THE UNIVERSITY OF TEXAS AT AUSTIN

DECOMMISSIONING PLAN

FOR THE UNIVERSITY OF TEXAS
TRIGA REACTOR
COLLEGE OF ENGINEERING
TAYLOR HALL
ROOM 131

LICENSE NO. R-92
DOCKET NO. 50-192

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CHAPTER 1

1.0 PLAN BACKGROUND AND MANAGEMENT

The University of Texas (UT) plans to cease the operation of the TRIGA Mark I research reactor located at Taylor Hall, Room 131, on the main compus at Austin and return the area to unrestricted use. The plan contained herewith presents the decommissioning alternative tasks; occupational and radiation protection programs during dismantling operation; radioactive materials and waste management; pertinent technical and environmental specifications, and a proposed termination radiation survey plan to provide the bases for verifying that the facility and site meet prescribed radioactivity levels for their unrestricted use.

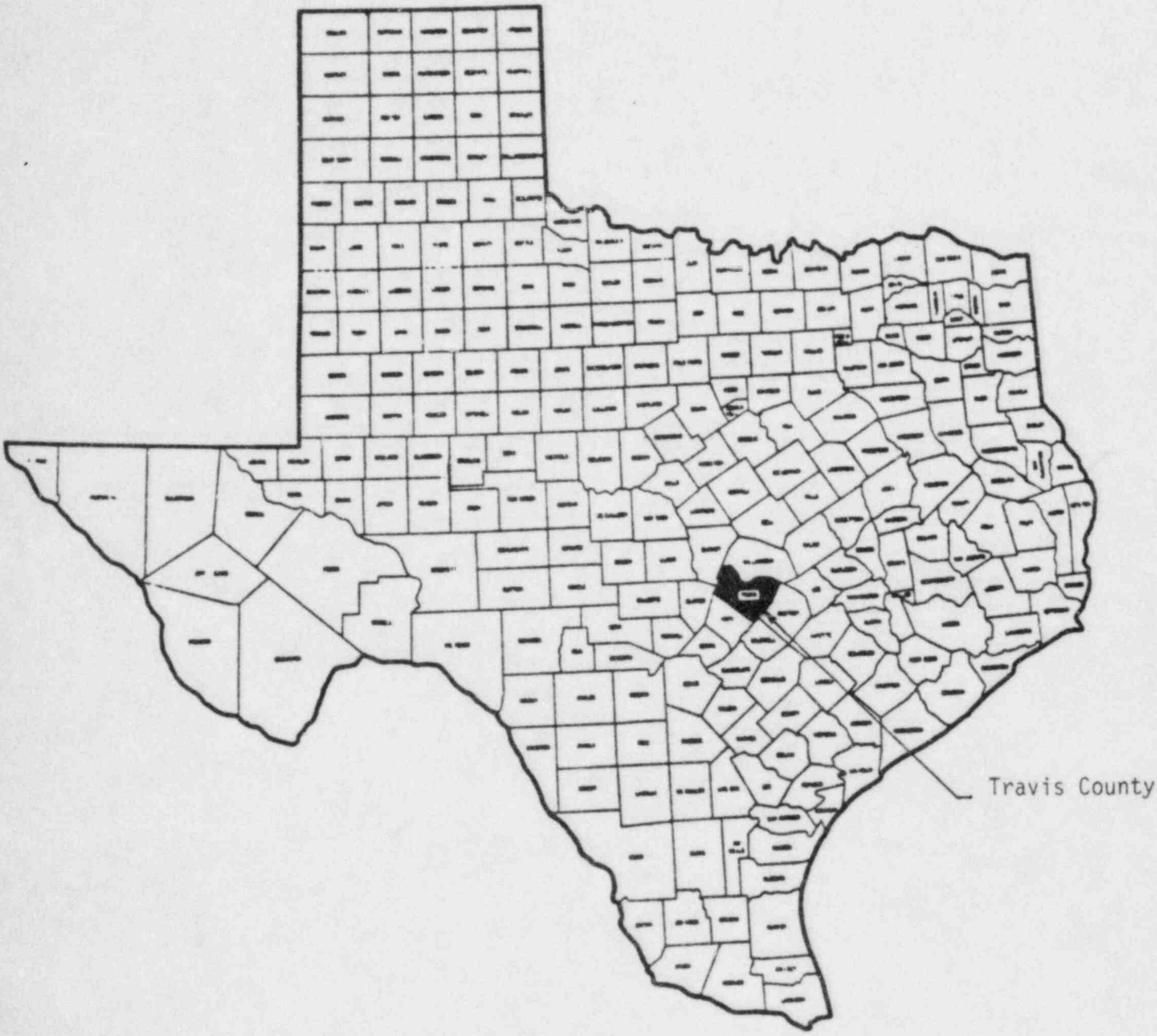
1.1 Summary Description

An overview of the most significant features of the UT reactor and of the proposed decommissioning plan is presented below.

1.1.1 Reactor Facility Description

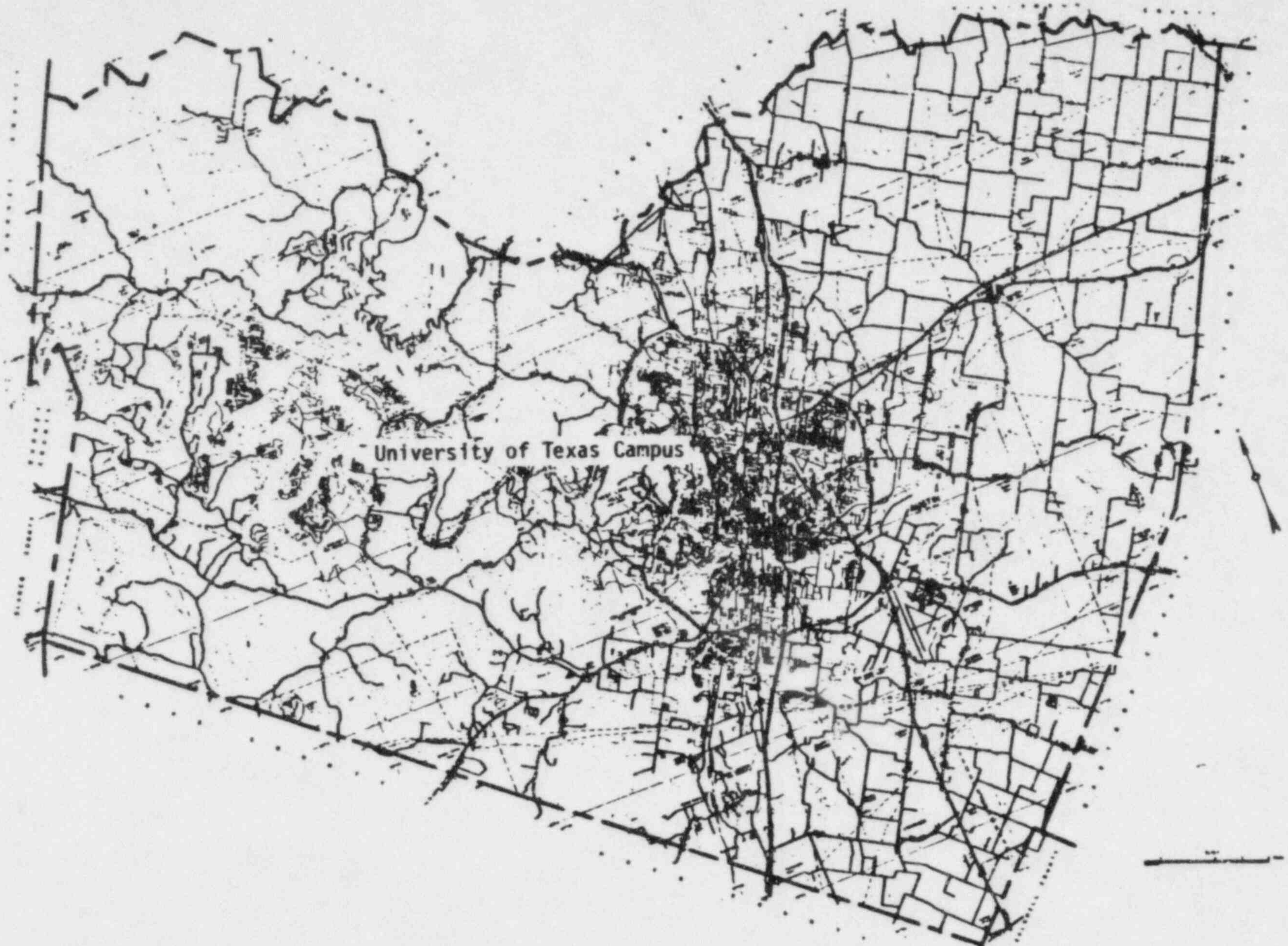
The UT reactor facility is situated on the main campus of the University of Texas, City of Austin, Travis County, Texas (see Figures 1.1, 1.2, 1.3). The reactor is located near the center of campus in room 131 of Taylor Hall (see Figures 1.4 and 1.5).

The central campus is located on the high point of a gentle knoll. The reactor site is located on the east slope and about 50 feet below the crest of the knoll and about 40 feet above Waller Creek, which is 2 blocks away. The reactor site is 1.7 mi north and 3.0 mi east of the Colorado River; approximately 1.5 mi east, north, and west of branches of the Missouri Pacific, Missouri Kansas Texas, and Texas New Orleans (Southern Pacific) railroads; 0.6 mi west of Interstate Highway 35; 0.8 mi north of the Texas State Capitol building; 2.0 mi southwest of the Austin Municipal Airport; and 8.0 mi northwest of Bergstrom Air Force Base.



TEXAS

COUNTY OUTLINE MAP



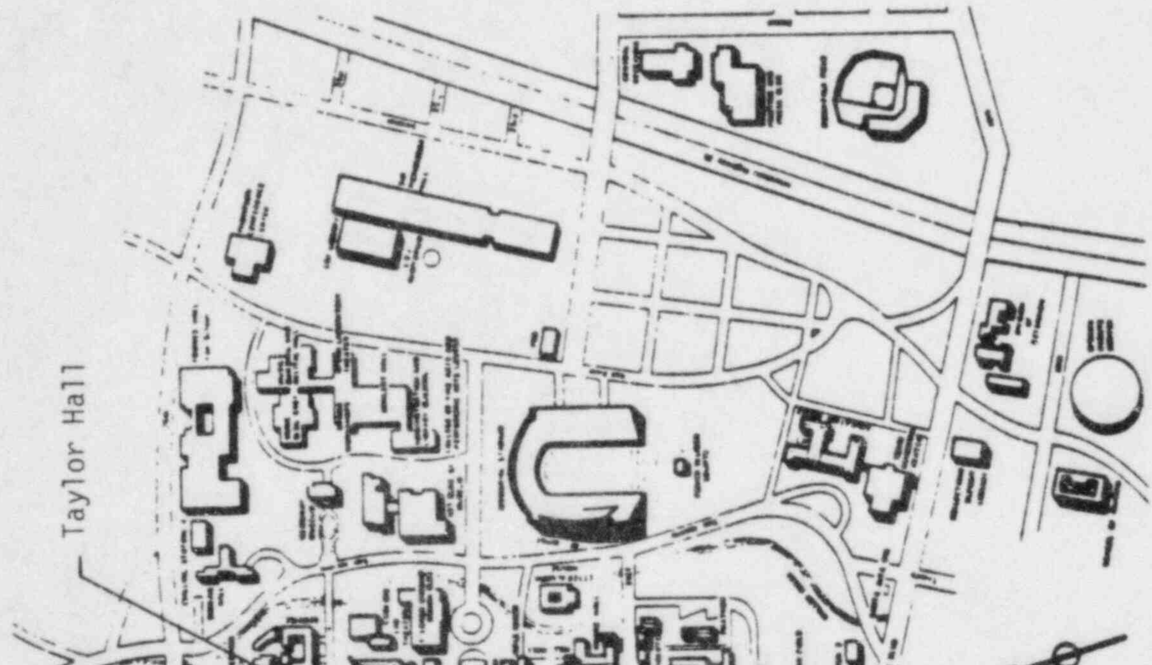
TRAVIS COUNTY, TEXAS

FIGURE 1.2

FIGURE 1.3

DOWNTOWN AUSTIN

FIGURE 1.4



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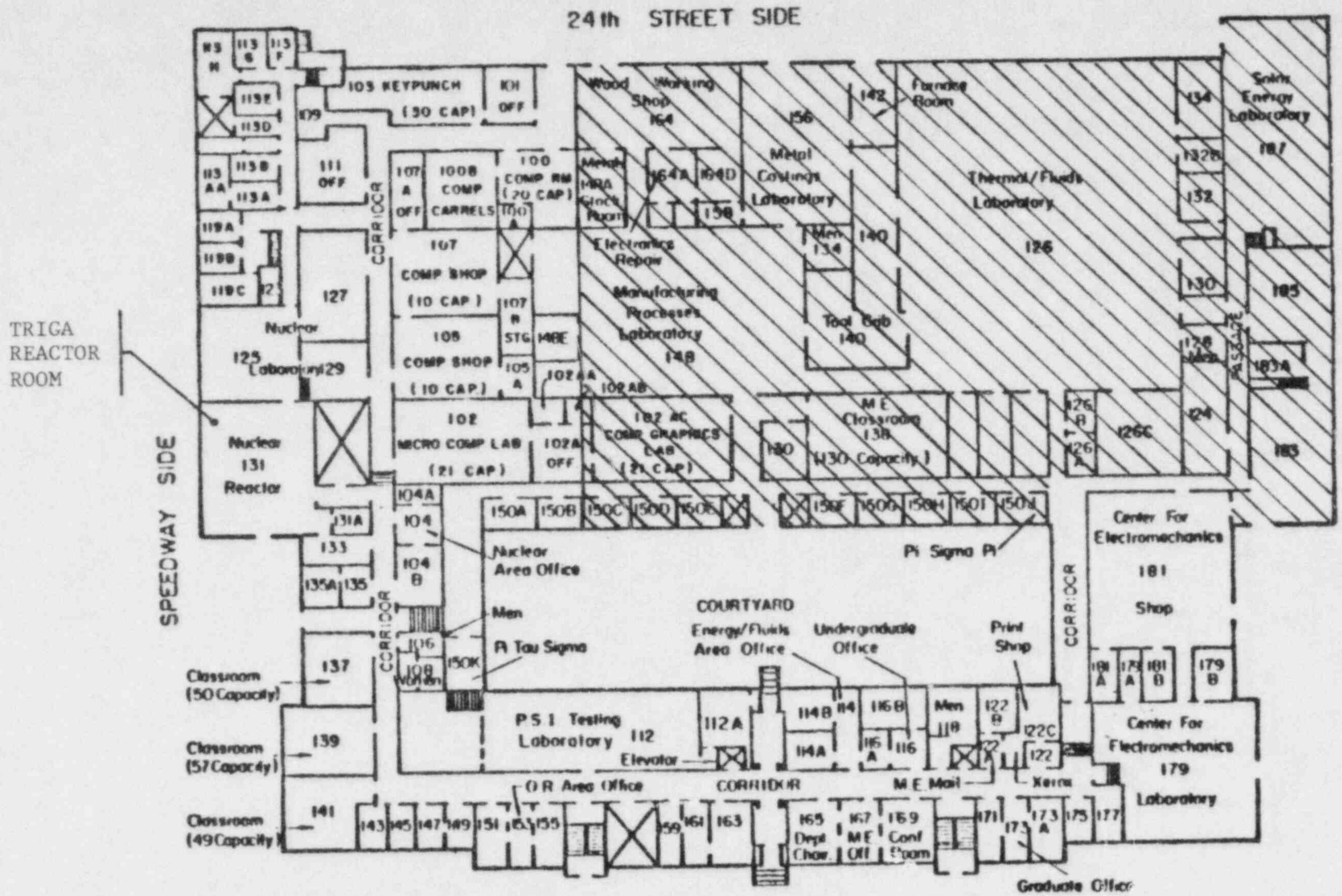


Building demolished (Dec.84)

TAYLOR HALL - FIRST FLOOR

FIGURE 1.5

The UT TRIGA Mark I is located in a modified room of an engineering building that includes laboratory space, offices, and classrooms. Room 131 of Taylor Hall contains the reactor with adjacent rooms to the north (rooms 125, 127, and 129) providing additional laboratory space and rooms to the south (rooms 133 and 135) providing office space. Normal laboratory access is from a building corridor east of the laboratory. The laboratory is accessible from the office areas to the south and laboratory areas to north. An outside double-door exit provides for equipment movement and emergencies. The outside door will be used only with the reactor shut down and under direct supervision of authorized personnel. The room is of brick and reinforced concrete construction.

The University of Texas reactor is a standard Mark I TRIGA heterogeneous pool-type reactor incorporating solid uranium-zirconium hydride fuel-moderator elements with an enrichment of less than 20% ^{235}U. The reactor core is submerged in a large, open tank of light water that acts as both a moderator and coolant. Reactor control is achieved by insertion and withdrawal of neutron-absorbing control rods. Pulses are initiated by the pneumatic ejection of a transient rod. Principal design parameters of the UT TRIGA reactor are indicated in Table 1.1 and an overall view of a typical Mark I TRIGA reactor is shown in Figure 1.6.

The reactor core consists of a lattice of approximately 72 cylindrical $U\text{-}ZrH_x$ fuel elements, 15 graphite dummy elements, and 3 control rods. The elements are held in concentric cylindrical rings by upper and lower aluminum grid plates. The active (or fueled) region of the reactor core forms a right circular cylinder $\sim$ 16.5 inches in diameter and 15 inches high and contains $\sim$ 2.8 kg of ^{235}U. Water coolant occupies approximately one third of the core volume.

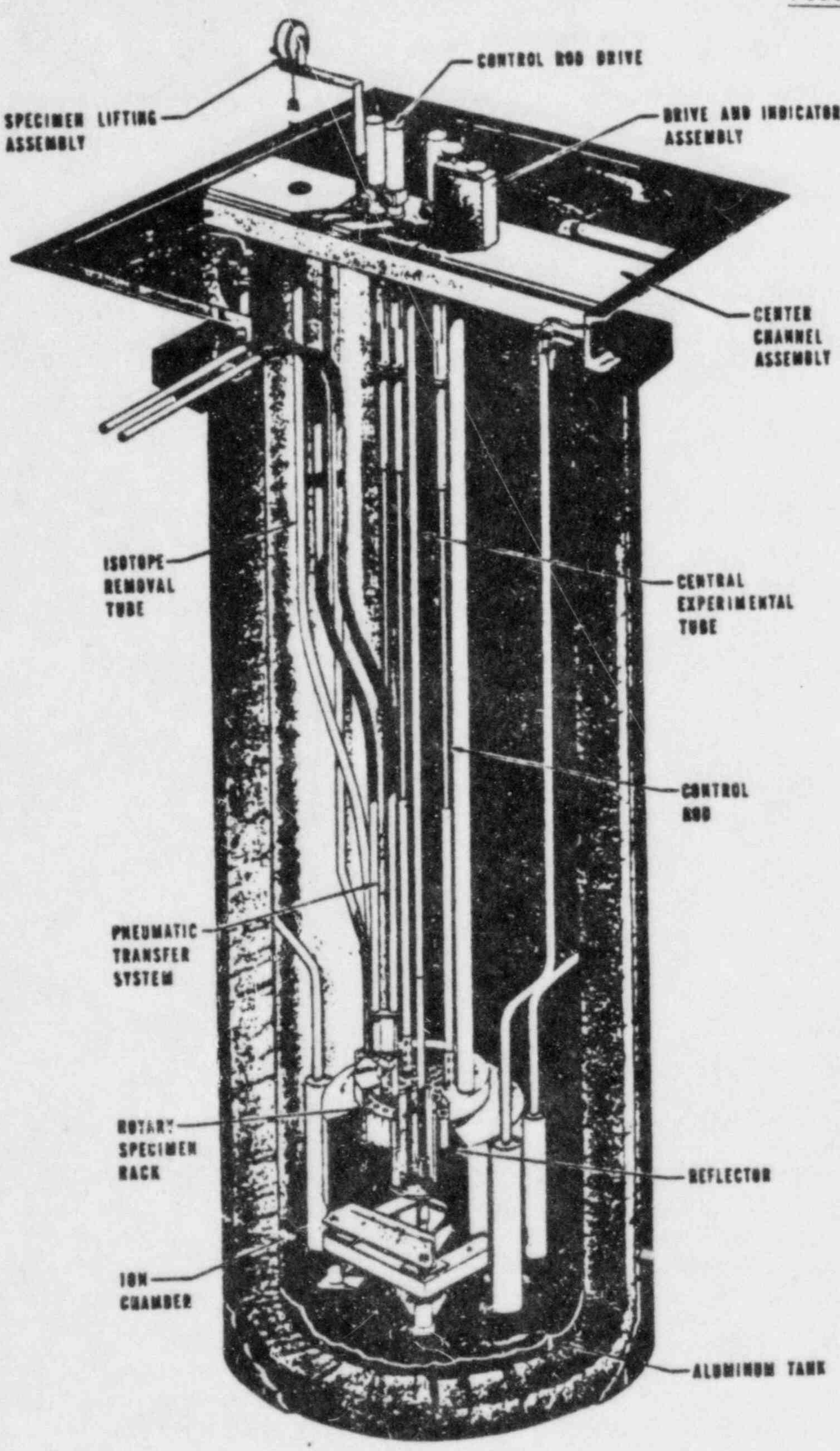
TABLE 1.1

PRINCIPAL DESIGN PARAMETERS

Parameter	Description
Reactor type	TRIGA Mark I
Maximum steady-state power level	250 kW
Maximum pulse	1.5% $\Delta k/k$ ($2.15)
Fuel element design	
Fuel-moderator material	$U\text{-}ZrH_{1.6}$*
Uranium content	8.5 wt %
Uranium enrichment	$<$ 20% U-235
Shape	Cylindrical
Length of fuel	38 cm (15 in.) overall
Diameter of fuel	3.63 cm (1.43 in.) outside diameter
Cladding material	Type 304 stainless steel
Cladding thickness	0.051 cm (0.020 in.)
Number of fuel elements	$\sim$63 (minimum core)
Excess reactivity, maximum	2.25% $\Delta k/k$ (cold, clean)
Safety-transient	1
Regulating	1
Shim	1
Total reactivity worth of rods	5.1% $\Delta k/k$
Reactor cooling	Natural convection of pool water

*The normal hydrogen-to-zirconium ratio is 1.60%, and the maximum value is 1.65%.

