept.

Regulatory Guide 2.5, Revision 0-12, May 1977 describes a method considered acceptable by the NRC staff for complying with the Commission's regulations with regard to overall quality assurance program requirements for research reactors. In effect, this Regulatory Guide references and supports the standard ANSI N402-1976, "Quality Assurance Program Requirements for Research Reactors" which describes a quality assurance program for use in research reactor facilities. Since the general requirements for establishing and executing a quality assurance program for the design, construction, testing, modification, and maintenance of research reactors that are included in ANSI N402-1976 provide a method acceptable to the NRC for complying with the Program requirements of 10 CFR 50.34, this standard will be used as a guide for all future design, construction, and testing connected with significant modifications to the UFTR. Testing and maintenance of the existing UFTR facilities continue to have their quality assured as in the past; maintenance and testing quality assurance records are kept in accordance with previously established procedures which have been found acceptable by the NRC.

Paragraph (b)(6)(ii) of Part 50.34 requires that each applicant for a license to operate a facility include in the Final Safety Analysis Report a description of the managerial and administrative controls to be used to assure safe operation. The required description of the managerial and administrative controls used to assure safe UFTR operations are contained in Chapter 13, "Conduct of Operations," and in certain sections of Chapter 12, "Radiation Protection" of this Safety Analysis Report. This managerial and administrative organization is considered adequate to continue to assure the safety of operations at the UFTR facility from the design stages of any proposed modifications through associated testing and maintenance as well as continued operation of the UFTR in its present capacity or in any altered capacity or arrangement are approved by the NRC.

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