FIGURE 1.6



TYPICAL MARK I TRIGA REACTOR

A graphite radial reflector nominally 12 inches thick and 22 inches high surrounds the core region. Top and bottom axial reflection is provided by ~ 3.5-inches -long graphite plugs incorporated into the individual fuel elements. Figure 1.7 illustrates a typical core diagram.

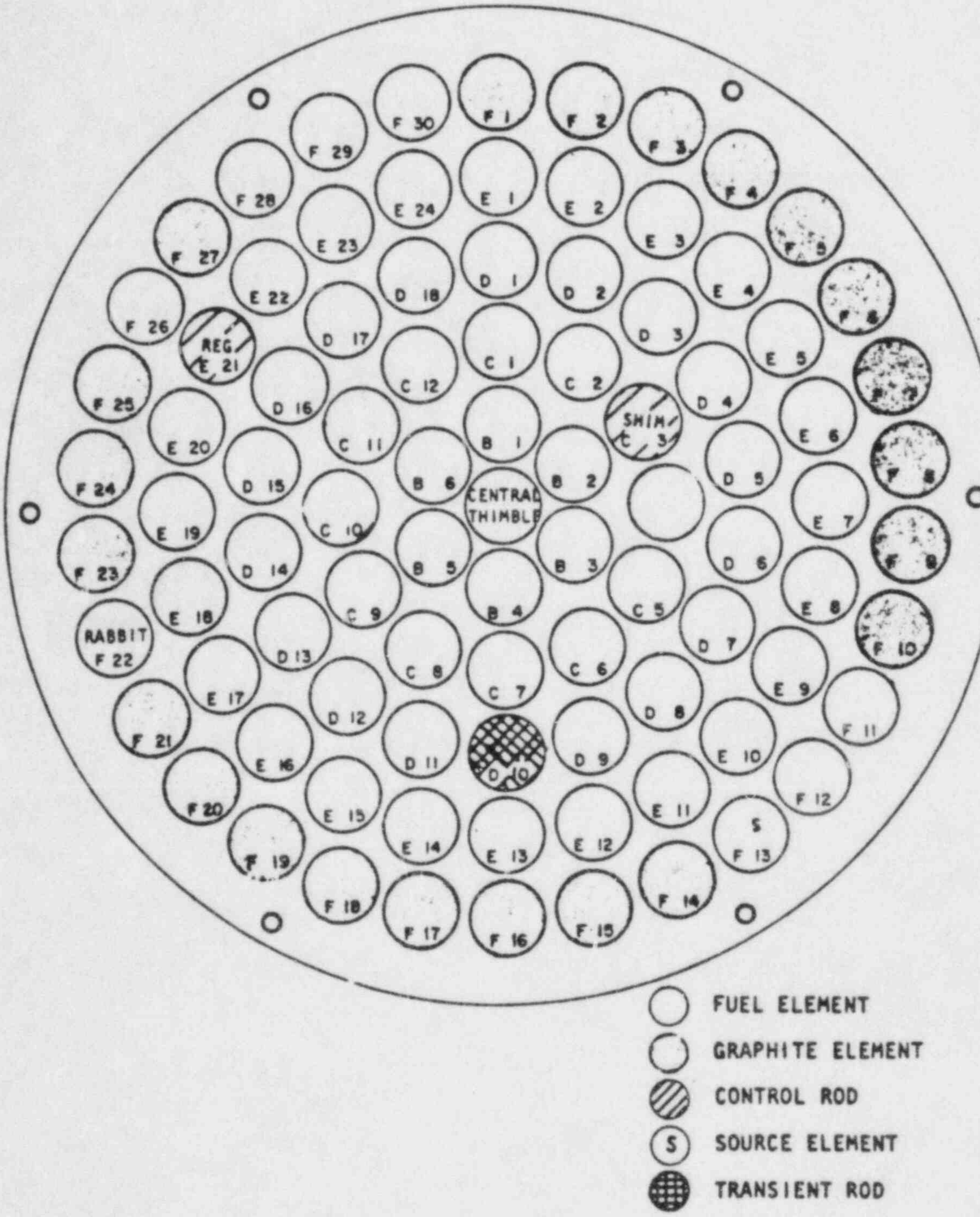
The fuel elements are positioned laterally at the top and bottom by two 0.75-inches -thick aluminum grid plates. The lower grid plate supports the weight of the fuel elements. Both grid plates are supported by pads welded to the radial graphite reflector assembly, which, in turn, is supported by an aluminum platform at the bottom of the reactor pool.

The operation of the TRIGA reactor is monitored by instrumentation channels that measure fuel element temperature and neutron flux. Thermocouples in an instrumented fuel assembly provide information on fuel material temperature during both steady-state and pulse operation. This signal is displayed in the control room and used to initiate a reactor scram if preset temperature limits are exceeded.

The reactor tank is a welded aluminum vessel located below ground level and surrounded by a reinforced concrete shield structure. The tank is in the form of an elongated cylinder 0.25 inches thick, 13 feet long, 6.5 feet wide, and 21 feet deep, with a capacity of 11,300 gallons. The outside of the tank is coated with a bituminous coating for corrosion protection. The tank assembly rests on a 2-feet thick concrete slab.

The reactor core is positioned near the bottom of the tank under 16 feet of light demineralized water, which serves as a radiation shield, neutron moderator, and reactor coolant. The natural thermal convection of the water adequately disperses the heat generated in the core during both steady-state and pulse operations. When necessary, the coolant water may be pumped

FIGURE 1.7



TYPICAL CORE DIAGRAM

through an external heat exchanger system that ultimately disposes of the heat to the atmosphere. Suspended fuel storage racks in the reactor-tank are available for routine storage of fuel elements and/or reactor components.

In addition three 10-inch diameter, 10-feet-deep storage wells outside of the reactor tank provide isolated storage for fuel elements or radiation sources. All storage racks have been designed to be critically safe for TRIGA fuel elements immersed in water.

The core is shielded horizontally by a minimum of 4 feet of ordinary concrete, 1.75 feet of water, and 1 feet of graphite. Vertical shielding is provided by 16 feet of pool water above the core. A view of the reactor tank and shield structure is shown in Figure 1.8.

1.1.2 Duration of the License and Approximate Usage

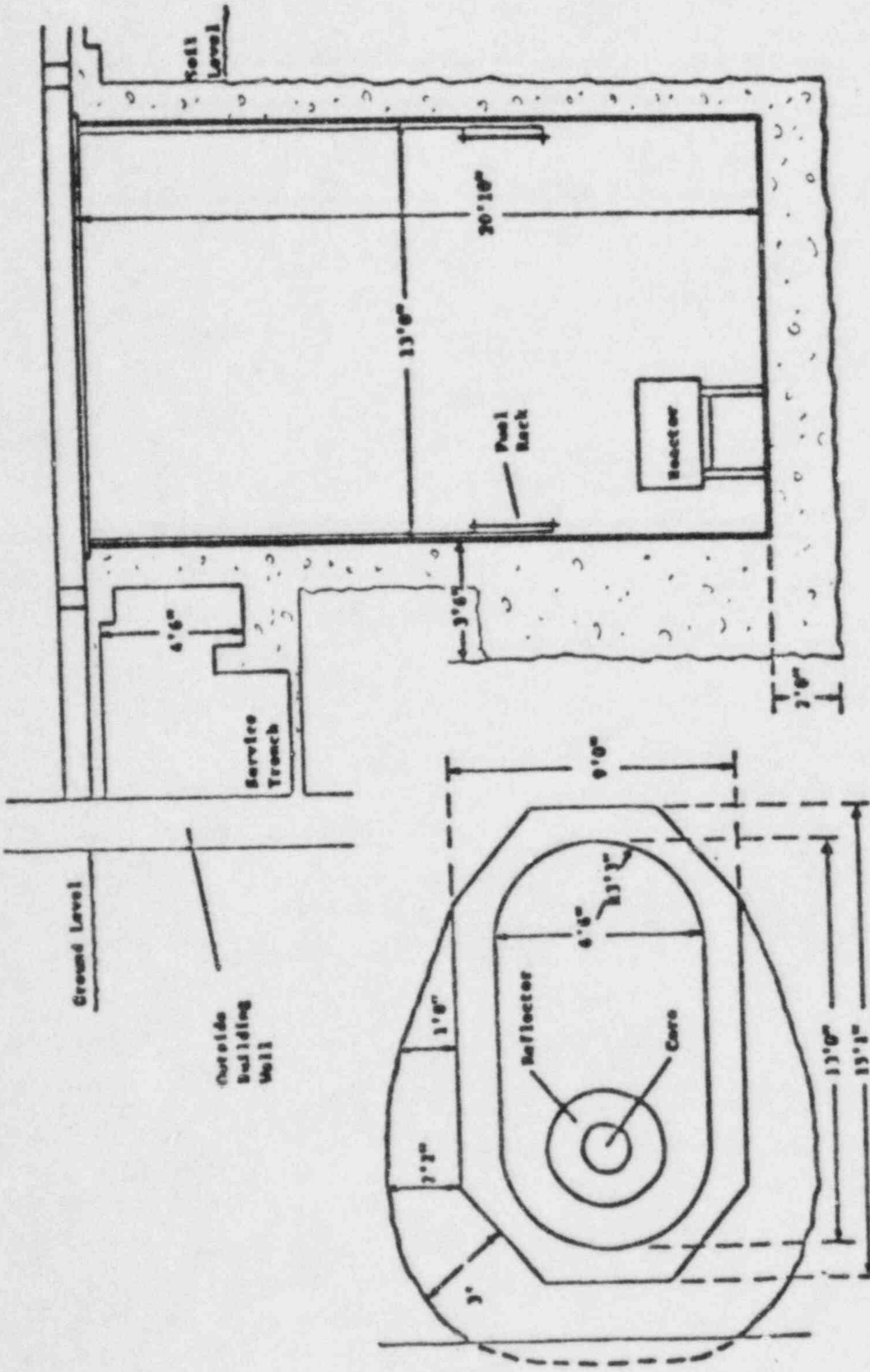
The University of Texas TRIGA reactor was installed in the Taylor Hall engineering laboratory in 1963 and was operated at an initial power level of 10 kW(th). The reactor was upgraded in 1968 to a power level of 250 kW(th).

Total hours of reactor usage from 1965 (which was the first year the utilization data were maintained) to 1983 were 2000.4 hours. Integrated burnup through 1983 is 20 Mw-days (See Table 1.2.) Estimated burnup expected at the time of the reactor shutdown is 22-24 Mw-days.

1.1.3 Synopsis of the Decommissioning Plan

The University of Texas intends to proceed with the decommissioning of Taylor Hall, Reactor Room 131 by implementing the uninterrupted dismantling and decontamination of the facility (DECON)

FIGURE 1.8



REACTOR TANK AND SHIELD STRUCTURE

TABLE 1.2

UT TRIGA REACTOR OPERATING HISTORY

Year	Burnup in Megawatt-days (Mwd)	Cumulative (Mwd)
1983	1.001	20.837
1982	.673	19.836
1981	.757	19.163
1980	.490	18.406
1979	3.026	17.916
1978	1.120	14.890
1977	.313	13.770
1976	.471	13.457
1975	.852	12.986
1974	1.152	12.134
1973	1.908	10.982
1972	2.016	9.074
1971	2.246	7.058
1970	1.520	4.812
1969	2.083	3.292
1968	1.174	1.209
1967	.025	0.035
1966	.010	0.010
1965	----	----

alternative, thus returning the site to unrestricted use. The selection of this alternative is based on the decision that Taylor Hall will be demolished and replaced by a new multistory structure for other university activities.

In order to continue its nuclear engineering research and education program the University plans to build a new facility for a TRIGA reactor at another site. When the new facility is approved, licensed and operational (estimated date spring of 1987), the University plans to remove the fuel, cobalt-60 irradiator and other radiation sources and reactor components and ship them to the new facility. These activities will be performed under the current license, after the cessation of reactor operation.

The decommissioning plan consists of several activities oriented towards dismantling of the remaining equipment, the reactor pit and liner and any associated systems and structures in a safe manner and in accordance with ALARA principles. The initial activity will be the radiation survey of the reactor room and of adjacent laboratory areas. Subsequent to the radiation survey, the remaining components of the reactor will be decontaminated, if necessary, dismantled and removed.

The major decommissioning task, the demolition of the reactor pit and liner, will proceed after the installation of a confinement barrier in the reactor room including a dedicated ventilation system to prevent the spread of airborne contaminants. The demolition will be a stepwise procedure designed to minimize worker's radiation exposure and to ensure industrial safety. Upon survey of the reactor pit, which will include borehole samplings to confirm analytical results as to the extent of neutron activation of the concrete, demolition will proceed. Demolition will begin in the area of the floor where the liner and reinforced concrete have not been neutron activated to permit safe access to the activated areas. Demolition of the activated areas of the liner and concrete floors and walls will then be conducted; however, the remaining non-activated liner and walls will be left in place.

All contaminated demolition materials will be placed in approved packaging and transportation containers and shipped to an approved disposal site. Any liquid and solid radioactive waste resulting from the decommissioning operations will be collected, processed and packaged on-site and then shipped to an approved disposal site.

At the completion of all decommissioning activities a termination radiation survey will be performed to demonstrate that the facility and site meet the criteria for unrestricted use.

1.1.4 Discussion of Decommissioning Alternatives

The University of Texas intent to reuse the Taylor Hall site for building a new facility has led to the decision to select the continuous dismantling and decontamination (DECON) decommissioning alternative. Other alternatives such as partial dismantling followed by safe storage and eventual completion of the dismantling (SAFSTOR), or safe entombment (ENTOMB) would not result in immediate release to unrestricted use of the site and are therefore not applicable to the University's needs.

1.1.5 Availability of Funds

Funds for the decommissioning of UT TRIGA reactor are allocated from university funds designated for construction projects. These funds are allocated from the separate State funds as opposed to the funds from the general University's operating budget. Plans for the new TRIGA facility include funds for dismantling of the present facility as part of a complete project. Allocation of funds by the university has been made for the preparation of construction plans of the new facility and for preparation of the decommissioning plan. After obtaining the appropriate permits based on these plans, allocation of funds will be made for the new facility construction and for the dismantling of the TRIGA reactor.

1.1.6 Major Tasks and Schedules

Major tasks to be performed during the decommissioning of the UT TRIGA reactor (following removal of fuel and other reusable reactor components), are as follows:

1. Perform a comprehensive radiation survey of the reactor room and of Taylor Hall.
2. Remove components with surface contamination only.
3. Remove components with induced radioactivity.
4. Install confinement barrier including ventilation system and associated systems.
5. Demolition of non-activated portion of the reactor tank floor.
6. Demolition of activated portion of the reactor pit.
7. Dispose or clean-up of equipment used during the dismantling operations and clean-up of Reactor Room.
8. Package and ship contaminated materials and radioactive wastes.
9. Perform a final radiation survey of the facility to demonstrate that it can be returned to unrestricted use.

   The decommissioning of the TRIGA reactor is planned to start approximately 12 months after construction of the new TRIGA facility. The projected time scale indicates spring of 1987. The duration of operations to accomplish the above dismantling and decontamination tasks is estimated to be 5-1/2 months.

1.1.7 Items Subject to Quality Assurance

For the duration of the decommissioning operations a quality assurance (QA) program will be carried out to ensure conformance with the decommissioning plan's procedures and for procurement of equipment involving personnel and public safety.

The quality assurance program will cover the following items:

- Review of new decommissioning procedures to ensure that adequate consideration is given to radiation, safety, security, QA/quality control aspects, reliability and the choice of processes and materials.

- Procurement document control to verify that any QA requirements for vital equipment and services are accurately identified in procurement documents or purchase orders.

- Formal documentation control of work instructions and procedures, drawings and information management, including changes as they occur.

1.1.8 Tasks to be Performed by Contractors

The University plans to augment its own staff with personnel provided by outside contractors for those tasks where in-house capability is not sufficient. The activities/tasks for which contractors are intended to be utilized include:

- Supervision of day-to-day decommissioning activities including direction of craft supervisors and crew leaders

- Health Physics assistance

- Quality Assurance assistance

- Crafts and labor to provide temporary construction work, perform decontamination and demolition tasks and to process, package and ship radioactive materials.

1.1.9 Final Radiation Survey Plan Summary

A final radiation survey will be conducted at the facility upon completion of all prior activities described in 1.1.6. The goal of the survey is to ensure that ambient radiation levels and surface contamination levels are well below the limits specified in Regulatory Guide 1.86 and to verify that the site can be released to unrestricted use by meeting the acceptance criteria of 5 uR/hr above background at 1 meter from the surface.

The survey will cover all pertinent structures, surfaces, systems and components focusing on those items identified as potentially troublesome during the pre-decommissioning and during the decontamination/dismantling phases.

A report will be compiled and submitted to the NRC upon completion of the survey and at least 30 days prior to the scheduled date of the site reuse for other purposes.

1.1.10 Collective Dose Equivalent

One of the goals of the ALARA program during decommissioning is to maintain the collective (man-rem) dose, i.e., the sum of doses to all persons involved in the decommissioning of the UT TRIGA Reactor, as low as reasonably achievable (ALARA).

In general, the collective dose during decommissioning will depend on:

- the number of tasks to be performed;

- the average dose rate at the location where the task is performed;

- the duration of the task; and

- the number of persons involved in the performance of the task.

The collective dose estimates for the decommissioning of the Taylor Hall TRIGA Reactor are based on the tasks listed in Section 1.1.6, the man-hours required to perform these tasks, and the expected dose rates associated with each task. Table 1.3 provides a summary of the results of these analyses. It should be noted that these estimates reflect the elimination of unnecessary exposure by implementation of the ALARA philosophy and practices of the Radiation Protection Program described in Section 2.0.

1.2 Facility Operating History

The TRIGA reactor installed at The University of Texas, Taylor Hall has been operated safely since 1963, with no accidents and only with a small number of equipment malfunctions. The malfunctions were corrected (documented in reports of Reportable Occurrences from University of Texas, Docket No. 50-192) and minor radioactivity spills or releases that occurred were promptly cleaned-up. Consequently, there are no areas in the facility which should impact on the decommissioning safety when implementing the tasks described in the decommissioning plan herein.

Activities during operation of the TRIGA reactor include instruction and research. Most facility functions consisted of reactor operation experiments and material irradiations. Typical material irradiations included neutron radiography and activation analysis. Some production of radioisotopes at infrequent intervals occurred. These activities generally caused activation to the core components and surrounding structures only. Other activities at the facility generally caused activation of solid or encapsulated materials without effects on reactor

PRELIMINARY COLLECTIVE DOSE ESTIMATE

TASK/ACTIVITY	MANPOWER REQUIREMENTS: Supervisor men	days	Craftsman men	days	Laborer men	days	Health Ph men	days	TOTAL MANPOWER man-days	EXPOSURE* PERIOD man-hrs	DOSE RATE RANGE mR/hr	TOTAL MAN-REMS Min	Max
1. Perform Comprehensive Radiation Survey (Task Duration 7 days)													
1a. Determine radiation levels in the reactor room	(1)	2	-	-	-	-	(1)	2	4	32	0.2	0.006	0.006
1b. Take borehole samples to confirm extent of activated areas	(1)	0.5	(1)	1	-	-	(1)	1	2.5	15	7.6	0.114	0.114
1c. Establish outside background rad. levels in Taylor Hall	(1)	1	-	-	-	-	(1)	3	4	32	0.2	0.006	0.064
1d. Establish outside background radiation levels	(1)	1	-	-	-	-	(1)	1	2	16	0.2	0.003	0.003
Subtotal Task 1:									12.5	87		0.129	0.129
2. Remove Components with Surface Contamination Only (Task Duration 5 days)													
2a. Disassemble components	(1)	1	(1)	1	-	-	(1)	1	3	18	0.25-2.5	0.004	0.045
2b. Remove surface contamination	(1)	1	(1)	2	(1)	2	(1)	1	6	36	0.25-2.5	0.009	0.090
2c. Package for disposal or reuse	(1)	1	(1)	2	(1)	2	(1)	1	6	36	0.25-2.5	0.009	0.090
Subtotal Task 2:									15	90		0.022	0.225
3. Remove Components with Induced Radioactivity (Task Duration 8 days)													
3a. Take dose-rate measurements	(1)	1	(1)	1	-	-	(1)	1	3	18	0.25-2.5	0.004	0.045
3b. Disassemble components (if needed)	(1)	0.5	(1)	1	(1)	1	(1)	0.5	3	18	0.25-2.5	0.004	0.045
3c. Package for disposal or reuse	(1)	1	(1)	1	(2)	1	(1)	0.5	4.5	27	2.5-5.0	0.067	0.135
3d. Remove and decontaminate water from reactor tank	(1)	1	(1)	1	(1)	1	(1)	0.5	3.5	21	0.25-2.5	0.005	0.052
3e. Remove surface contamination from reactor pit	(1)	1	(1)	2	(2)	2	(1)	1	8	48	2.5-5.0	0.120	0.240
3f. Install protective cover over reactor pit	(1)	0.5	(1)	1	(1)	0.5	-	-	2	12	0.25	0.003	0.003
Subtotal Task 3:									24	144		0.203	0.520

*Based on 6 hours spent in radioactive environment, except Task 1 (activities 1a, 1c and 1d), Task 4 and Task 9

PRELIMINARY COLLECTIVE DOSE ESTIMATE

TASK/ACTIVITY	MANPOWER REQUIREMENTS Supervisor men	days	Craftsman men	days	Laborer men	days	Health Ph men	days	TOTAL MANPOWER man-days	EXPOSURE* PERIOD man-hrs	DOSE RATE RANGE mR/hr	TOTAL MAN-REMS Min	Max
4. Install Confinement Barrier, Ventilation & Associated Systems (Task Duration 10 days)													
4a. Modify lighting to serve functional areas	(1)	0.5	(1)	1	(2)	1	-	-	3.5	28	0.25	0.007	0.007
4b. Connect utilities (water, power, etc)	(1)	1	(2)	2	(2)	2	-	-	9	72	0.25	0.018	0.018
4c. Install confinement barrier & seal joints & penetrations	(1)	1	(3)	4	(2)	4	(1)	0.5	21.5	172	0.25	0.043	0.043
4d. Install and test ventilation systems	(1)	2	(2)	3	(2)	3	(1)	1	15	120	0.25	0.030	0.030
4e. Calibrate, test and install health physics equipment	(1)	1	-	-	-	-	(2)	2	5	40	0.25	0.010	0.010
4f. Reinforce floor support to waste package/handling	(1)	2	(2)	4	(3)	4	-	-	22	176	0.25	0.044	0.044
Subtotal Task 4:									76	608		0.152	0.152
5. Demolish Non-Activated Portion of the Reactor Pit (Task Duration 5 days)													
5a. Place shield slabs over the activated area of the floor	(1)	0.25	-	-	(1)	0.25	(1)	0.25	0.75	4.5	7.6	0.034	0.034
5b. Cut and remove aluminum liner of the non-activated portion of the tank floor	(1)	2	(1)	2	(1)	2	(1)	2	8	48	2.5-5.0	0.120	0.240
5c. Demolish and remove concrete in the entire thickness (2 ft) of the portion of the reactor tank pit floor	(1)	2	(1)	2.5	(2)	2	(1)	1.5	10	60	2.5-5.0	0.150	0.300
Subtotal Task 5:									18.75	112.5		0.304	0.574
6. Demolish Activated Portion of the Reactor Pit (Task Duration 60 days)													
6a. Remove shield slabs and cover with plastic mats the non-activated areas of the floor	(1)	0.5	(1)	0.5	(1)	1	(1)	0.5	2.5	15	7.6	0.114	0.114
6b. Cut aluminum liner from activated portion of the reactor tank floor	(1)	2	(2)	3	(1)	3	(1)	1.5	12.5	75	7.6	0.570	0.570
6c. Package and remove cut-up liner from tank pit	(1)	1	(2)	1	(2)	1	(1)	0.5	5.5	33	7.6	0.251	0.251

(CONTINUED ON SHEET 3 OF 4)

PRELIMINARY COLLECTIVE DOSE ESTIMATE

TASK/ACTIVITY	MANPOWER REQUIREMENTS: Supervisor men	Supervisor days	Craftsman men	Craftsman days	Laborer men	Laborer days	Health Ph men	Health Ph days	TOTAL MANPOWER man-days	EXPOSURE* PERIOD man-hrs	DOSE RATE RANGE mR/hr	TOTAL MAN-REMS Min	TOTAL MAN-REMS Max
6d. Demolish the entire thickness of concrete from the activated portion of reactor pit floor 6e. Package and remove concrete rubble from tank pit	(1)	10	(2)	10	(1)	10	(1)	10	50	300	7.6	2.280	2.280
6f. Install work platforms on the entire reactor tank pit bottom and cover with heavy mats	(1)	1	(1)	2	(1)	2	-	-	5	30	0.9	0.027	0.027
6g. Cut activated aluminum from tank wall 6h. Package and remove cut-up liner from tank pit	(1)	2	(1)	4	(1)	4	(1)	1	11	66	0.9	0.059	0.059
6i. Demolish 20 inches of thickness from activated concrete portion of the wall 6j. Package and remove concrete rubble from tank pit	(1)	10	(2)	15	(1)	10	(1)	15	65	390	0.9	0.351	0.351
6k. Take dose rate measurements from remaining concrete	(1)	1	(1)	1	-	-	(1)	1	3	18	0.2-0.5	0.004	0.009
Subtotal Task 6:									154.5	927		3.656	3.661
7. Clean-up Reactor Room (Task Duration 28 days)													
7a. Remove surface contamination in the reactor tank pit; place cover and seal	(1)	0.5	(1)	1	(1)	1	(1)	0.5	3	18	0.9-7.6	0.016	0.137
7b Decontaminate waste packages; make dose rate check and move outside for shipping	(1)	3	(1)	4	(3)	4	(1)	4	23	138	2.5-5.0	0.345	0.690
7c Clean for reuse or for disposal contaminated tools and equipment; package for disposal	(1)	2	(1)	4	(3)	7	(1)	4	31	186	0.9-5.0	0.167	0.930
7d Remove all loose contamination from reactor room including confinement barrier	(1)	1	(1)	1	(2)	2	(1)	2	8	48	0.9-7.6	0.043	0.365
7e. Dismantle confinement barrier and package for disposal	(1)	0.5	(1)	1	(2)	2	(1)	0.5	6	36	0.25	0.009	0.009
7f. Remove surface contamination from the entire reactor room	(1)	7	-	-	(1)	14	(1)	7	28	168	0.25-2.5	0.042	0.420
Subtotal Task 7:									99	594		0.622	2.551

PRELIMINARY COLLECTIVE DOSE ESTIMATE

TASK/ACTIVITY	Supervisor men	Supervisor days	Craftsman men	Craftsman days	Laborer men	Laborer days	Health Ph men	Health Ph days	TOTAL MANPOWER man-days	EXPOSURE* PERIOD man-hrs	DOSE RATE RANGE mR/hr	TOTAL MAN-REMS Min	TOTAL MAN-REMS Max
8. Package and Ship Contaminated Materials and Radioactive Wastes													
(Task Duration - Ongoing for 4-1/2 months)													
[Packaging Tasks shown in 2(c); 3(c); 6(c,e,h,j); 7 (c,e)]													
8a. Load packages onto trailers	(1)	2	(1)	2	(1)	2	(1)	2	8	48	0.25-7.6	0.012	0.365
8b. Make final dose rate check of loaded trailers	(1)	1	-	-	-	-	(1)	1	2	12	0.25-7.6	0.003	0.091
8c. Ship packages to disposal site or to new place of use	-	-	(1)	1.25	(1)	1.25	-	-	2.5	15	2.0	0.030	0.030
Subtotal Task 8:									12.5	75		0.045	0.486
9. Perform Final Radiation Survey													
(Task Duration 19 days)													
9a. Perform radiation survey of the reactor room	(1)	2	-	-	-	-	(1)	2	4	32	0.2	0.006	0.006
9b. Perform radiation survey of Taylor Hall	(1)	2	-	-	-	-	(1)	2	4	32	0.2	0.006	0.006
9c. Perform radiation survey of out-of-doors	(1)	2	-	-	-	-	(1)	2	4	32	0.2	0.006	0.006
9d. Perform soil and water sampling	(1)	3	-	-	-	-	(1)	3	6	48	0.2	0.010	0.010
9e. Prepare final radiation survey report preparation	-	-	-	-	-	-	-	-	-	-	-	-	-
Subtotal Task 9:									18	144		0.028	0.028
Subtotal (Tasks 1 through 9)									430.25	2781.5		5.161	8.326
Additional Personnel Entering the Reactor Room													
(Task Duration - Ongoing for 4-1/2 months)													
- Decommissioning Superintendent									20	120	0.25-2.5	0.030	0.300
- QA Personnel									40	240	0.25-2.5	0.060	0.600
- Safety Officer									40	240	0.25-2.5	0.060	0.600
- Utility Operator (HEPA Filters, Water Systems, etc)									20	120	.90-7.6	0.108	0.912
Subtotal Additional Personnel:									120	720		0.258	2.412
TOTAL (Tasks and Add'l Pers.):									550.25	3501.5		5.419	10.738
Contingency - Additional 25% of total man-rems												1.354	2.684
GRAND TOTAL												6.77	13.42

room environment. Instruction and research activities with unsealed radioisotope materials, although occasionally performed, were not a significant function of the facility.

The type of facility activities combined with the integrated power history are related to both the amount of activated radioactive materials and the potential dispersal of those materials within the facility. Periodic measurements within the reactor room of air particulate samples, surface contamination and coolant activity indicate the lack of significant amounts of uncounted or non-localized radioactive materials. Air particulate sample activities of 8×10^{-11} Ci/m^3 are routinely measured. These activities consist primarily of short-lived radioisotopes from the natural decay of uranium and thorium materials in the environment. Environmental samples of air particulates indicate activities of 2×10^{-13} Ci/m^3 after decay of the short-lived radionuclides.

Periodic surveys of floors and work surfaces have measured removable activities of 9×10^{-12} $Ci/100\ cm^2$ which is 5 percent the 1000 dpm value allowed in unrestricted areas. Measurements of pool coolant samples indicate typical activity levels of 3.5×10^{-8} Ci/m^3 with values as high as 2.7×10^{-6} Ci/m^3 which normally indicates the presence of the short-lived isotope Na^{24}. Although these data indicate the general extent of non-isolated radiological material, some areas with significant concentrations may be present at very localized points within the facility.

1.3 Current Radiological Status of Facility

1.3.1 General

The current radiological status of the facility has been established by dose rate calculations based on information available at this time.

The neutron flux is first calculated at various points in the tank wall and surrounding concrete shield wall. Then, the concentration of radionuclides as a result of the neutron-activation of structural material is calculated. Finally, external dose rates at locations in the reactor tank are calculated.

The dose rates serve as the basis for:

a. determining to what extent structural components (concrete, liner) will have to be removed in order to comply with the limit of 0.005 mR/hr above background 1 meter away from the exposed surface and the other limits imposed by Regulatory Guide 1.86.

b. specifying the necessary safety measures and procedures for the various dismantling, removal, decontamination, storage and disposal operations so that exposure to personnel is maintained ALARA.

1.3.2 Neutron Flux Calculations

The calculations of the neutron flux at various points inside the reactor tank and surrounding concrete shield wall has been based on the following assumptions.

a. In the absence of other data, the maximum flux measured at the liner at the core midplane elevation, 3×10^9 n/cm^2 sec, was taken as the thermal neutron flux at the surface of the liner.

b. The liner material is 5052 Aluminum.

c. The wall material is ordinary concrete.

The reactor and the tank wall including concrete were modeled as a finite cylinder to calculate group neutron fluxes using the DOT-3.5 computer code, a two-dimensional discrete ordinates code. The

nuclear data used in this DOT-3.5 calculation were taken from the BUGLE-80 Cross Section Library for Light Water Reactor Shielding Calculation.

Calculated thermal neutron fluxes at the core midplane and at the core centerline (axial) are plotted in Figures 1.9 and 1.10 respectively.

1.3.3 Neutron Activation Analysis

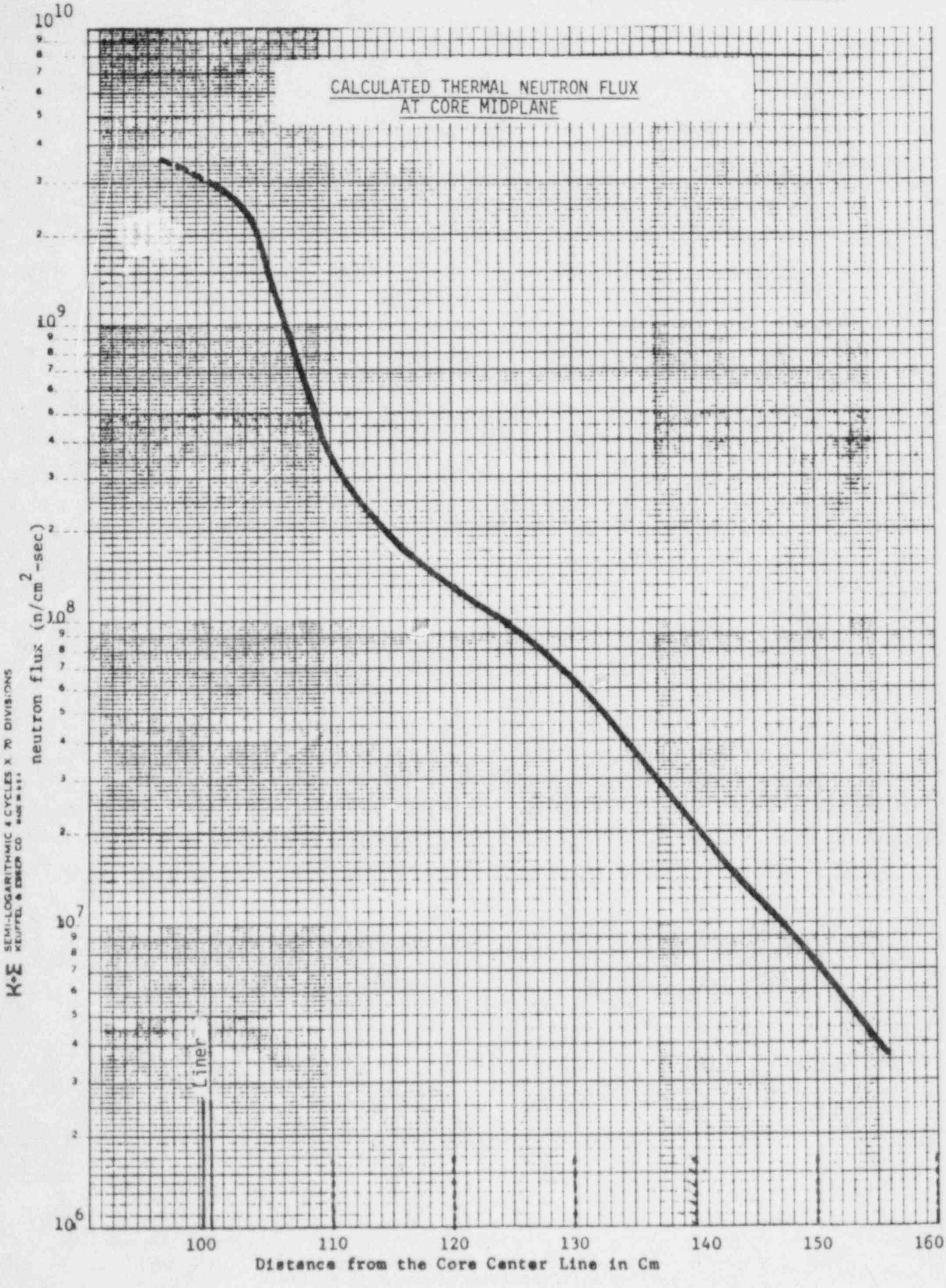
Neutron activation occurs with thermal neutrons as well as with epithermal and fast neutrons. However, for the purpose of calculating the external dose rate for the decommissioning plan, only the thermal neutron induced neutron capture reaction has been considered because it is the dominant reaction. The neutron capture cross-section for most elements follows the 1/v rule rendering fast neutron capture relatively less important compared to thermal neutron capture.

Based on the calculated neutron flux levels, the concentration of neutron-activated radionuclides in the structural material is calculated.

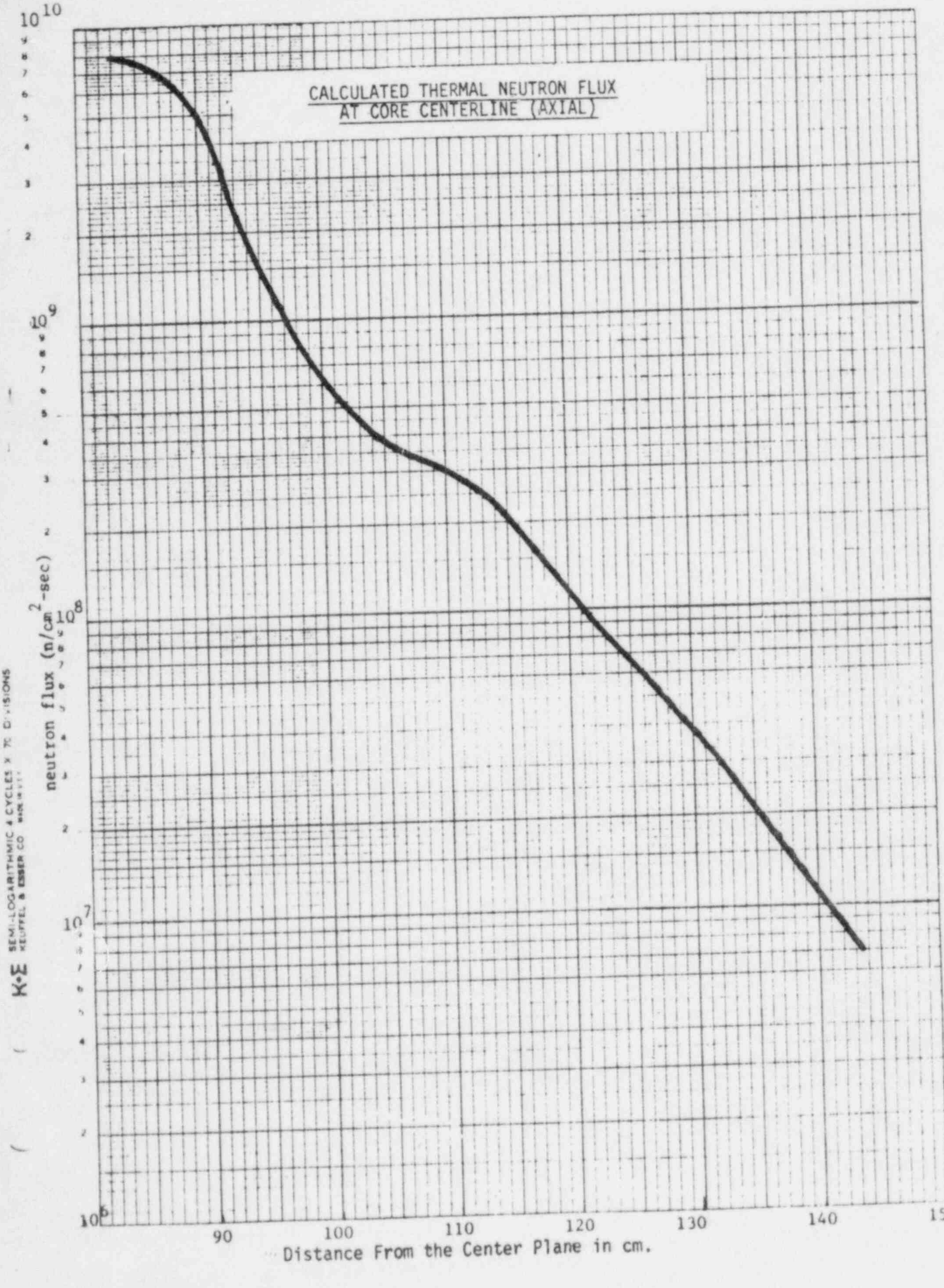
The two materials considered here are 5052 Aluminum (tank) and ordinary concrete (biological shield). Both materials contain elements whose isotopes upon absorption of neutrons are transformed into radioactive isotopes. A list of the most significant of these isotopes and their radiation characteristics is given in Table 1.4.

One rare earth element, Europium, has been included in the list. Europium exists in trace amounts in ordinary concrete: its isotope Eu-151 (abundance 47.9%) is activated by neutrons to produce the radioactive isotope Eu-152 ($t_{1/2}$ = 13.2 years). Sample measurements of the biological shield at the UT reactor have indicated a concentration of 0.3 microgram of Europium per gram of concrete.

FIGURE 1.9



CALCULATED THERMAL NEUTRON FLUX
AT CORE MIDPLANE
10^{10}
10^9
10^8
10^7
10^6
neutron flux (n/cm^2-sec)
Liner
100
110
120
130
140
150
160
Distance from the Core Center Line in Cm
K+E SEMI-LOGARITHMIC 4 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO.


CALCULATED THERMAL NEUTRON FLUX
AT CORE CENTERLINE (AXIAL)
10^{10}
10^{9}
10^{8}
10^{7}
10^{6}
neutron flux (n/cm^2-sec)
90
100
110
120
130
140
150
Distance From the Center Plane in cm.
K·E SEMI-LOGARITHMIC 4 CYCLES X 70 DIVISIONS
KEUFFEL & ESSER CO.

SIGNIFICANT NEUTRON-ACTIVATION RADIONUCLIDES

AND THEIR RADIATION CHARACTERISTICS

RADIO-NUCLIDE	PRODUCTION REACTION	HALF-LIFE	MAJOR RADIATIONS (Type, Energy in MeV, Intensity)	REMARKS
Ar-39	Ar-38 (n,γ)	270 y	γ : No γ	(1)
			β : 0.565 β^- (max)	(1)
Ca-41	Ca-40 (n,γ)	10^5 y	γ : Potassium X-rays	(1)
Ca-45	Ca-44 (n,γ)	167 d	γ : 0.0125 (< 1.0%)	(1)
			β : 0.255 β^-	(1)
Mn-54	Mn-55 (n, 2n)	303 d	γ : 0.835 (100%)	
Fe-55	Fe-54 (n,γ)	2.7 y	γ : Mn X-rays	(1)
Co-60	Co-59 (n,γ)	5.3 y	γ : 1.17 (100%), 1.33 (100%)	
			β : 1.48 β^- (max)	(1)
Ni-59	Ni-58 (n,γ)	7.5 x 10^4 y	γ : Co X-rays	(1)
Ni-63	Ni-62 (n,γ)	100 y	γ : No γ	(1)
			β : 0.067 β^- (max)	(1)
Zn-65	Zn-64 (n,γ)	245 d	γ : 1.115 (49%), Cu X-rays	
			β : 0.327 β^+ (max)	(1)
Eu-152	Eu-151 (n,γ)	13 y	γ : 0.222 (37%), 0.245 (8%), 0.344 (27%), 0.779 (14%), 0.965 (15%), 1.087 (12%), 1.1[illegible] (14%), 1.408 (22%)	
			β : 1.48 β^- (max), 0.71 β^+ (max)	(1)

Note: Sources for half-life and major radiations: Lederer, C.M., Table of Isotopes, (7th edition); U S HEW, Radiological Health Handbook, 1970.

(1) Insignificant, as far as external radiation dose is concerned, and thus not included in dose calculations.

At the time of reactor shutdown for decommissioning, it is assumed that the UT TRIGA reactor will have accumulated 22,932 kWd of total burnup. This amounts to a total of 2201 Effective Full-Power Hours (EFPH) at a power level of 250 kW for the entire lifetime of the reactor.

Using this irradiation time and the thermal neutron flux profile shown in Figures 1.9 and 1.10, the activity concentration of radionuclides produced by neutron activation in each structural component is calculated. The concentrations of significance are calculated at the time of shutdown (i.e., no decay after termination of reactor operation). However, the decommissioning activities are not expected to begin until (at least) 3-months after reactor shutdown.

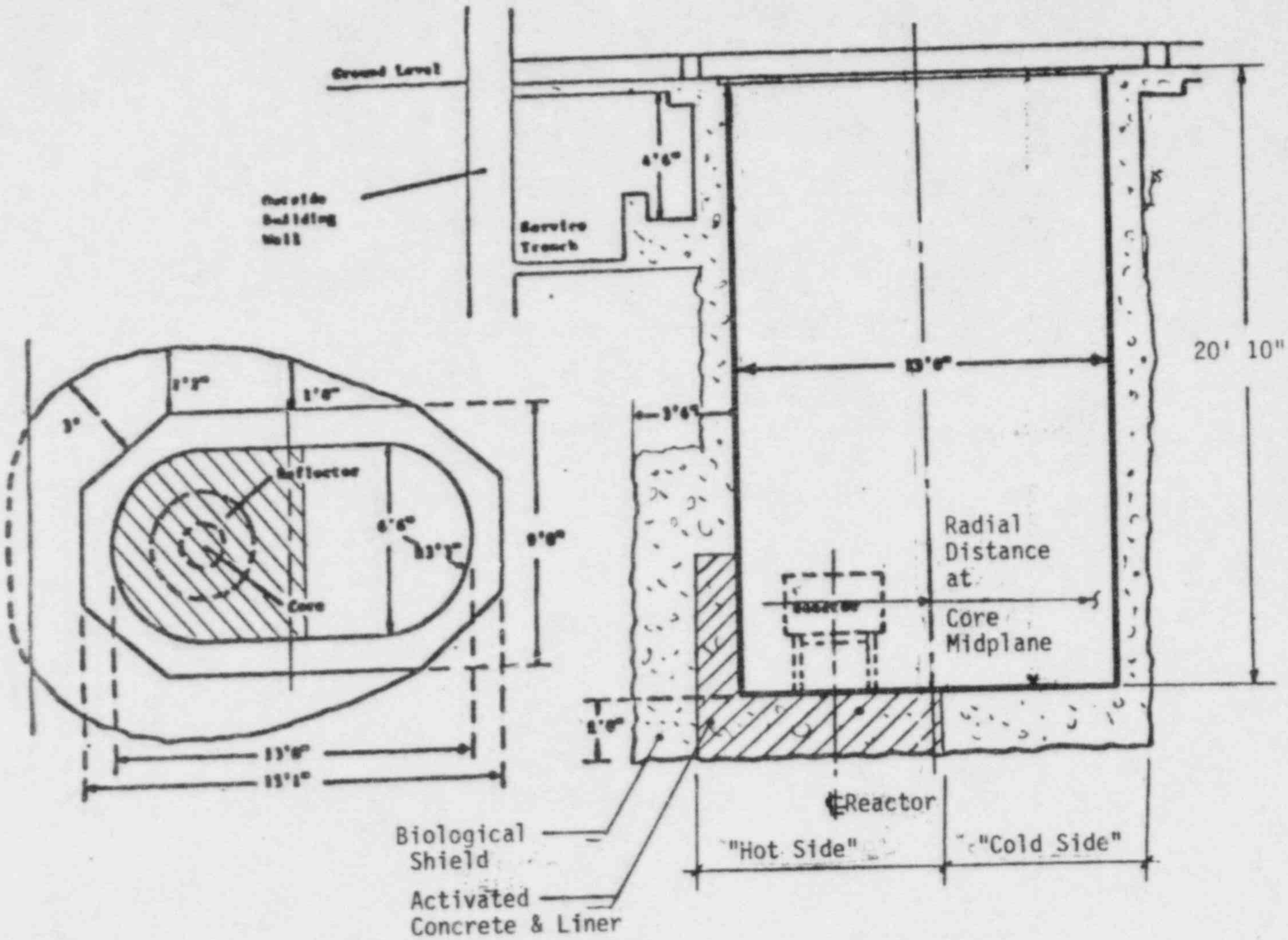
1.3.3.1 Neutron Activation of Biological Shield and Floor

The biological shield (wall) material is ordinary concrete. The elemental composition of ordinary concrete is the same as that of the ordinary concrete used in the reference research reactor of NUREG/CR-1756, "Technology, Safety and Costs of Decommissioning Reference Nuclear Research and Test Reactors". The activity concentration of various radionuclides has been calculated at various penetration depths into the concrete wall at the hot end of the tank (see Figure 1.11) and at an elevation corresponding to the reactor core midplane.

Table 1.5 lists the activity concentration of each radionuclide at various depths of concrete at the hot end of the tank and at the core midplane. The activity concentration at other locations is estimated based on the neutron flux ratio.

The results reflected in these figures refer only to the "hot side" of the reactor tank. The neutron flux at the "cold side" is estimated by extrapolating from the

FIGURE 1.11



REACTOR TANK CONFIGURATION

SHUTDOWN INVENTORY OF RADIONUCLIDES IN NEUTRON-ACTIVATED CONCRETE

ACTIVITY CONCENTRATION (MICROCURIES/CUBIC CM)

CONCRETE THICKNESS (cm)

WALL

RADIO NUCLIDE	5	15	25	35	45	55	75	105
Ar-39	3.18E-04	3.69E-05	1.93E-05	7.66E-06	2.45E-06	8.30E-07	9.44E-08	3.66E-09
Ca-41	5.78E-05	6.70E-06	3.51E-06	1.39E-06	4.46E-07	1.51E-07	1.72E-08	6.65E-10
Ca-45	2.89E-02	3.35E-03	1.76E-03	6.96E-04	2.23E-04	7.54E-05	8.58E-06	3.32E-07
Mn-54	1.39E-03	1.61E-04	8.43E-05	3.34E-05	1.07E-05	3.62E-06	4.12E-07	1.60E-08
Fe-55	2.49E-01	2.88E-02	1.51E-02	5.99E-03	1.92E-03	6.49E-04	7.38E-05	2.86E-06
Co-60	5.49E-03	6.37E-04	3.34E-04	1.32E-04	4.24E-05	1.43E-05	1.63E-06	6.31E-08
Ni-59	9.83E-06	1.14E-06	5.97E-07	2.37E-07	7.59E-08	2.56E-08	2.92E-09	1.13E-10
Ni-63	1.16E-03	1.34E-04	7.03E-05	2.79E-05	8.92E-06	3.02E-06	3.43E-07	1.33E-08
Eu-152	2.67E-03	3.10E-04	1.62E-04	6.43E-05	2.06E-05	6.97E-06	7.93E-07	3.07E-08

FLOOR

RADIO NUCLIDE	5	15	25	35	45	55
Ar-39	7.13E-04	1.09E-04	5.76E-05	1.96E-05	6.90E-06	1.91E-06
Ca-41	1.30E-04	1.98E-05	1.05E-05	3.57E-06	1.25E-06	3.47E-07
Ca-45	6.48E-02	9.89E-03	5.24E-03	1.78E-03	6.27E-04	1.74E-04
Mn-54	3.11E-03	4.75E-04	2.51E-04	8.56E-05	3.01E-05	8.33E-06
Fe-55	5.57E-01	8.51E-02	4.50E-02	1.53E-02	5.40E-03	1.49E-03
Co-60	1.23E-02	1.88E-03	9.95E-04	3.39E-04	1.19E-04	3.30E-05
Ni-59	2.20E-05	3.36E-06	1.78E-06	6.06E-07	2.13E-07	5.90E-08
Ni-63	2.59E-03	3.96E-04	2.10E-04	7.13E-05	2.51E-05	6.95E-06
Eu-152	5.98E-03	9.14E-04	4.84E-04	1.65E-04	5.80E-05	1.60E-05

TABLE 1.5

calculated values; i.e., based on the flux values calculated by DOT 3.5 for R   96.84 cm (last point in water), extrapolations are made to estimate fluxes for R   96.84 cm. Table 1.6 lists the neutron flux as a function of travel distance through water; i.e., the distance from the reactor centerline at the core midplane elevation (see Figure 1.11).

1.3.3.2 Neutron Activation of Aluminum Liner

The reactor tank material is primarily 5052 Aluminum; small components (lifting lugs, gussets, etc) are of 6061-T6 Aluminum; these are considered to have a negligable effect on the neutron activation calculations. Table 1.7 shows the elemental composition of 5052 Aluminum. Zn-65, a product of neutron activation of Zn-64, is the most significant radionuclide in the activated liner. The calculated radioactivity concentrations of Zn-65 in the liner at the core midplane and on the floor at the core axis are 7.3×10^{-3} $\mu Ci/m^3$ and 1.7×10^{-2} $\mu Ci/m^3$, respectively. The neutron activation of the trace amounts of cobalt (essentially 100% Co-59) contained in the aluminum alloy has been calculated to be significantly lower (approximately by a factor of 10^{-3}) than that of Zn-64, based on data reported in NUREG/CR-1756. The radioactivity concentration at other locations has been estimated based on the ratio of the neutron flux at each of these locations to the neutron flux at the core midplane. Table 1.8 lists the activity concentration of each radionuclide for the aluminum tank at the hot end at the core midplane and the floor below the core axis.

1.3.4 Radiation Dose Rates

The expected external dose rates inside the reactor tank were calculated based on the neutron activation analysis results. The

NEUTRON FLUX VS. RADIAL DISTANCE FROM CORE AXIS

R (cm)	FLUX (n/cm^2-s)
54.42	5.68E + 05 (1)
60.37	1.31E + 05
66.32	3.51E + 04
72.27	9.87E + 03
78.23	2.70E + 03
84.18	7.01E + 02
90.13	1.84E + 02
96.84	5.83E + 01
102.20	1.85E + 01 (2)
108.80	5.85E + 00
114.40	1.85E + 00
126.60	2.97E - 01
138.80	5.93E - 02
150.90	1.48E - 02
174.40	1.56E - 03
198.80	2.80E - 04
222.90	7.71E - 05
247.70	3.01E - 05
271.90	1.26E - 05
296.60	5.68E - 06

NOTES:

(1) For $54.42 < R < 96.84$ neutron flux was calculated by DOT 3.5 computer code.

(2) For $R > 102.20$ cm neutron flux was estimated by extrapolation.

ELEMENTAL COMPOSITION OF 5052 ALUMINUM

ELEMENT	w/o
Silicon	0.25
Iron	0.40
Copper	0.10
Manganese	0.10
Magnesium	2.50
Chromium	0.35
Zinc	0.10
Trace Elements (total)*	0.15
Aluminum	96.05

* Includes Cobalt

TABLE 1.8

SHUTDOWN INVENTORY OF RADIONUCLIDES IN NEUTRON-ACTIVATED 5052 ALUMINUM LINER

RADIONUCLIDE	ACTIVITY CONCENTRATION ($\mu Ci/cm^3$)	
	Side	Floor
Mn-54	1.0×10^{-4}	2.3×10^{-4}
Fe-55	7.1×10^{-3}	1.6×10^{-2}
Co-60	3.5×10^{-6}	8.0×10^{-6}
Ni-63	8.7×10^{-7}	2.0×10^{-6}
Zn-65	7.3×10^{-3}	1.7×10^{-2}

radiation characteristics of the radionuclides considered in the activation analysis are given in Table 1.4. For the purpose of calculating external dose rates, the most important radionuclides are Co-60, Zn-65, Mn-54 and Eu-152.

The expected maximum total dose rate inside the reactor tank is 8.3 mR/hr (7.6 mR/hr from the floor and 0.7 mR/hr from the walls).

1.3.4.1 Assumptions

The following assumptions have been made concerning calculations of dose rates:

a. The reactor core, grid plates and support structures have been removed.

b. All water has been removed from the reactor tank.

c. The reactor has been in a shutdown mode for at least 3 months prior to dismantling so that all short-lived isotopes have decayed.

d. Only the radioactive activation products in the aluminum liner and concrete biological wall are considered as radiation sources.

e. No other radiation sources (e.g., surface contamination, radioactive crud accumulation) have been considered.

1.3.4.2 Calculational Models and Results

Geometric Consideration. The physical shape of the reactor tank and the location of the reactor closer to one end of the tank (as shown in Figure 1.11), have resulted in a

non-symmetric distribution of the neutron flux, activation product concentration and dose rates.

Dose Rates from Biological Shield Wall. For external dose rate calculations, only a few of the radionuclides listed in Table 1.4 are important; namely, Co-60, Eu-152, and Mn-54, in decreasing order of importance. Among these, Co-60 is dominant, accounting for 79% of the total dose rate.

The radioactivity concentration of these radionuclides in the biological shield is a function of the radial distance into the concrete; the concentration decreases with increasing distance.

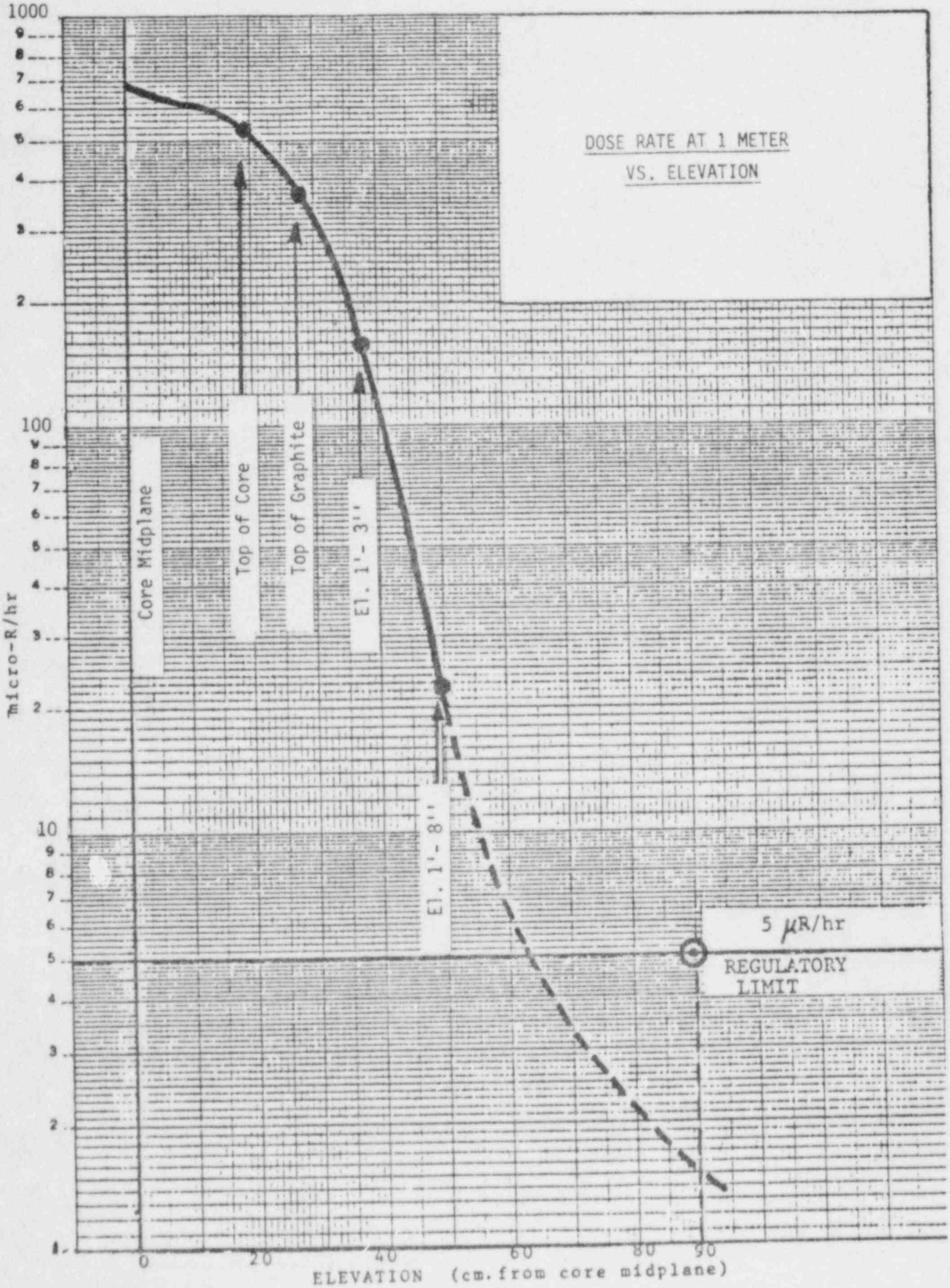
The calculational model accounts both for the spatial dependence of the activity concentration and the self-shielding in the concrete.

Geometrically, the biological shield at the "hot-end" is a half-cylinder annulus approximately of 3.5 feet (105 cm) thickness. In the dose rate calculations, the wall is modelled as a collection of unit half-ring sources.

All dose rates are calculated at a point 1 meter away from the surface. No other effect due to geometry is accounted. Consequently, the calculated values are considered conservative in this respect.

In the "hot end", the maximum 1-m dose rate from the biological shield (excluding the liner contribution) will be at the core midplane elevation and is not expected to exceed 0.7 mR/hr. This dose rate decreases as the dose point moves away from the core midplane elevation. Figures 1.12 and 1.13 depict the calculated dose rates and the dose

FIGURE 1.12


DOSE RATE AT 1 METER
VS. ELEVATION
micro-R/hr
1000
100
10
1
0
20
40
60
80
90
ELEVATION (cm. from core midplane)
Core Midplane
Top of Core
Top of Graphite
El. 1'- 3''
El. 1'- 8''
5 μR/hr
REGULATORY
LIMIT

DOSE RATE RATIO @ 1 METER VS ELEVATION

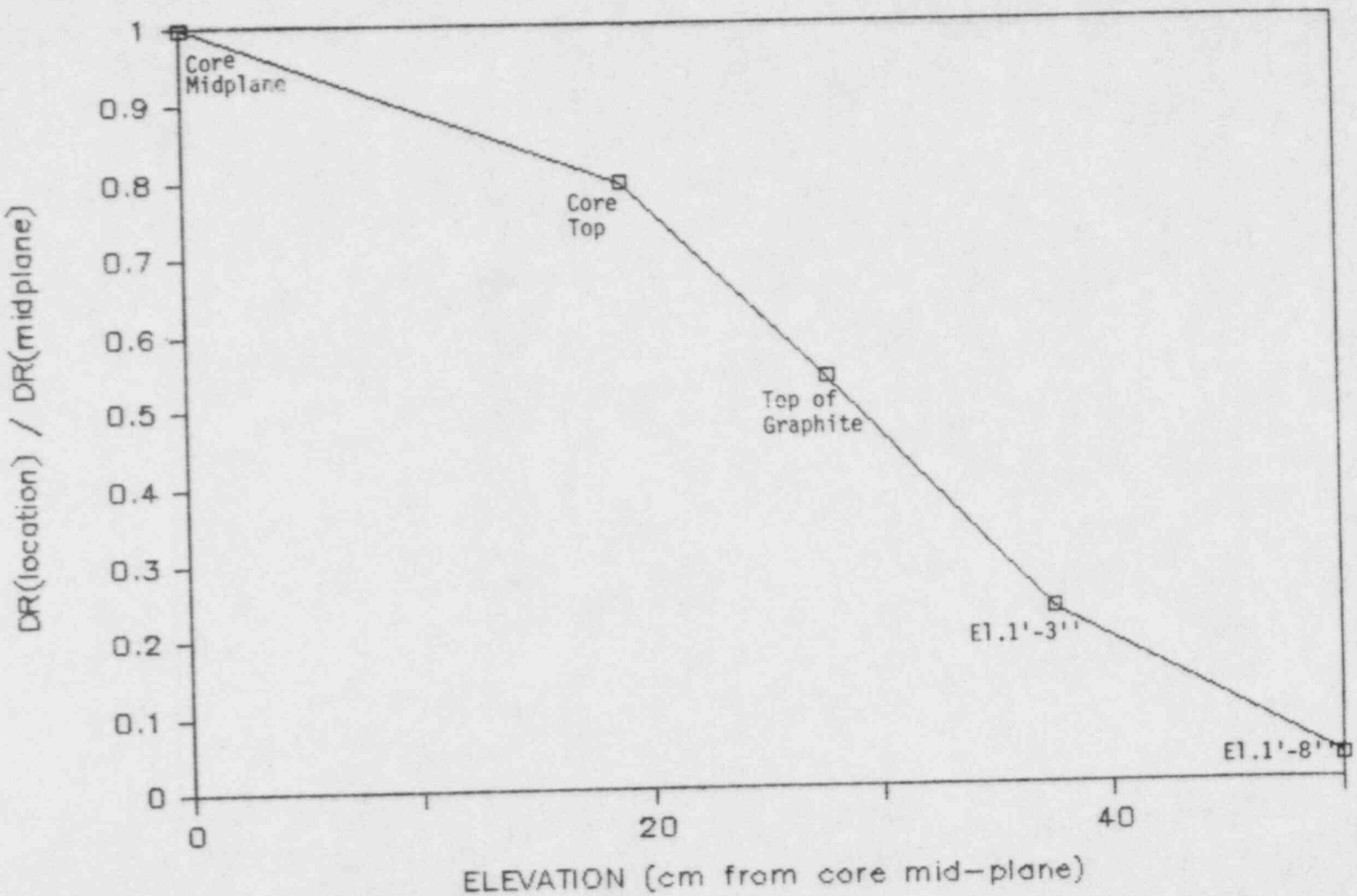


FIGURE 1.13

rate relative to the core midplane value (core midplane = 1) as a function of elevation. As indicated in Figure 1.12, the expected dose rate at 1 meter from the biological shield wall will be less than 0.005 mr/hr at an elevation of about 63 cm (25 inches) above the core midplane. However, for conservatism, a 91 cm (36 inches) distance of concrete wall above the core midplane elevation and the entire portion from the core midplane to the pit floor is assumed to need removal. Calculations indicate that 44 cm (18 inches) of the biological shield wall must be removed to result in dose rates below 5 μr/hr at 1 meter from the remaining concrete. For conservatism, a 50 cm (20 inches) depth of wall concrete removal is assumed.

At the "cold end" the corresponding dose rates become insignificant since the thermal neutron flux is estimated to decrease by a factor of 1×10^{10} at the cold end liner.

Dose Rates from Floor Slab. The concrete floor slab will have been activated by neutrons. In fact, since the neutron flux is higher in the floor slab, the activity concentration of activation products and the resulting dose rates for the floor will be higher than the corresponding values for the wall.

Since it is expected that the "cold end" will not contribute significantly to the total dose rate, the "hot end" portion of the floor is modelled as a disk source. The disk has its center at the point of intersection of the reactor centerline and the floor surface. The source strength is treated as uniformly distributed over the disk.

The same radionuclides and radiation characteristics are used as in the case of the biological shield walls. The spatial dependence of the activity concentration and the self-shielding in the concrete is accounted for.

The maximum dose rate at 1 meter from the surface is 7.5 mR/hr (excluding the liner contribution).

Radially, as the dose point moves away from the hot end on the floor, the dose rate is expected to decrease rapidly. At a distance of 5 feet (150 cm) from the reactor centerline the dose rate is insignificant.

Based on the above, it is anticipated that as a minimum, the entire 2 feet thickness of the floor slab to a distance of 4 feet (120 cm)from the reactor centerline will have to be removed to result in dose rates below 5 μR/hr at 1 meter from the remaining concrete.

Dose Rates from the Liner. The activation products in the aluminum liner will contribute to the dose rate in the tank. The dominant radionuclide is Zn-65 which accounts for over 95% of the dose rate. Again, the vertical portion of the liner is modelled as a ring source and the horizontal section as a disk source.

The maximum dose rate from the vertical portion 1 meter away is at the core midplane elevation and is 0.001 mR/hr. From the floor portion, the maximum dose rate is at the reactor-axis and is 0.07 mR/hr.

1.3.4.3 Radiation Zone Map

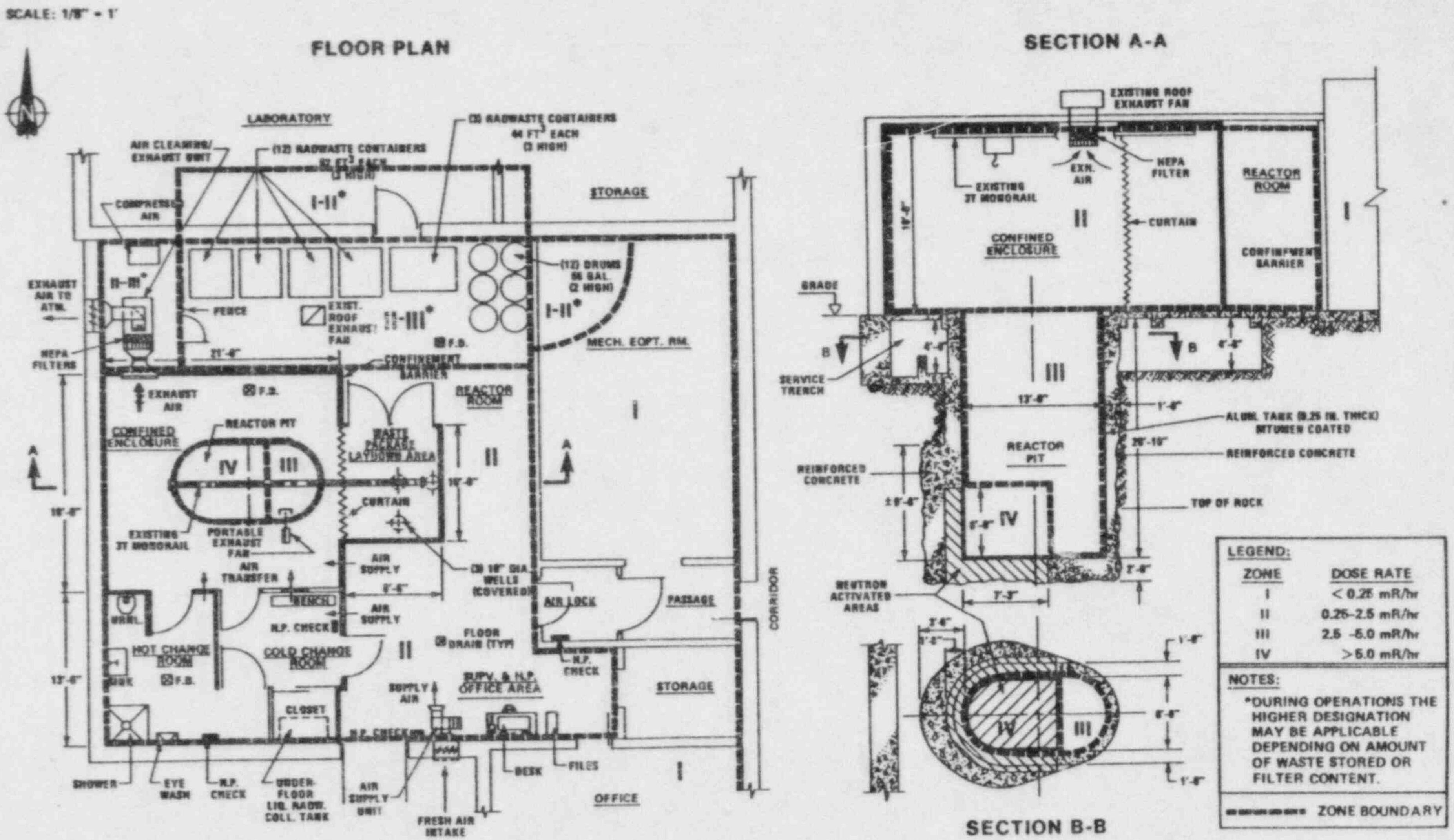
Based on the assumptions, calculation models and results discussed above, dose rate levels in the Reactor Room were estimated. Results are presented in Figure 1.14, Radiation Zone Map.

1.3.5 Inhalation Dose Rates

Bounding calculations were performed to determine maximum concentrations of the significant radionuclides (as listed in Table 1.4) resulting from

RADIATION ZONE MAP DURING DISMANTLING

Figure 1.14



complete Ar-39 release from the demolished concrete, plus maximally contaminated concrete dust levels corresponding to the Threshold Limit Value for nuisance particulates of 10^{-8} g/cm^3, as given in "TLVs, Threshold Limit Values for Chemical Substances and Physical Agents in the Work Environment and Biological Exposure Indices", published by the American Conference of Governmental Industrial Hygienists.

The concentrations were calculated both for areas within the Reactor Room and immediately outside the exhaust vent exhausting air from the Confined Enclosure located within the Reactor Room at an assumed rate of 250 to 300 cfm, through a High Efficiency Particulate Adsorber (HEPA) Filter with a decontamination factor (DF) of 100 (except for argon, with a DF of 1). All of the particulates were assumed to be respirable. The resulting Reactor Room radionuclide concentrations were found to be less than 10% of the limits of 10 CFR 20, Maximum Permissible Concentrations in air for restricted areas (Appendix B, Table I). For the area immediately outside the building exhaust vent, the radionuclide concentrations were found to be less than 10% of the 10 CFR 20 limits in air for unrestricted areas (Appendix B Table II).

The corresponding potential doses are therefore considered to be negligible both inside and outside the Reactor Room during the demolition, even if no respiratory protection such as face masks are utilized for the decommissioning workers.

1.4 Decommissioning Alternative

In order to permit unrestricted use of the facility, the University of Texas intends to implement the uninterrupted dismantling and decontamination (DECON) decommissioning alternative.

The decommissioning plan presented here includes measures to reduce radiation to levels that permit unrestricted use as follows:

- Decontamination of equipment, components and surfaces within the limits set forth in Regulatory Guide 1.86 for acceptable surface contamination for unrestricted use.

- Removal of all radioactive materials, neutron activated components and structures to meet the acceptance criteria of 5 uR/hr at 1 meter.

- Shipping off-site the radioactive materials resulting from decontamination and decommissioning of the facility.

1.5 Decommissioning Organization and Responsibilities

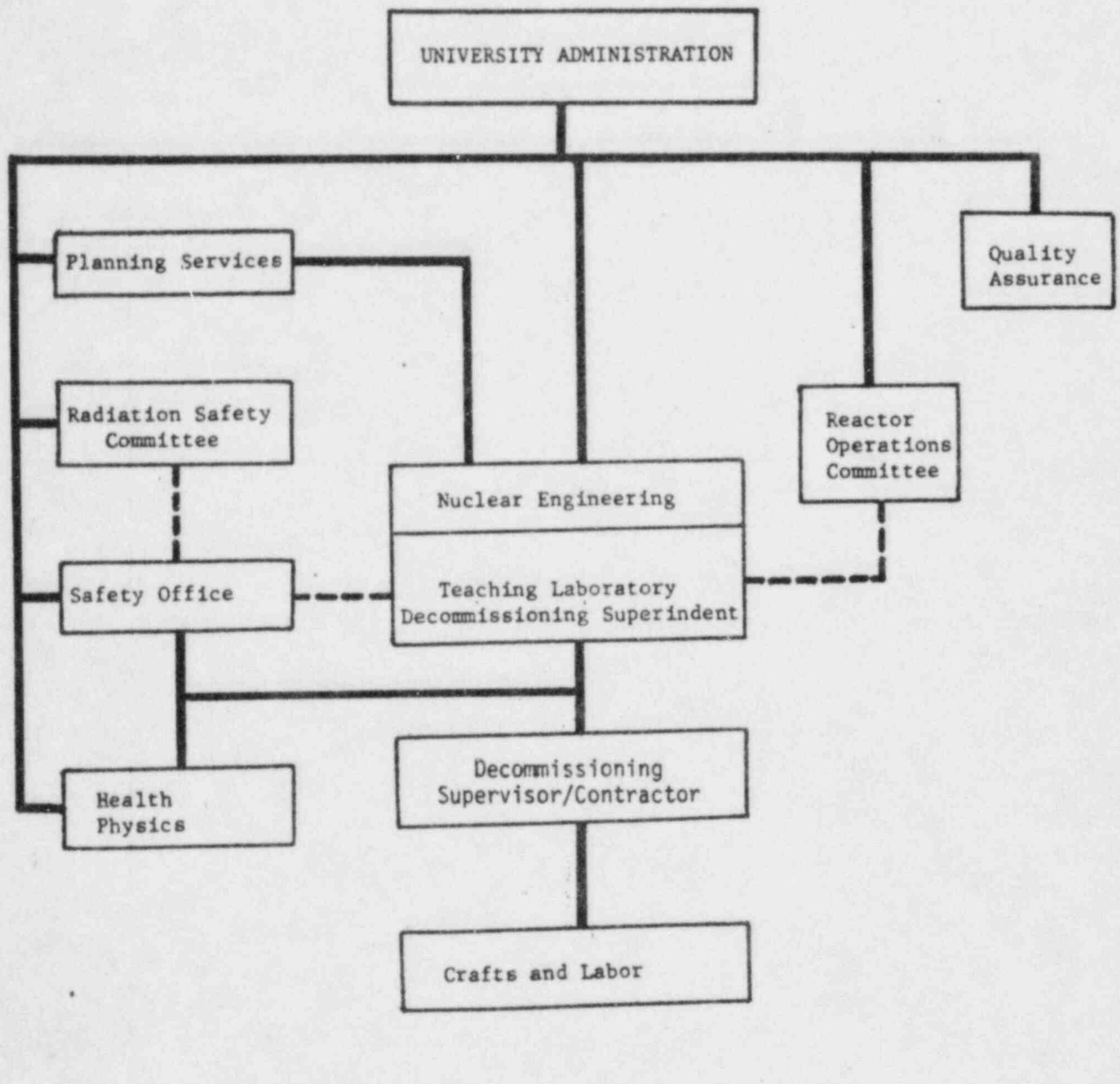
The University of Texas has planned a decommissioning organization which includes personnel with experience in reactor operation and radioactive material handling. To the largest extent possible the personnel in the planned organization are drawn from existing university personnel experienced in TRIGA reactor operations. Contractors will be employed to perform specific tasks of the decommissioning. The ultimate responsibility for decommissioning activities rests with the university administration.

The decommissioning staff line organization is illustrated in Figure 1.15. The minimum requirements for key personnel, their education and training and experience requirements are discussed below.

University Administration and Related Services. Academic administration, business administration, and services such as planning, university police and utilities are provided by persons within the university organization that are either appointed or hired to fulfill the requirements of the many university functions. Education, training and experience are determined by university policy for each of the various personnel positions. Preparation of contract documents and

FIGURE 1.15

DECOMMISSIONING ORGANIZATION



dispersal of funds are two of the responsibilities delegated to specific personnel within the University Administration and Related Services.

Reactor Operations Committee. This committee, consisting of three members knowledgeable in fields which relate to nuclear and/or radiation safety, has the role of: a) advising the university administration on the decommissioning activities; and b) providing overall review of planning and direction to the decommissioning superintendent.

The University plans to assign the decommissioning tasks review function of this committee to members of the current Reactor Committee.

Members of the Reactor Operations Committee are appointed by the Dean of the College of Engineering.

Radiation Safety Committee. This committee will advise university administration on matters of radiological and industrial safety. It will also provide review of overall planning and direction to the Safety Office on matters of radiation safety. A member of the safety office will inform the committee of the overall planning and dismantling activities.

The University plans to assign these responsibilities to the current six person committee that reviews activities of radiation safety of various University programs. The members are appointed by the President of the University.

Safety Office. Responsibilities for industrial and radiation safety programs are delegated to the staff of the Safety Office. Persons employed by the office consist of a Safety Manager, Radiation Safety Officer, Fire Marshal, Environmental Safety Specialist, and Occupational and Health Specialist. Decommissioning planning and tasks will be reviewed by the designated persons of the safety office to determine the safety specifications applied to decommissioning activities. Industrial safety requirements will be implemented by actions of safety office personnel or incorporated into the specifications for specific

dismantling tasks. A review of radiological aspects of the decommissioning activities will be made by the Radiation Safety Officer and the information provided to the Radiation Safety Committee.

Decommissioning Superintendent. This person will be responsible for the safe dismantling of the UT TRIGA reactor and will oversee all decommissioning activities. The University plans to assign this position to the Assistant Director of the Nuclear Engineering Teaching Laboratory, who has the most in-depth knowledge of the reactor's operations.

The Assistant Director functions as Supervisor of reactor operations, is licensed as a senior operator, and has advanced degrees in nuclear engineering studies. Some of the responsibilities of the Decommissioning Superintendent will include provisions for physical security, knowledge of quality assurance, and interactions with the Safety Office and Operations committee.

Decommissioning Supervisor. This person will oversee day-to-day decommissioning activities and will direct craft supervisors and crew leaders. In view of the special expertise needed in construction management, the University intends to assign this responsibility to a person provided by the contractor designated to manage both the new facility construction and the present facility dismantling. The contractor will have experience in similar construction and dismantling projects.

Crafts and Labor. Craft supervisors and crew leaders will provide execution of many of the various decommissioning tasks. Persons are to be contracted by the university that are knowledgeable of the proposed activities and that have the appropriate experience for each task. Selection assistance of available contractors for each task may be provided by the same contractor that provides the Decommissioning Supervisor.

Health Physics. A full time health physicist will be assigned to the decommissioning of UT TRIGA reactor. This person will be responsible for recommending and enforcing safety policy, both radiological and industrial. Responsibilities include maintenance of radioactive exposure records, implementation of the environmental survey program, ensuring compliance with work procedures, training and, if necessary, assigning additional health physics technicians to specific work tasks. In addition, the health physicist will be responsible for the development and implementation of the in-plant radiation protection program, the survey instrumentation program including calibration, bioassay of personnel, airborne radioactivity monitoring, supervision and documentation of radioactive waste packaging, and ALARA planning.

Quality Assurance. The Quality Assurance Supervisor will be responsible for performing and implementing the quality assurance plan for decommissioning, working with all branches of the organization. In order to insure independence of the quality assurance program this person will report directly to the University administration. This person maintains audit and job performance records and verifies that established safety review procedures are followed. A person experienced in Quality Assurance activities or trained for the specified dismantling activities will be designated to perform and implement quality assurance procedures.

1.6 Regulations, Regulatory Guides and Standards.

A number of regulatory documents and standards contain information pertinent to aspects of reactor decommissioning in general and to the operation and activities described in the Decommissioning Plan. A list of these is given below. In addition, certain University of Texas administrative procedures and limits, such as for radiation exposure of individuals during decommissioning will be applied. Reference to individual documents in this list, and specific sections thereof, is made in the appropriate sections of this plan.

Code of Federal Regulations

10 CFR Part 19	Notices, Instructions, and Reports to Workers; Inspections
10 CFR Part 20	Standards for Protection Against Radiation
10 CFR Part 30	Rules of General Applicability to Domestic Licensing of Byproduct Material
10 CFR Part 50	Domestic Licensing of Production and Utilization Facilities
10 CFR Part 51	Licensing and Regulatory Policy and Procedures for Environmental Protection
10 CFR Part 71	Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions
10 CFR Part 140	Financial Protection Requirements and Indemnity Agreements
29 CFR Part 1910	General Industry Safety and Health Standards Application to Construction
29 CFR Part 1926	Occupational Safety and Health Standards for the Construction Industry.
49 CFR Parts 170-199	Department of Transportation Hazardous Material Regulations

US NRC Regulatory Guides

1.8 Personnel Qualification and Training

1.16 Reporting of Operating Information

1.86 Termination of Operating Licenses for Nuclear Reactors

1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

8.2 Guide for Administrative Practices in Radiation Monitoring

8.3 Film Badge Performance Criteria

8.4 Direct-Reading and Indirect-Reading Pocket Dosimeters

8.6 Standard Test Procedures for Geiger-Muller Counters

8.7 Occupational Radiation Exposure Records Systems

8.8 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable

8.9 Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program

8.10 Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable

8.15 Acceptable Programs for Respiratory Protection

ANSI Standards

ANSI N13.13 Control of Radioactive Surface Contamination of Material, Equipment, and Facilities to be Released for Uncontrolled Use (DRAFT)

ANSI Z88.2-1980 Practices for Respiratory Protection

1.7 Training and Qualifications.

Training and qualifications of personnel will depend primarily on the experience of university personnel and employed contractors. Qualifications of persons that direct activities of the project will be documented and maintained until after the license termination is complete. Supplemental training will be applied on a case by case basis as required for the specific task performance. Because of the limited nature of the various tasks and the short duration expected for activities, any training required is expected to be task related.

In the event that training is required to supplement personnel qualifications, records will be kept of the task associated with the training, the persons subjected to the training and the subjects of the training effort. Conditions are not expected that would require retraining of previously trained personnel. Training responsibilities will be delegated to a person with the appropriate qualifications or knowledge related to the specific task or experience with similar tasks.

Training and qualifications of personnel will be reviewed by the decommissioning and dismantling Superintendent.

CHAPTER 2

2.0 OCCUPATIONAL AND RADIATION PROTECTION PROGRAMS

2.1 Radiation Protection Program

The purpose of a Radiation Protection Program (RPP) for the UT TRIGA reactor decommissioning project is to translate and codify UT's commitment to ALARA policy into a set of procedures and practices for the performance of the tasks involved in the project.

An effective RPP consists of all actions and measures planned or taken to protect workers and the environment, to monitor radiation and radioactive materials, control distribution and releases of radioactive materials and keep radiation exposure-individual and collective-within the limits of 10CFR20 and at levels as low as reasonably achievable.

This section presents a brief discussion of the major parts of the Radiation Protection Program. These are: (i) Objectives; (ii) Organization and Personnel; (iii) Training; (iv) Administrative and Radiological Controls; (v) implementing the RPP; (vi) Radiation Protection Facilities, Instrumentation, and Equipment.

2.1.1 Objectives of the Radiation Protection Program

The principal objectives of the RPP are:

a. To ensure the radiological safety of personnel.

b. To identify and separate contaminated from non-contaminated structures, surfaces, systems and components.

c. To dispose properly and safely of contaminated and non-contaminated components.

d. Finally, to ensure that the facility meets all radiological requirements and is ready to be released to unrestricted use.

2.1.2 Organization and Personnel

Given the resources and structure of the licensee and the anticipated duration of the decommissioning project, the organizational aspects of the RPP will maximize utilization of existing resources/personnel at UV and, at the same time, complement them with experienced personnel from outside contractors.

This structure is detailed in Section 1.5 and shown in Figure 1.15.

2.1.3 Training

The RPP includes a radiation protection training program for all persons who will be involved in the decommissioning project.

The primary objective of the radiation protection training program is to accomplish the following:

a. Ensure that all involved personnel are instructed about radiation, its source and types, radiation exposure and its effects.

b. Provide information on the Radiation Protection Program for the decommissioning projects so as to enable each person to comply with the safety rules and respond properly to all conditions.

c. Provide instruction in the fundamentals of radiation protection that will enable individuals to keep their own exposure and collective exposure ALARA.

d. Provide information on the radiation protection devices, instruments and equipment available and how to use them.

e. Inform each person about NRC, state, and local license regulations and requirements.

Although all persons should receive training it is evident that not all persons should receive the same type of training. To prevent duplication and to make efficient use of time, project personnel will be grouped in a few categories and will be given training commensurate with potential radiological problems to be encountered in their scope of work.

Typically, it is anticipated that at least three groups will be formed:

1. non-radiation workers;

2. workers directly involved in handling radioactive/contaminated materials and entering radiation areas; and

3. persons directing the activities of those in group 2.

Training will be conducted by the Health Physicist or a person designated by him. Records of training will be maintained which will include: trainee's name, date, type of training, equipment in which training was received, and instructor's name.

2.1.4 Administrative and Radiological Controls

Administrative and radiological controls comprise the measures taken to limit unnecessary exposure and spread of contamination, and the numerical limits for exposure.

2.1.4.1 Exposure Limits

Limits on the radiation exposure of individual workers involved in radiation-related worke have been set for the nuclear industry by the USNRC and are applicable here. These limits are stated in 10CFR20, "Standards for Protection Against Radiation".

However, in order to ensure that individual and collective doses are kept ALARA, the UT Administration through its Radiation Safety Committee has proposed additional administrative limits for the radiation exposure of individuals during the decommissioning of the TRIGA reactor.

Table 2.1 lists the regulatory and administrative limits for the radiation exposure of individuals during decommissioning.

2.1.4.2 Access Control and Other Measures

The University will institute the following measures:

a. Based on radiological surveys and analytical calculations, develop updated "radiation maps" for the facility.

b. Arrange the available work area into segregated sections (e.g., contaminated, "clean", working area, examination area).

c. Institute and implement access controls to:

    i. control the spread of contamination from contaminated to "clean" areas;

    ii. limit access to all personnel who are not directly involved in the specific task.

d. Clearly identify and mark all removed items and note their place of origin and any other radiological information available.

REGULATORY AND THE UNIVERSITY OF TEXAS ADMINISTRATIVE LIMITS FOR RADIATION EXPOSURE OF INDIVIDUALS DURING DECOMMISSIONING

WHOLE BODY DOSES

Regulatory Limits	- As specified in 10 CFR 20.101
Administrative Policy	- Same as 10 CFR 20.101, except doses not to be extended beyond an annual limit of 5 Rem.*

*Requires dose history of individual.

e. Package contaminated wastes in appropriate containers (as prescribed by NRC and DOT regulations) and maintain accurate records throughout this operation.

f. Monitor work areas so as to alert personnel of any unexpected changes in the radiological conditions.

g. Ensure that all workers are familiar with the personnel monitoring and protective devices and know how to use them.

h. Maintain accurate and updated records of personnel exposure, surveys, and lessons learned so as to improve and revise procedures as necessary.

2.1.4.3 Radiation Work Permit

A Radiation Work Permit (RWP) is required for all work during which workers may be exposed to radiation or contamination in excess of prescribed exposure limits.

Conditions are not anticipated where a work permit will be required.

Authority to issue a RWP rests with the Health Physicist upon consultation with the Decommissioning Superintendent and the Radiation Safety Committee if he deems necessary.

2.1.5 Implementing the Radiation Protection Program

Most of the radiation dose will be received during the dismantling and decontamination activities. Consequently, detailed ALARA procedures must be implemented to minimize exposures associated

with these activities. Some of the general guidelines for these detailed procedures are described below:

Preparation and Planning

a. Before performing a task in a radiation area:

- conduct a radiological survey of the work area to determine current radiation levels, and utilize as much information as is available to plan the task in advance and review it with the workers to identify possible hazards.

- make sure that all necessary equipment are available and close at hand.

- make sure that all monitoring equipment and instruments are functioning properly.

b. During Operations:

- monitor the radiological conditions in the work area to ensure that no sudden changes have occurred.

- maintain a reliable channel of communication between the working team and "outside" coordinators.

- provide adequate supervision and radiation protection surveillance.

c. After Operations:

- After the task is completed collect all pertinent information and update all records.

- An inventory of equipment/instruments used should be taken and their operational status reviewed.

2.1.6 Radiation Protection Facilities, Instrumentation and Equipment

2.1.6.1 Facilities

Supporting facilities will be provided for the implementation of a comprehensive radiation protection program to include the following:

a. Facilities and equipment to clean, repair, and decontaminate personnel protective equipment, monitoring instruments, tools or other material.

b. Change room(s) which allow for the segregation of contaminated from non-contaminated clothing.

c. Control stations for entrance or exit of personnel into radiation or contaminated areas; for movement of radioactive waste material; and for movement of potentially contaminated equipment and instruments.

d. Equipment to facilitate communication between workers and supervisory personnel between radiation and non-radiation areas.

e. Calibration facilities for the instruments which will be used during decommissioning.

It is intended to make maximum utilization of the facilities already existing at the University of Texas so as to minimize the cost impact and to take advantage of the university staff experience.

City services, such as those provided by the Fire Department and Emergency Ambulance will be employed, should events requiring them arise during the decommissioning

activities. Within ten minutes driving distance from UT Campus are two hospitals with which the University will arrange to provide care of injured workers, if needed.

2.1.6.2 Instrumentation

A wide range of portable and non-portable instruments and lab-counting equipment will be used during the decommissioning for radiation surveys, radioactive contamination surveys, personnel monitoring, area monitoring, air monitoring and sample analysis. Table 2.2 lists types of instruments/equipment, expected to be used during decommissioning activities described in this plan.

Calibration of portable and non-portable radiation protection instruments will be conducted in accordance with established guides and practices.

In general, The University performs semi-annual calibrations; secondary calibrations should be performed quarterly or more frequently as needed. The calibration source strength should be sufficient to encompass those dose rate ranges of the instrument that are normally expected to be encountered during the decommissioning activities; at least two points for each range shall be included. Detailed calibration records (including date, method, source description, results and person) will be kept as quality assurance records and will be auditable under a quality assurance program.

On a daily basis, or as frequently as required, each type of instrumentation will be checked to verify its proper functioning.

TABLE 2.2

RADIATION SURVEY AND MONITORING INSTRUMENTATION/EQUIPMENT

The following typical instrumentation and equipment will be used during the decommissioning:

- Portable air ionization changes
- Thin window portable Geiger tube detectors
- Portable scintillation sodium iodide detector
- Alpha-beta proportional counter and sample changer
- Liquid scintillation counter and sample changer
- High purity germanium detector and changer
- X-ray energy dispersive Si(Li) detector
- TLD reactor exposure bulbs and mini rods
- Air samplers
- Beta-gamma survey meters with frisker probes
- Hand and foot monitors
- Area monitors
- Pocket dosimeters

2.1.6.3 Radiation Protection Equipment

Other radiation protection equipment will be available for use as needed. Typically, such equipment are:

- Anti-Contamination Clothing

- Respiratory Protection Devices

- Contamination Control equipment, such as hoods, plastic containers, bags, filters

- Signs, Labels, Tags

- Special Tools

- Decontamination Equipment

- Mobile or Temporary Shields

2.2 Industrial Safety and Hygiene

The industrial health and hygiene measures for workers during decommissioning activities will be implemented in accordance with 29CFR Part 1910, General Industry Safety and Health Standards Application to Construction, and with 29 CFR Part 1926, Occupational Safety and Health Standards for the Construction Industry.

This section presents the major provisions for industrial safety and hygiene which will be implemented during decommissioning activities. Several of these provisions were already considered in the task/activity concepts presented in this decommissioning plan and will be further applied during the detailed procedures development.

2.2.1 Occupational Health and Environmental Control

The occupational health of workers during decommissioning activities will be protected by providing adequate facilities and systems as follows:

- First aid supplies within work area.
- Medical facilities availability within short (10 min) driving distance.
- Transportation equipment for injured person.
- Environmental controls in the work space to include adequate ventilation and dust control, illumination, noise control, potable water and sanitary facilities.
- Radiation protection controls (discussed in Section 2.1)

2.2.2 Personal Protective Devices

Protective devices provided for workers involved in decommissioning activities will include:

- Helmets for protection against impact and penetration of falling or flying objects.
- Hearing protection devices for the demolition workers.
- Eye and face protection for workers exposed to injury from physical, chemical or radiation agents.
- Respiratory protection devices.

2.2.3 Fire Protection and Prevention

Fire protection devices will be made available in the Reactor Room during decommissioning tasks. Portable fire extinguishers charged with dry chemical will be strategically located in the Reactor Room to serve areas as partitioned for the various decommissioning activities.

Fire prevention measures will be implemented to avoid ignition hazards from electrical wiring and equipment and from combustible materials. No smoking will be permitted in areas where open flame is present.

2.2.4 Hand and Power Tools and Cutting Equipment

The condition of the hand and power tools used during decommissioning activities will be routinely checked for proper operation and for utilization in compliance with the applicable provisions of 29 CFR 1926, Subpart I, Tools-Hand and Power, and 29 CFR 1926, Subpart J.

2.2.5 Lifting Equipment

Lifting equipment (monorail, fork lift and truck loading crane) used in the decommissioning activities will comply with the applicable provisions of 29 CFR 1926, Subpart N, Cranes, Derricks, Hoists, Elevators and Conveyors. These provisions include:

- Compliance with the manufacturer's specifications and limitations applicable to equipment operation.

- Posting of rated load capacities, operating speeds, special hazard warnings or instructions.

- Inspection of equipment by competent personnel prior to each use and during use to make sure it is in safe operating condition.

- Limiting the travel of rail mounted equipment with limit switches.

- Removing from service any equipment which has damaged wire ropes.

2.2.6 Excavations

Excavations required during decommissioning activities will comply with applicable provisions of 29 CFR 1926, Subpart P, Excavations, Trenching and Shoring. These provisions include:

- Protection of workers with personal protective devices as discussed in Section 2.2.2 above.

- Provisions to prevent workers from standing under loads handled by lifting equipment.

- Daily inspection of excavations by competent personnel for evidence of cave-ins or slides.

- Supporting systems (e.g., underpinning, etc.) designed by qualified engineering personnel and inspected daily.

- Excavated materials and other material stored at least 2 feet from the edge of the excavation (i.e., reactor pit edge).

- When using heavy equipment (e.g., fork lift) in the vicinity of excavation, the sides of the excavation (i.e., reactor pit edges) braced to resist extra pressure by superimposed loads.

- Adequate barrier physical protection provided around the excavated area.

- Ladders and scaffolding used in excavation complying with the applicable provisions of 29 CFR 1926, Subpart L, Ladders and Scaffolding.

2.3 Contractor Assistance

Outside contractors will be used to supplement the University's own staff for the following activities:

- Supervision of day-to-day decommissioning activities including direction of craft supervisors and crew leaders;

- Health Physics assistance, including equipment installation, calibration, and testing; and conduct of radiological surveys;

- Quality Assurance assistance, including procedures preparation and quality compliance during decommissioning; and

- Crafts and labor to provide temporary construction work, perform decontamination and demolition tasks and to process, package and ship radioactive materials.

These will be on-going activities during the entire decommissioning period, with the University personnel overseeing and reviewing the work as it takes place. The University will retain responsibility for health and safety during all aspects of decommissioning.

2.4 Cost Estimate and Funding

This section will be submitted later as an amendment.

CHAPTER 3

3.0 DISMANTLING AND DECONTAMINATION TASKS AND SCHEDULES

The University of Texas plans to relocate operation of the TRIGA reactor at Taylor Hall while planning a new facility for the reactor at another site. The interrelation of these two plans is important in that, once approved and licensed, the new facility is planned to accept major components and radiation sources from the Taylor Hall TRIGA reactor facility. The University plans to remove these radiation sources under its current license. The decommissioning plan herein covers the dismantling and decontamination tasks associated with the balance of equipment and structures remaining after removal of particular radiation sources. After license termination, the vacated reactor room and portions of Taylor Hall are scheduled to be demolished and a new multistory structure will be erected on the site.

3.1 Components to be Removed Prior to Dismantling

Reactor components and other radioactive sources will be removed under the current operating license. These include the TRIGA fuel elements, the cobalt-60 irradiator, other radiation sources and experimental components as indicated in Table 3.0 below. The disassembly of these components is expected to be accomplished by The University staff.

3.2 Components to be Removed During Dismantling

After components listed in Table 3.0 have been removed, the remaining equipment and structures to be dismantled under this plan can be categorized into four groupings:

Group 1 Equipment which does not have induced radioactivity but which may have surface contamination.

Group 2 Core components and other components which have induced radioactivity (excluding the reactor tank).

TABLE 3-0

COMPONENTS REMOVED PRIOR TO DISMANTLING

- 92 U-$ZrH_{1.6}$ TRIGA fuel elements with 8.5 wt % U
- 156 cobalt-60 stainless steel encapsulated sources (1.3 diameter x 28.6 cm capsules)
- 25 TRIGA graphite dummy elements
- 1 pneumatic core terminal and control system and tubes
- 1 center tube assembly and extension tube
- 1 rod oscillator assembly and control motor
- 1 cobalt-60 irradiator platform tubes and canisters
- fuel racks
- fission chamber neutron detectors
- ionization chamber neutron detectors
- 1 transient rod drive mechanism and control rod
- 2 motor drive mechanisms and control rods
- heat exchanger and primary piping

Group 3 Reactor tank liner, anchors and concrete in the proximity of the former location of the reactor core and which have been neutron activated.

Group 4 Equipment tools and systems which have been contaminated during decommissioning operations.

Tables 3.1 and 3.2 list equipment and components to be dismantled for Groups 1 and 2 above. Table 3.3 lists materials and quantities to be dismantled for Group 3, and Table 3.4 indicates types of equipment used during decommissioning.

TABLE 3.1

COMPONENTS WITH POTENTIAL SURFACE CONTAMINATION

(Group 1)

- o Purification System:
  - purification loop and deionizer tank
  - piping
  - demineralizer

- o Other components:
  - cables and conduits
  - pool deck plates
  - rotary rack drive
  - reactor bridge structure

TABLE 3.2

COMPONENTS WITH INDUCED RADIOACTIVITY

(Group 2)

- Rotary specimen rack
- Control rod guide tubes and detector tubes
- Top grid plate
- Bottom grid plate
- Reflector
- Core support
- Fasteners and connectors

TABLE 3.3

REACTOR TANK ACTIVATED COMPONENTS

(Group 3)

- Reactor pit liner
- Concrete
- Anchors
- Reinforcement bars

TABLE 3.4

EQUIPMENT USED IN AID OF DECOMMISSIONING OPERATIONS

(Group 4)

- General ventilation system
- Localized ventilation system
- Confinement barrier
- Waste treatment equipment
- Contaminated tools
- Contaminated clothing
- Monorail

3.3 Tasks Description and Analyses

The dismantling and decontamination tasks are planned in a sequence oriented towards limiting the spread of radioactivity, insuring personnel safety and as low as reasonably achievable doses, and providing schedule and cost efficiency. These tasks are described and analyzed below. Figure 3.1 illustrates the current layout and general arrangement of the reactor room.

Task 1: Comprehensive Radiation Survey.

The initial decommissioning activity will be a comprehensive radiation survey of the reactor room, the balance of the Taylor Hall building, and the immediate outdoor environment of the building. The goals of the survey are:

- to establish radiation levels in areas where decommissioning work will be performed so that detailed and comprehensive radiation protection procedures can be developed;

- to confirm the results of analytical computations so that final decisions can be made as to how much activated structural material (concrete, aluminum liner) will be removed;

- to establish "background" radiation levels in other areas of Taylor Hall which may be occupied during the decommissioning;

  to establish "background" radiation levels in the immediate outdoor environment so that the radiological impact (e.g., contamination) of decommissioning activities on the surroundings can be assessed.

It is understood that various other activities which may occur concurrently at the site (e.g., construction, excavation, etc) may contribute to the background radiation levels inside and outside Taylor Hall. Such contributing factors will be considered when establishing the "background" radiation levels.

In order to achieve these goals, the following tasks will be performed. Included in the description of each task is a brief discussion of the radiological health problems anticipated during the performance of the task.

a. <u>Determination of Radiation Levels in the Reactor Room</u>.

- A gross-gamma and contamination survey shall be performed of the reactor room (including the reactor pit), especially in those areas which, based on the computational results, are expected to be highly radioactive.

- Subsequent detailed beta-gamma and contamination surveys of the entire area will be planned and conducted to prevent undue exposure to personnel and to identify and assess "hot spots".

- Removable surface contamination levels will be determined by wipe tests. Wipe tests will also be used when a high background radiation level exists due to other sources in the vicinity.

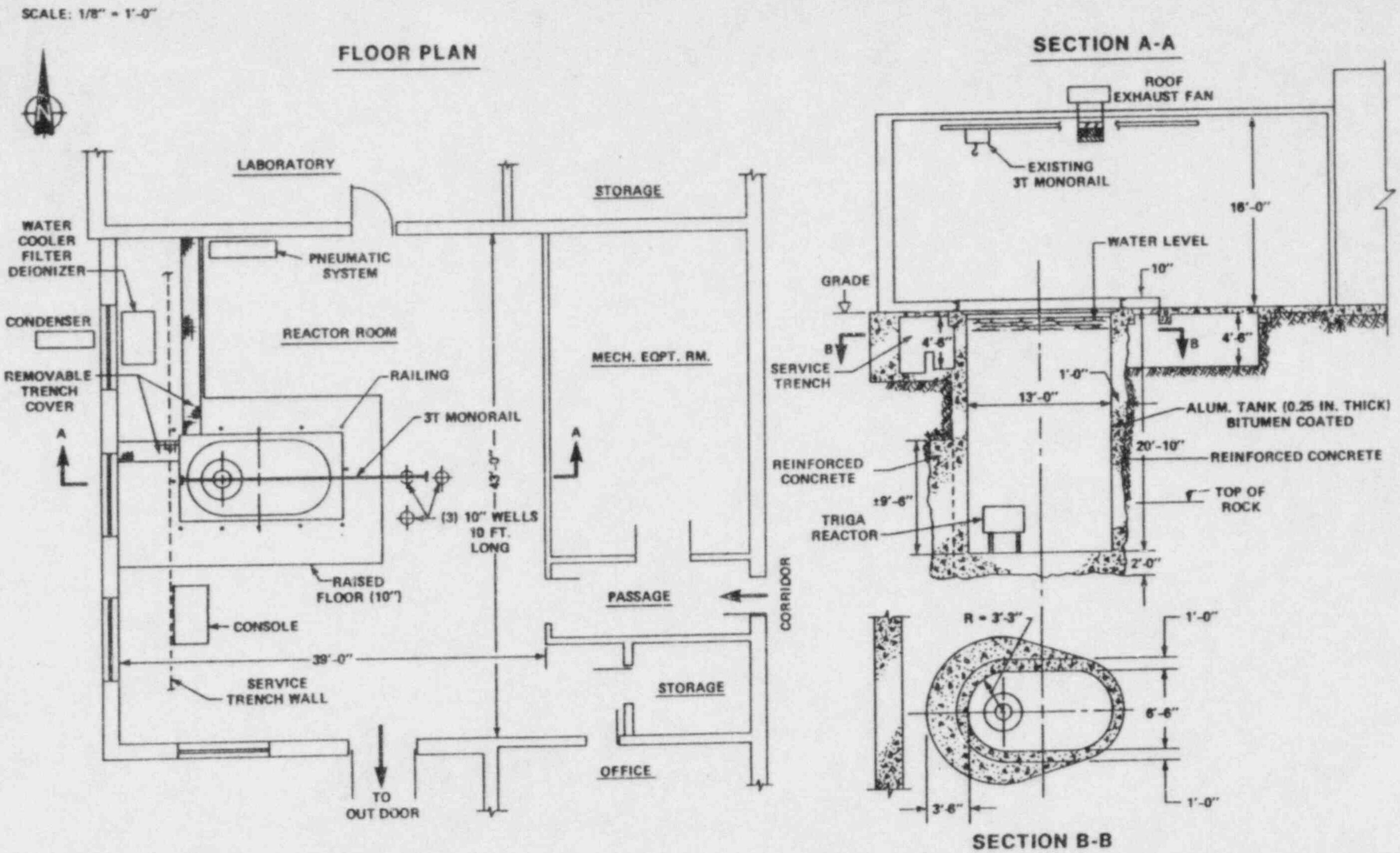
- Measurements of airborne radioactivity will be made to establish "background" levels before any decommissioning activity begins.

The direct result of this task will be the generation of representative radiation zone maps which will reflect the radiological status of the reactor room.

The primary risk of exposure during performance of this task is exposure of personnel conducting the survey to unidentified radiation sources. It is expected that respiratory protection devices will not be required during this phase unless measurements of airborne concentrations dictate so. However, protective clothing will be necessary (especially gloves and boots) to prevent deposition of contaminants on the surveyor's extremities and clothing.

Figure 3.1

REACTOR ROOM - ARRANGEMENT BEFORE DISMANTLING



b. Determination of Concrete Activation Levels and Extent.

Determination of the extent of neutron-activation of concrete will confirm thickness of concrete to be removed and disposed of as radioactive waste. Following the Task 3 (Activity 3d) removal of pool water, activation levels will be determined by taking samples of the concrete at various depths by bore-holing. Laboratory analysis and measurement of the samples will allow determination of the activity concentration of gamma and beta emitters in the concrete.

During boring of the concrete to extract the samples, dust will be generated which will contain radioactive nuclides. This necessitates monitoring of airborne concentrations and the use of protective clothing (e.g., coveralls, gloves, boots). It is expected that nose and mouth masks will be utilized to prevent inhalation of airborne particulates.

c. Determination of Radiation Levels in Taylor Hall.

It is necessary to establish "background" radiation levels in the balance of Taylor Hall in order to be able to assess the radiological impact of the decommissioning activities on these areas.

This task of the radiation survey will address two main concerns:

a. shine in areas adjacent to the reactor room (on the same floor and above), and

b. airborne radioactivity concentration in areas (laboratories, offices, etc) where occupancy is expected to continue during decommissioning operations.

The following tasks will be performed in this phase of the radiation survey:

- gross gamma survey of areas (rooms, labs, offices) in Taylor Hall with particular attention to the rooms adjacent to the reactor room.

- air samples from rooms in Taylor Hall, particularly from those which are served by a common ventilation system.

It is expected that these types of measurements will be repeated periodically during the decommissioning operation to ascertain that nothing has changed.

Since the radiation fields expected to be encountered during this phase of the survey will not be significant, the use of protective clothing and respiratory devices is not foreseen.

4. <u>Determination of Outdoor Radiation Levels</u>

"Background" radiation levels outside the building will be determined to facilitate assessment of the impact of decommissioning activities.

The following tasks will be performed during this phase:

- The area immediately adjacent to Taylor Hall will be surveyed. Gamma measurements will be made along a single straight line (radius) starting from (approximately) the axis of the (removed) reactor and extending to 10 meters beyond the Reactor Room boundaries.

- Beta-gamma and gamma measurements should be made.

- Soil samples will be counted with appropriate instruments to determine the background contamination level in the soil.

It is not expected that special safety and/or radiation protection measures will be necessary during this phase of the survey.

5. <u>Evaluation of Results</u>.

Upon completion of the survey, a comprehensive evaluation of the results will be performed and their impact on the planning and conduct of the decommissioning will be assessed.

Task 2: Remove Components with Surface Contamination.

Components with potential surface contamination include spent fuel racks, cable and conduits, pool deck plates and rotary rack drive (see Table 3.1). The level of fixed and removable surface contamination of these components will be determined in Task 1. Based on these results the surface contamination of non-usable components will be removed until the levels prescribed in Regulatory Guide 1.86 are met to permit disposal as non-radioactive materials. A few components may be appropriately packaged and transported for reuse and these may not require complete decontamination.

Major activities associated with this task are

- Components disassembly;
- Surface contamination removal and
- Packaging for disposal or reuse.

Those components connected by pipes and tubes will be disconnected and swipes taken at connections to determine if the internals have been contaminated. If the contamination levels at these points fall within Regulatory Guide 1.86 limits, the components will be either reused elsewhere or discarded as scrap metal. If the contamination levels are above Regulatory Guide 1.86 limits, the components will be dismantled and further cleaned; valves, fittings and irregular pieces will be appropriately disposed of as low level radwaste as discussed in Chapter 5.0, Radioactive Materials and Waste Management. The pool water purification system will be left intact for subsequent clean-up of the pool water (in Task 3) and later removal (in Task 4).

Locally significant airborne concentration of radioactive material could result from the dismantling operations described above. Consequently, proper containment of such concentrations will be effected by use of

plastic bags and other techniques. The other main radiological concern during this operation is contamination of exposed skin surfaces and contamination of tools.

General ventilation provided by the existing exhaust fan, which is equipped with roughing and HEPA filters, is considered adequate for this task.

Task 3: Remove Components with Induced Radioactivity.

Components with induced radioactivity remaining in the facility at the time of initiation of decommissioning operations include control rod guide tubes and detector tubes, top and bottom grid plates, reflector, core support, rotary specimen rack, fasteners and connectors (see Table 3.2).

These components will be stored underwater in the reactor tank until they are packaged for disposal or reuse.

Major activities associated with this task are:

- Take dose-rate measurements;
- Disassemble components (if needed);
- Package for disposal or reuse;
- Remove and decontaminate water from reactor tank;
- Remove surface contamination from tank liner, and
- Install protective cover over tank with shielding as needed.

The University plans to reuse some components at another site, with the exception of the rotary specimen rack. Reusable components will be

packaged in suitable containers for shipping, in accordance with the NRC and DOT regulations for transportation of radioactive materials. Packaging will be performed underwater when indicated by measured activity levels.

The induced activity of the rotary specimen rack is estimated to be on the order of 100 R/hr. Actual activity levels will be determined from measurements taken as the first step of this task.

Based on present estimates it appears that the rotary specimen rack can be packaged in one piece in a shielded container of 46 in. x 72 in. x 23 in. high (44 $feet^3$). Avoiding the cutting of the rack will significantly reduce worker doses and will prevent generation of contaminated metal dust.

The water from the reactor tank will be decontaminated (using the existing purification system), checked for radioactivity, and then discarded by appropriate methods.

Ventilation during performance of this task will be the same as that described in Task 2 above.

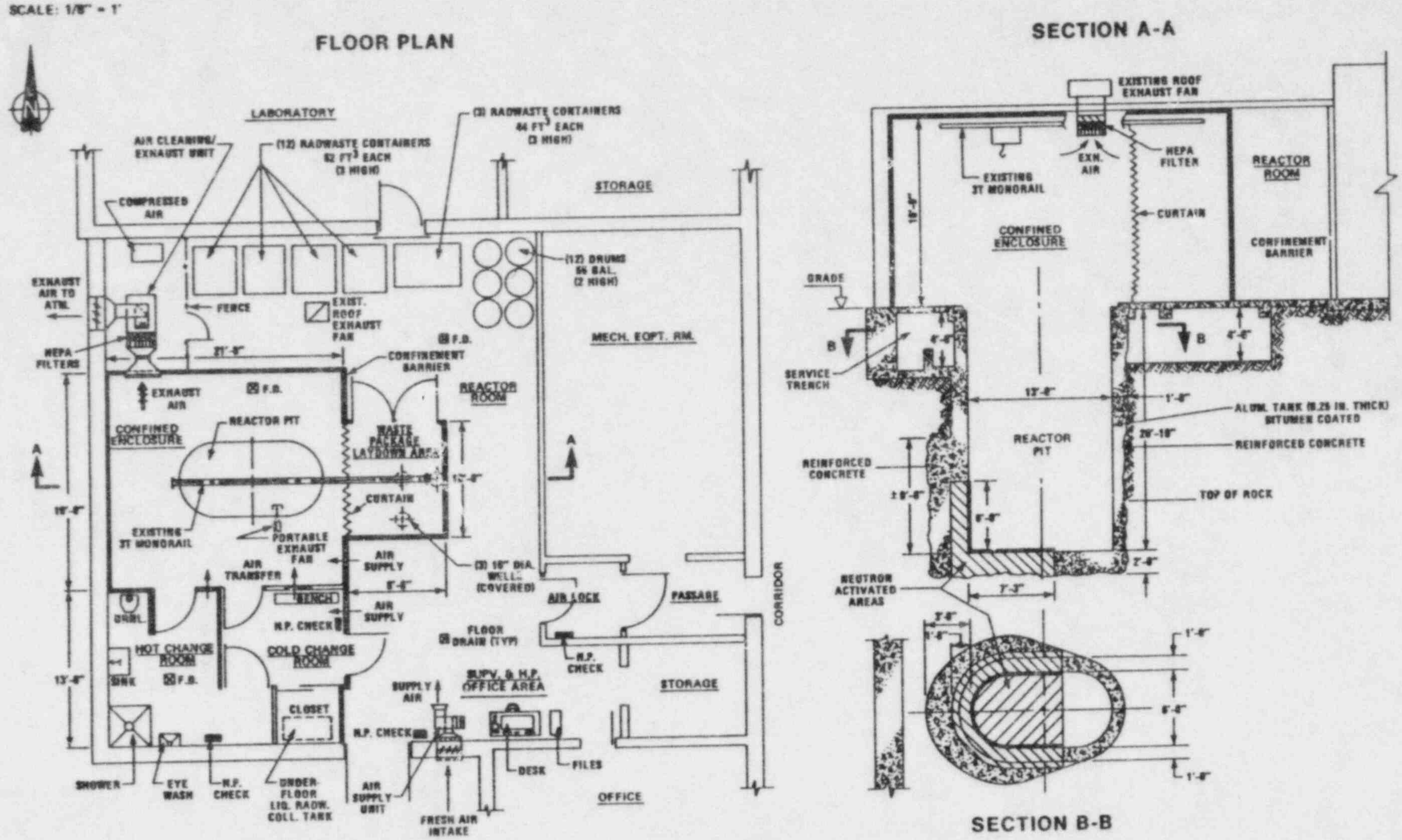
Task 4: Install Confinement Barrier, Ventilation and Associated Systems.

After components, equipment and parts listed in Tables 3.1 and 3.2 have been removed from the reactor room, a confinement barrier will be installed. The purpose of this barrier is to contain airborne contaminants generated during reactor pit demolition, and to prevent their spread in the reactor room and possibly in the surrounding areas. Figure 3.2 illustrates the general arrangement of the Reactor Room during dismantling including the confinement barrier outline.

The confinement barrier will consist of a plastic enclosure on a rigid frame which will surround the reactor pit. Associated with this

REACTOR ROOM - ARRANGEMENT DURING DISMANTLING

Figure 3.2



enclosure will be an independent localized ventilation system which will ensure a negative pressure with respect to the reactor room while providing high efficiency filtration of the exhausted air, and a source of clean air supply within the enclosure.

Major activities associated with this Task are:

- Modify lighting to serve functional areas;
- Connect utilities (water, power, etc);
- Install confinement barrier and seal joints and penetrations;
- Install and test ventilation system;
- Install, calibrate and test radiation monitoring Health Physics equipment, and
- Reinforce floor to support waste packages storage/handling (if needed).

As illustrated in Figure 3.2, the confinement barrier will be provided by the existing west wall and by three free standing walls. A hung ceiling, encompassing the monorail, will provide protection of the roof steel. This configuration is determined by the work area needs, storage space requirements, and access considerations. A secondary enclosure, separated by curtains from the main confinement barrier, will be provided for monorail parking and waste packages laydown area.

The free standing walls will be of plywood on a wood frame covered with heavy plastic sheets. Wood studs will be installed along the existing wall vertical and a heavy gauge plastic sheet attached on the face. The ceiling of the enclosure will be a wooden frame hung from the roof trusses and covered with heavy gauge plastic sheets. All seams and penetrations will be sealed.

The floor of the reactor room, for the most part, is epoxy coated; no additional treatment is contemplated except for coating the small area presently bare in the raised portion of the floor.

Personnel access to and from the Confined Enclosure will be through the Change Rooms. For entering the Confined Enclosure, personnel will dress in appropriate work clothes and leave their street clothes in the Cold Change Room. For exiting the Confined Enclosure, personnel will leave their work clothes, wash and shower in the Hot Change Room, and then enter the Cold Change Room to dress in their street clothes. Because of the short duration, non-permanent, nature of this work site, only minimum toilet facilities are provided within the Change Room; a portable toilet will be made available (outdoors, in the immediate vicinity of the south wall door) to be used by personnel after they change into street clothes.

Ventilation within the confinement barrier will be provided by a mobile air cleaning/exhaust unit which will exhaust the air from the enclosure at the floor level, monitor it, and discharge it outdoors at high elevation. The unit will be equipped with a roughing filter and a high efficiency particulate filter (HEPA). In addition, the exhaust grill at the confinement barrier wall penetration will be provided with a roughing filter.

The confinement barrier will be maintained at a negative pressure with respect to the balance of the reactor room. Ventilation for the balance of the reactor room will be provided by the existing roof exhaust fan which is equipped with roughing and HEPA filters. Fresh air intake to the Reactor Room (in office area) will be provided by an air supply unit. Supply air to the confined enclosure and to the change room will be through intake grills from the Reactor Room. A portable exhaust air fan equipped with flexible hose extending to the bottom of the pit will be located on the Reactor Room floor, to provide local exhaust and to draw fresh air into the pit.

The reactor room will be maintained at a negative pressure with respect to the surrounding areas but lesser than the pressure differential maintained between the confinement barrier and the reactor room. This will insure that the air will travel from the non-contaminated area to the increasingly contaminated areas.

During this task, modification of the lighting system to serve the new room areas will be made; utilities will be connected, and the health physics and radiation monitoring equipment installed, calibrated and tested. Air locks/check points areas will be provided at entries to the Reactor Room. The structural loading capability of the Reactor Room floor will be investigated and reinforcing provided as necessary to permit the temporary storage of radwaste packages.

A dewatering pump will be available in case ground water is encountered during floor demolition, as described in Chapter 5.0.

Task 5: Demolition of Non-Activated Portion of the Reactor Tank.

The only non-activated portion of the reactor tank and concrete to be removed prior to demolition of activated areas is the floor (liner and concrete). Considering that over 50% of the volume of the floor slab is expected to have been significantly activated and must be demolished, removal of the remaining non-activated volume of the floor is desirable for overall tank stability (in addition to providing a level work floor).

Major activities associated with this Task are:

- Take borehole samples to confirm extent of concrete activation;
- Place shield slabs over the activated area of the floor;
- Cut and remove aluminum liner of the non-activated portion of the tank floor, and

- Demolish and remove concrete in the entire thickness (2 feet) of the non-activated portion of the reactor tank floor and cut reinforcing steel bars.

Prior to beginning of demolition, boreholes samples will be taken at strategic locations to confirm the analytical results regarding the extent of the activated areas. The demolition, which includes both the removal of the aluminum liner and surrounding concrete, will be performed by conventional methods, yet with strict administrative controls to assure no infringement into the activated areas of the tank.

The rubble resulting from concrete demolition will be checked for radioactivity, removed from the Reactor Room and discarded by appropriate means.

There are two potential safety concerns during performance of this task: (i) external exposure from the activated components of the tank, and, (ii) inhalation of airborne radioactive material. To minimize the risk, work areas will be monitored frequently and radiation levels will be monitored continuously, to determine sudden changes in the radiological conditions. Temporary shields and respiratory protection devices will be readily available.

Task 6: Demolish Activated Portion of the Reactor Pit.

Figure 3.2 indicates portions of the reactor pit which have been neutron activated. The task of removing these portions are as follows:

- Remove shield slabs and cover with plastic mats the remaining non-activated area of the floor;

- Cut aluminum liner from activated portion of the reactor tank floor,

- Package and remove the cut-up liner from the tank pit;

- Demolish the entire thickness of concrete and cut reinforcing steel bars from the activated portion of the reactor pit floor;

- Package and remove concrete rubble from tank pit bottom and cover remaining pit floor area with wood platform and heavy mats;

- Cut activated aluminum from tank wall;

- Package and remove cut-up liner from tank pit,

- Demolish activated portion of the wall (estimated to be 20 inches in thickness); place supports as needed;

- Take dose rate measurements from remaining concrete, and

- Package and remove concrete rubble from tank pit.

Our calculations indicate that the expected maximum total dose rate inside the reactor pit is 7.6 mR/hr at 1 meter from the reactor pit floor, and 0.9 mR/hr at 1 meter from the wall core midplane at the "hot end" of the pit (See Figure 1.11). The calculations also indicate that, in addition to the aluminum liner activation in the wall area, the concrete activation is significant to a depth of approximately twenty inches. Activation of concrete has occurred in the entire two feet depth of the floor slab. Activation of the reinforcing steel has occurred only in the floor slab. It must be expressly noted that confirmation of the analytical results and of the preliminary measurements will be necessary during demolition operations, and adjustments made accordingly in the depth of concrete to be removed to account for actual field conditions.

The activated aluminum liner will be cut by using a rotary saw and the activated thickness of the concrete will be broken off with a jack hammer. To minimize dust dispersal, a localized fine water mist will be sprayed over the area being demolished.

Activated concrete will be removed a section at a time and supports will be placed in the cavity formed as needed, before proceeding with the next section.

At the completion of activated concrete removal, dose rate measurements will be made to determine if all necessary portions have been removed.

As the demolition of activated material proceeds, the radioactive material will be packaged for shipment and disposal as discussed in Chapter 5.0.

The safety concerns in this task are similar to but more severe than those in Task 5. Aside from the radiation monitoring equipment mentioned in Task 5, it is expected that respiratory protection devices will be utilized continuously during the demolition. Means will be employed to contain the dust locally and thus minimize the spread of contamination. It is anticipated that this task will be the most significant contribution to the total collective dose (man-rem).

Task 7: Clean-up Reactor Room.

Upon completion of all dismantling operations the Reactor Room will be cleaned-up by performing the following major activities:

- Remove all loose contamination from Reactor Room including confined enclosure area;
- Decontaminate waste packages, make dose rate check and move outside for shipping;
- Clean for reuse or disposal contaminated tools and equipment; package for disposal;

- Remove surface contamination in the reactor tank pit, and place a tight cover over it;

- Dismantle confinement barrier and package for disposal, and

- Remove surface contamination from the entire Reactor Room.

As indicated in Table 3.4, equipment and supplies used during decommissioning will be checked for potential contamination and either properly decontaminated for reuse or disposed of as solid radwaste.

The confinement barrier will be dismantled and the plastic sheets compacted and packaged. Surface contamination will be removed from contaminated portions of the ventilation system and they will then be packaged for disposal. Contaminated clothing will also be disposed of by compacting and appropriately packaging them.

<u>Task 8: Package and Ship Contaminated Materials and Radioactive Wastes.</u>

Throughout the decommissioning activities, radioactive wastes generated will be collected and placed in approved shipping/disposal containers.

The following major activities will take place during this task:

- Placing the empty packages in the reactor pit;

- Packaging of activated aluminum liner;

- Packaging of activated concrete;

- Removal of filled packages from reactor pit;

- Check dose rate of each package and clean as necessary;

- Loading packages onto trailers;

- Making final dose rate check of packages and of loaded trailers, and

- Shipping packages to disposal site or to new place of use.

It is expected that the number of packages generated will not require more space than that available in the reactor room. Therefore the packages will be collected for shipment to the disposal site at the end of the decommissioning activities.

A more detailed discussion of the packaging and shipping of radioactive wastes is given in Chapter 5.0.

Particular attention will be given to loading and handling of packages while in the reactor pit, and to their removal from the pit. The existing monorail will be used to lower and lift the packages from and to the Reactor Room floor. To protect packages from undue contamination in the pit environment during loading, it is envisioned that the packages will be covered with a disposable plastic jacket; this jacket will be removed upon preparing packages for shipping.

Task 9: Perform Final Radiation Survey.

A final radiation survey will be conducted at the facility upon completion of all activities described in previous tasks. The goal of the survey is to ensure that ambient radiation levels and surface contamination levels are well below the limits specified in Regulatory Guide 1.86 and to verify that the site can be released to unrestricted use.

The following major activities will be performed in this task:

- Perform radiation survey of the Reactor Room;

- Perform radiation survey of Taylor Hall;

- Perform radiation survey of potentially contaminated outside areas;

- Perform soil and water sampling, and

- Prepare final radiation survey report.

Further description of the Final Radiation Survey is provided in Chapter 7.0.

3.4 Schedule of Major Activities

The overall schedule for accomplishing the decommissioning of the TRIGA reactor at Taylor Hall is shown in Figure 3.3. As indicated, the total duration of the decommissioning tasks is estimated to be 5-1/2 months, with the task of demolishing and removing the rubble from the activated portion of the reactor pit being the longest (60 days).

ESTIMATED DECOMISSIONING SCHEDULE

TASK/ACTIVITY	MONTHS 1	2	3	4	5	6
1. Perform Comprehensive Radiation Survey	–7–					
1a. Determine radiation levels in the reactor room	(2)					
1b. Take borehole samples to confirm extent of activated areas	(2.5)					
1c. Establish outside background radiation levels in Taylor Hall	(3)					
1d. Establish outside background radiation levels	(2)					
2. Remove Components with Surface Contamination Only	–5–					
2a. Disassemble components	(1)					
2b. Remove surface contamination	(2)					
2c. Package for disposal or reuse	(2)					
3. Remove Components with Induced Radioactivity (Except Reactor)	–8–					
3a. Take dose-rate measurements	(1)					
3b. Disassemble components (if needed)	(1)					
3c. Package for disposal or reuse	(1)					
3d. Remove and decontaminate water from reactor tank	(2)					
3e. Remove surface contamination from reactor pit	(2)					
3f. Install protective cover over reactor pit	(1)					
4. Install Confinement Barrier, Ventilation & Associated Systems	–10–					
4a. Modify lighting to serve functional areas	(1)					
4b. Connect utilities (water, power, etc)	(2)					
4c. Install confinement barrier and seal joints & penetrations	(4)					
4d. Install and test ventilation systems	(3)					
4e. Calibrate, test and install health physics equipment	(2)					
4f. Reinforce floor to support waste package/hauling (as needed)	(4)					

FIGURE 3.3

TASK/ACTIVITY	MONTHS 1	2	3	4	5	6
5. Demolish Non-Activated Portion & Reactor Pit		-5-				
5a. Place shield slabs over the activated area of the floor		(0.5)				
5b. Cut and remove aluminum liner of the non-activated portion of the tank floor		(2)				
5c. Demolish and remove concrete in the entire thickness (2 ft) of the portion of the reactor tank pit floor		(2.5)				
6. Demolish Activated Portion of the Reactor Pit			-60-			
6a. Remove shield slabs and cover with plastic mats the non-activated areas of the floor		(1)				
6b. Cut aluminum liner from activated portion of the reactor tank floor		(3)				
6c. Package and remove cut-up liner from tank pit		(1)				
6d. Demolish the entire thickness of concrete from the activated portion of reactor pit floor		(20) (INCL. 6e ACTIVITY)				
6e. Package and remove concrete rubble from tank pit		(2)				
6f. Install work platforms on the entire Reactor tank pit bottom and cover with heavy mats			(1)			
6g. Cut activated aluminum from tank wall			(4) (INCL. 6h ACTIVITY)			
6h. Package and remove cut-up Liner from tank pit						
6i. Demolish 20 inches of thickness from activated concrete portion of the wall			(21) (INCL. ACTIV. 6j)			
6j. Package and remove concrete rubble from tank pit			(1)			
6k. Take dose rate measurements from remaining concrete				(5)		
7. Clean-up Reactor Room				-28-		
7a. Remove surface contamination in the reactor tank pit, place cover and seal				(1)		
7b Decontaminate waste packages make dose rate check and move outside for shipping				(4)		
7c Clean for revise, or for disposal contaminated tools and equipment; package for disposal				(7)		

FIGURE 3.3

Sheet 3 of 3

TASK/ACTIVITY	MONTHS 1	2	3	4	5	6
7d Remove all loose contamination from reactor room including confinement barrier				(2)		
7e. Dismantle confinement barrier and package for disposal				(2)		
7f. Remove surface contamination from the entire reactor room				(14)	(14)	
8. Package and Ship Contaminated Materials and Radioactive Wastes [Packaging Tasks shown in 2(c); 3(c), 6(c,e,h,j); 7 (c,e) above]						
8a. Load packages onto trailers					(2)	
8b. Make final dose rate check of loaded trailers					(1)	
8c. Ship packages to disposal site or to new place of use					(1.25)	
9. Perform Final Radiation Survey					-19-	-19-
9a. Perform radiation survey of the reactor room					(2)	
9b. Perform radiation survey of the Taylor Hall					(2)	
9c. Perform radiation survey of out-of-door						(2)
9d. Perform soil and water sampling						(3)
9e. Prepare final radiation survey report						(10)

NOTE

Schedule is based on 7 days per week work.

LEGEND

▬▬▬ Continuous Activity

▬ ▬ ▬ Intermittent Activity

-x- Total Task Duration

(x) Days for Activity

0228z

CHAPTER 4

4.0 SAFEGUARD AND PHYSICAL SECURITY

The removal of all special nuclear materials from the facility will occur under the current operating license before the initiation of decommissioning and dismantling. Changes to the facility physical security plan will be made accordingly.

The plan for security of the Reactor Room during the demolition and removal of radioactive materials will consist of control of barriers and entrance points. Access control when the facility is unoccupied will have sufficient resistance to penetration that a casual unauthorized entry cannot occur. Intrusion or alarm systems may be used but are not necessary for control of the area after removal of the special nuclear materials.

CHAPTER 5

5.0 RADIOACTIVE MATERIALS AND WASTE MANGEMENT

During the decommissioning activities radioactive materials (radwaste) in liquid, solid and gaseous forms are expected to be generated. Management of these wastes is an integral part of the decommissioning plan and includes provisions for minimizing the amount of waste generated and waste collection, treatment, packaging and shipment off-site for disposal.

5.1 Fuel Disposal

As indicated in earlier sections of this plan, the University intends to remove the fuel from its TRIGA Mark I research reactor under the current operating license. Therefore, at the beginning of decommissioning operations there will not be fuel on-site.

5.2 Liquid Radwaste

Liquid radwastes generated during decommissioning activities will be collected, monitored and solidified prior to shipment to an approved disposal site.

Collection of liquid radwastes will be in a tank installed in the existing underfloor space of the Reactor Room. Temporary floor drains will be installed in the reactor room floor, as shown in Figure 3.2, with collection pipes discharging by gravity into the collection tank. From the collection tank the liquid will be pumped to a radwaste treatment mobile unit provided by the demolition contractor. The collection tank will be monitored for radioactivity and liquid level; the pump, an integral part of the mobile unit, will be started on high liquid level and/or high radioactivity level.

Expected sources of liquid radwaste are:

- Decontamination of components, parts, etc

- Decontamination of structures, floors, etc

- Change room fixtures (sink, shower, eye wash)

Efforts will be made throughout the decommissioning activities to minimize the generation of liquid waste. Whenever possible scrubbing with swabs will be used instead of spraying. The water mist used during the demolition of activated concrete will be closely controlled and rags will be used around the floor to absorb any run-offs.

A stand-by dewatering pump will be available for the eventuality that ground water may be encountered during reactor pit demolition. This pump will discharge any potential water into an auxiliary collection tank located on the floor of the Reactor Room. The water in this tank will be monitored for radioactivity and if contamination in excess of the 10 CFR 20 limits is found, than the water will be transferred by gravity to the liquid radwaste tank; otherwise, the water will be released to the sewage system.

Figure 5.1 illustrates the liquid radwaste system concept.

5.3 Solid Radwaste

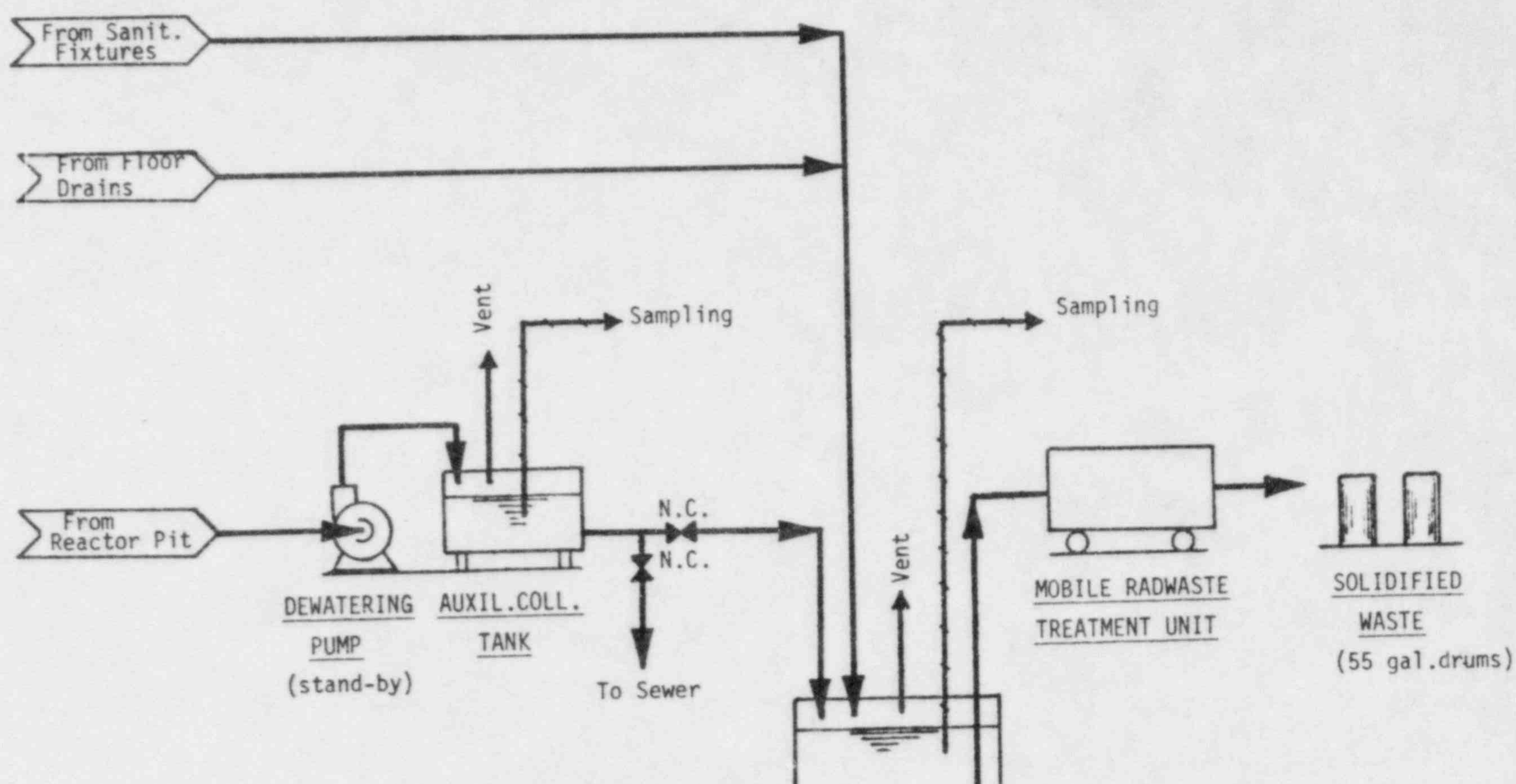
The solid radwastes generated during decommissioning activities will be packaged on site in containers suitable for shipping and disposal.

5.3.1 Packaging

The types of solid radwaste to be packaged include the following:

Demolition Materials. These include the aluminum liner, concrete rubble, reinforcing steel and steel anchors.

The total area of activated aluminum to be removed from reactor pit floor and wall is expected to be 152 square feet. If this aluminum



LEGEND

N.C. Valve Normally Closed

LIQUID RADWASTE BLOCK DIAGRAM

FIGURE 5.1

is cut in sections suitable for a 44 cubic feet metal container, approximately 110 linear feet of cutting will be required. Only about 1/5 of the free container volume is assumed to be filled, when allowing for the expected bending of the aluminum during its removal from the concrete; the balance of the container will be filled with miscellaneous solid radwastes.

The total volume of activated concrete to be removed from the reactor pit floor and wall is expected to be 284 cubic feet. Assuming a packing density of 50% and 52 cubic feet metal containers, 11 containers will be necessary. The weight of a filled container will be approximately 4500 lbs (empty container weight is 775 lbs), well within the capability of the existing 3 ton monorail to lift the container from the reactor pit.

Other potentially activated demolition materials are some of the reactor tank steel anchors and the reinforcing steel from the activated portion of the reactor pit floor. The small volume of these materials will permit their packaging in a fraction of a container, most probably mixed with other solid radwastes.

Equipment and Tools. This includes such items as saws, jack hammers, monorail, forklift, shovels, pumps, tanks, ventilation system components, filters, piping, etc. Only some, and parts of some of this equipment are expected to be discarded as radwaste. A determination of volumes of solid radwaste generated from this category will be possible only during the clean-up task (7) when measurement of contamination level and evaluation for decontamination will be made.

It is planned that in order to minimize the solid radwaste volumes, compaction will be used whenever feasible (e.g., on ductwork, tanks, etc). A mobile compactor is assumed to be provided by the demolition contractor. For estimating purposes, six 55 gallon

drums and one 44 cubic feet metal container are assumed for packaging of these equipment parts and tools.

Auxiliary Materials and Clothing. This includes the confinement barrier plastic sheets, protective mats, rags, work platforms, and protective clothing. It is assumed that these materials will be compacted and packaged in six 55 gallon drums.

Activated Equipment. This includes primarily the rotary specimen rack which is planned to be packaged in one piece in a 44 cubic feet shielded metal container. Other small components would probably be packaged in the same container as the rotary specimen rack will occupy only part of the volume.

Table 5.1 summarizes the solid radwaste volumes and number of packages required.

5.3.2 Containers Handling

Because of the small work areas available, the demolition materials resting at the reactor pit floor will require removal as the demolition proceeds. An area of the pit floor will be kept clear during demolition to allow laydown of an empty container. After the equivalent of a container content is demolished, (approximately 26 cubic feet of concrete) a container will be lowered with the aid of the monorail. It is anticipated that a protective plastic jacket will be placed on the outside of the container to minimize deposition of contaminated dust; the upper edges will be folded inside the container. After the container is layed down, the loading of concrete rubble will proceed. The filled container will then be lifted from the pit with the aid of the monorail and moved inside the laydown area within the confined enclosure of the Reactor Room floor. At this point the container cover will be preliminarily placed and the plastic jacket cut-off around the cover (fold-over plastic left in the container). The containers

TABLE 5.1

SOLID RADWASTE QUANTITIES AND PACKAGING

Material/Component	Area		Volume		Number of Packages		
	ft^2	m^2	ft^3	m^3	55 gal drums	44 ft^3 container	52 ft^3 container
Aluminum Liner	152	14	3.0	0.08	--	0.20 (1)	--
Concrete Rubble	152	14	284.0	8.1	--	--	11
Rebar & Steel Anchors			negl.		--	--	--
Contaminated Equipment	--	--	--	--	6	1	--
-ventilation system -radwaste system -material handling equipment -tools							
Auxiliary Materials and Clothing	--	--	--	--	6	--	--
-protective mats -confinement barrier -clothing							
Rotary Specimen Rack	--	--	--	--	--	1	--
Contaminated Soil (2)	--	--	--	--	--	--	1 (2)
TOTAL :					12	3	12

NOTES:

(1) 80% of package filled with miscellaneous wastes

(2) Unknown if any - included here for contingency

will then be moved with a fork lift to the storage area within the Reactor Room and the cover sealed.

5.4 Gaseous Radwaste

The gaseous radwaste system consists of two separate ventilation systems operating in the Reactor Room: a general ventilation filtered exhaust provided by the existing roof fan, and a new localized ventilation filtered exhaust serving the confined enclosure around the reactor pit.

The two ventilation systems will operate to ensure a negative pressure in the Reactor Room with respect to surrounding areas of Taylor Hall and a (greater) negative pressure within the confinement barrier with respect to the Reactor Room. The purpose of this arrangement is to assure that the air flows from the non-contaminated area towards increasingly contaminated areas.

Both ventilation exhaust systems will be equipped with roughing filters, to capture large size particles, and with high efficiency particulate adsorption filters (HEPA) to provide up to 99.99% particulates retention. HEPA filters will be changed upon high radiation levels readings/alarm in the exhaust duct, or based on maximum pressure differential readings indicating that the filters are filled with dust. Radiation monitoring will be provided for the exhaust air to atmosphere; readings above prescribed limits will shut off the exhaust fans. The ventilation unit serving the confined enclosure will be mobile type and connected with flexible ducts to the enclosure and to the exhaust duct. The exhaust duct will discharge the air to the outdoors at a high elevation above ground. The exhaust duct will be provided with gravity louvers which will automatically close in the event of failure of the ventilation exhaust unit.

Fresh air intake to the Reactor Room (in the Office Area) will be provided by a supply fan. Supply air to the Confined Enclosure and to the Change Room will be through intake grills from the Reactor Room.

Figure 5.2 illustrates the gasesous radwaste/ventilation system concept.

5.5 Waste Classification

The criteria for waste classification for low-level waste disposal is contained in 10 CFR 61. The allowable concentration of short-lived and long-lived radionuclides of significance to the UT TRIGA research reactor decommissioning (Co-60, Ni-59 and Ni-63) were compared with the expected activity concentrations shown in Table 1.5 of this report. The comparison led to the conclusion that the radioactive wastes from the TRIGA reactor can be classified as Class A. Class A wastes need not be segregated for disposal, providing they meet the stability criteria described in paragraph 61.56 of the regulation (i.e., wastes do not structurally degrade and affect overall stability of the disposal site through slumping, collapse, etc).

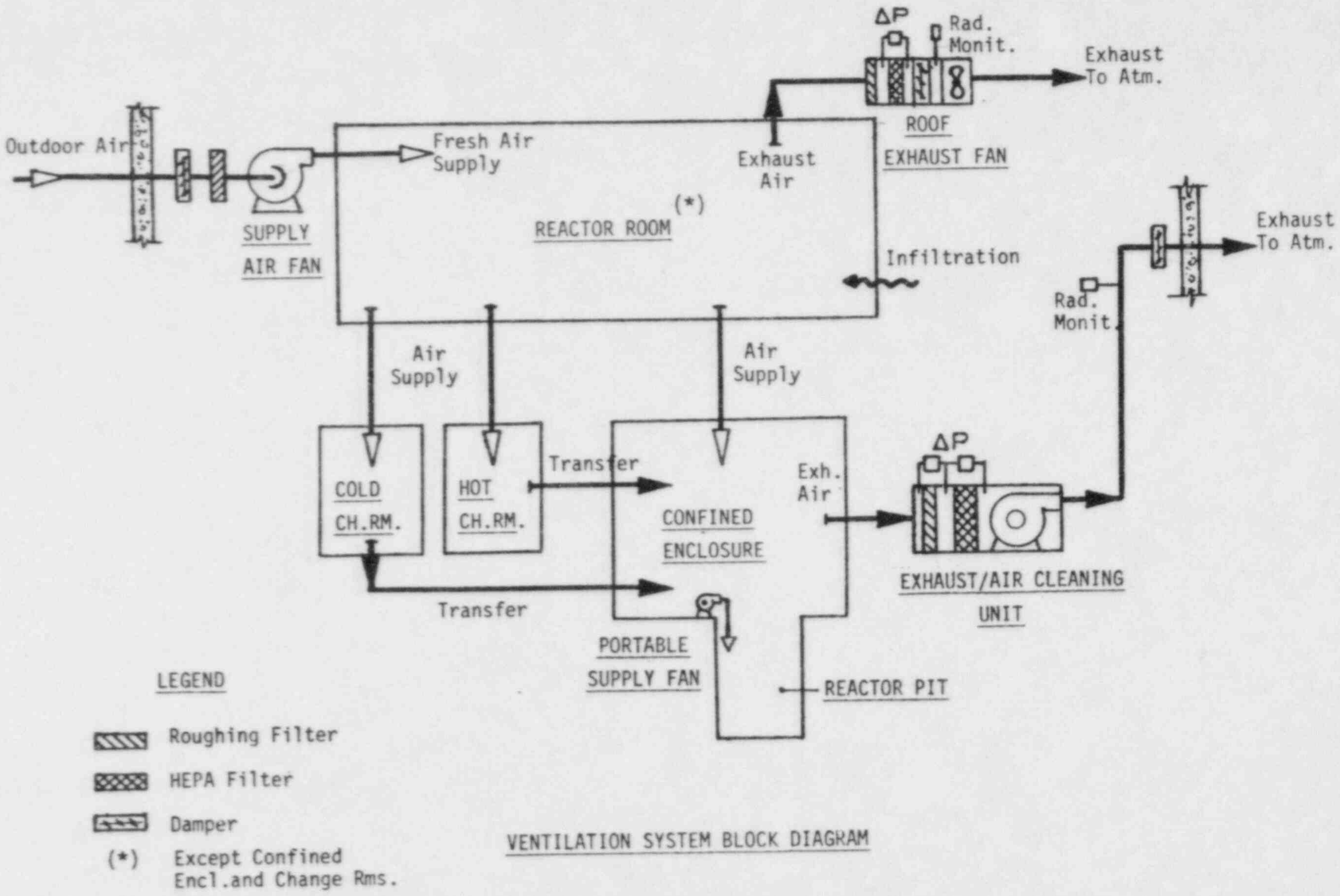
The radioactive waste from UT TRIGA reactor decommissioning will meet these criteria because they will be in solid form (aluminum liner, concrete and other metal components). To further comply with the regulations, the container will be filled such that voids will be kept to a minimum for insuring container's structural stability when overburden or other packages are placed over them.

5.6 Shipping of Radioactive Wastes

Department of Transportation Regulation 49CFR173.441 provides radiation level limitations for transportation of packages of radioactive materials in closed, exclusive use transport vehicles as follows:

- 1000 millirem per hour on the accessible external surface of the package;

- 200 millirem per hour at any point on the outer surface of the transport vehicle;



VENTILATION SYSTEM BLOCK DIAGRAM

FIGURE 5.2

- 10 millirem per hour at 2 meters from the vertical outer lateral surfaces of the transport vehicle;

- 2 millirem per hour in any normally occupied position in the transport vehicle.

Based on the radiation dose rate calculations from the activated concrete and aluminum liner materials presented in Section 1.3.4, no problems are anticipated in maintaining dose rates below these limits, even for waste pacakges containing the most highly activated concrete from the floor beneath the reactor core.

For activated core components to be disposed of, such as the rotary specimen rack, it is expected that the limits will be prevented from being exceeded by the provision of additional shielding within the waste package depending on dose rate measurements made at the time of its packaging.

CHAPTER 6

6.0 TECHNICAL AND ENVIRONMENTAL SPECIFICATIONS

Since the DP encompasses the decommissioning phase and the subsequent release of the facility to unrestricted use, Technical and Environmental Specifications will be developed which will cover both phases.

The Technical and Environmental Specifications (TES) will control conditions, parameters and variables so that:

a. during decommissioning activities the radiation exposure to workers and the public shall be maintained ALARA;

b. after decommissioning the site can be released to unrestricted use.

The TES will include items in the following categories:

1. Safety limits

2. Surveillance requirements

3. Administrative controls

4. Design features

6.1 Safety Limits

Safety limits for the DP are those bounds within which certain parameters important to safety must be maintained for adequate control of the decommissioning activities and for subsequent unrestricted use. In certain cases, University of Texas administrative limits (such as for radiation exposure of individuals during decommissioning) may be lower and will be applied in addition to the following.

6.1.1 Safety Limits During Decommissioning

During decommissioning, safety limits shall be imposed on the following parameters.

<u>External Exposure</u>: external exposure for individuals in restricted areas during decommissioning shall not exceed the limits specified in 10CFR 20.101.

<u>Internal Exposure</u>: internal exposure from inhalation of radioactive material in air in restricted areas shall not exceed that which would result from the inhalation of the limiting quantities specified in 10 CFR 20.103.

<u>Concentration of Airborne Radioactive Material in Restricted Areas</u>: concentration of airborne radioactive material shall not exceed the limits set in 10 CFR 20.103.

<u>Concentration of Airborne Radioactive Material in Unrestricted Areas</u>: concentration of airborne radioactive material shall not exceed the limits specified in 10 CFR 20.106.

<u>Concentration of Airborne Non-radioactive Contaminants</u>: concentration of such contaminants shall not exceed the limits specified in the pertinent industrial hygiene regulations.

Limiting safety limit settings for the above variables which allow sufficient time for corrective action shall be determined based on realistic models and analytical calculations, and on the results of the radiological surveys.

The ultimate goal of the safety limits shall be to maintain individual and collective doses ALARA. The goal of the limiting safety settings shall be to determine the need for use of specific radiological protection devices and measures.

6.1.2 Safety Limits for Unrestricted Use

Since the facility shall be released to unrestricted use, safety limits must be set so that the total maximum annual dose to man from all exposure pathways does not exceed 10 mrem per year above background[1].

Safety limits are also imposed on the radiological condition of the decommissioned facility as follows:

a. The dose rate at 1 meter from an exposed surface (i.e., activated surfaces of the Reactor Pit) shall not exceed 5 micro-Roentgen per hour above background[1].

b. Surface contamination levels shall not exceed the limits set in Regulatory Guide 1.86, Table 1.

In general, the available exposure pathways are: irradiation from surface and subsurface deposits of radionuclides; inhalation of resuspended radionuclides; and ingestion of food products contaminated by plant uptake.

Given the low power level and the operating history of The University of Texas TRIGA reactor, it is expected that the external irradiation pathway will be the dominant one. Realistic models of the pathway will be developed to calculate the total maximum annual dose, which will account for occupancy times, outdoors times and shielding by building materials.

1. Guidance and Discussion of Requirements for an Application to Terminate a Non-Power Reactor Facility Operating License. Rev. 1, September 15, 1984, USNRC.

6.2 Surveillance Requirements

6.2.1 Surveillance Requirements During Decommissioning

Surveillance requirements during decommissioning activities comprise the set of test, calibration and inspection activities necessary to ensure that systems, components and instruments important to safety are operating in such manner that the monitored parameters and/or variables are maintained within the safety limits specified in 6.1.1.

Typical surveillance requirements to be instituted and implemented during decommissioning are described below.

a. All instruments/systems used for monitoring purposes shall be properly calibrated prior to initiation of the decommissioning of the facility. "prior to" is intepreted to mean a period of one month, maximum. All instruments on which maintenance has been performed shall be recalibrated before returned to use.

   Operation checks of all monitoring instruments shall be performed prior to their employment in any activity and at appropriate intervals during the activity.

   Personnel radiation protection devices (e.g., respirators) shall be tested according to industry standards prior to use.

   Visual inspection of the operating status of components (e.g., pumps) associated with safety systems shall be made at regular intervals during operation. Visual inspection of indicator displays of instruments associated with safety shall be made at regular intervals during their operation. As a minimum two such inspections shall be made during a 8-hour shift.

6.2.2 Surveillance Requirements After Decommissioning

Since the facility will be released to unrestricted use after decommissioning no surveillance is required.

6.3 Administrative Controls

6.3.1 Administrative Controls During Decommissioning

Administrative Controls during decommissioning are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure completion of the decommissioning of the University of Texas TRIGA reactor in a safe manner.

a. Responsibility

The Decommissioning Superintendent shall be responsible for the completion of the decommissioning of the facility at the University of Texas.

b. Organization

The organizational structure for management and performance of the decommissioning activities is shown in Figure 1.15. The functions and responsibilities, and minimum required qualifications and experience of each position are detailed in Section 1.5 of the Decommissioning Plan.

c. Records and Reports

Accurate and complete records shall be maintained by the University of Texas of the performance and completion of all activities which may result in exposure of workers or the public to radiation.

Reports pertaining to decommissioning activities shall be written and submitted to the proper authorities pursuant to Regulatory Guide 1.86.

d. Review

Responsibility for review of procedures, practices and performance shall rest with the appropriate individuals and/or committees detailed in Section 1.5 of the DP.

6.4 Design Features

Design features are those features employed during the decommissioning phase and/or after release to unrestricted use (such as special shielding designs and access barriers) which if modified would have a significant effect on the safety of workers or the public at any time.

Based on the analytical results for the radiological status of the facility, the proposed dismantling/demolition plan shown in Chapter 3.0, and the site characteristics, it is expected that any such design features, if necessary, will be limited to the following during demolition activities in the decommissioning phase:

- Maintenance of a negative pressure in the confined area of the Reactor Room with respect to the rest of the Reactor Room, and also of the rest of the Reactor Room with respect to the surrounding environments. This will be accomplished by a ventilation exhaust system equipped with monitored high efficiency filtration system.

- Provisions for dust suppression/collection systems during decommissioning activities prone to generate radioactive airborne particulates.

CHAPTER 7

7.0 PROPOSED TERMINATION RADIATION SURVEY PLAN

Since the goal of decommissioning is to ultimately release the site to unrestricted use, a termination radiation survey will be conducted in order to ensure that radiologically the site satisfies the requirements contained in Table 1 of Regulatory Guide 1.86; and also satisfies the NRC acceptance criteria for unrestricted use of 5 uR/hr above background at 1 meter from a surface which has been exposed as a result of decommissioning activities.

The termination radiation survey will be conducted after all other decommissioning activities described in Chapter 3.0 have been completed.

The detailed plan for the final radiation survey will depend on:

a. the details of the dismantling/decontamination process;

b. the results of radiation surveys during that process, and

c. the radiological history and other specific characteristics of the facility.

It is expected that other "non-radiological" activities occurring concurrently at the site (e.g. construction, excavation, etc) may contribute to the background radiation levels inside and outside Taylor Hall. Methods will be employed to allow quantification of these "temporary" contributions and thus, the establishment of true radiation levels.

The instruments and tools to be used in the termination radiation surveys will be selected on the basis of the type and level of radiation anticipated as a result of the pre-decommissioning and in-decommissioning radiological surveys. These instruments will have been properly and timely calibrated.

Procedures will be developed for the collection and analysis of data to ensure that:

a. exposure to personnel involved will be maintained ALARA;

b. data collected is representative, accurate and verifiable.

7.1 Description of Survey Tasks

In order to assess the radiological status of the facility and to ensure that it is suitable for release to unrestricted use, the following tasks are planned. The description of the tasks is based on the understanding that the termination survey will focus primarily on the areas which have been impacted by the decommissioning activities.

1. Building Surveys

- o The floor, walls and ceiling of the Reactor Room will be surveyed, in 2 x 2 m blocks.

- o Each block will be surveyed by:

  i. taking measurements of alpha, beta-gamma and surface gamma, at 1 meter. (Repeated measurements may be necessary to improve statistical reliability).

  ii. taking a smear and analyzing it in a counting laboratory.

- o Each block will be scanned and the maximum dose point identified and characterized.

- o A number of air sample measurements will be made to assess airborne concentrations.

Potentially contaminated areas adjacent to the Reactor Room will also be surveyed. These adjacent areas (rooms, laboratories, offices) will be surveyed at the same level as described in the initial Comprehensive Radiation Survey (Section 3.3, Task 1) since they are not expected to change their radiological status as a result of decommissioning.

2. Outdoor Surveys

The area immediately adjacent to the Taylor Hall section which houses the reactor will be surveyed to a distance of 10 m (33 feet), primarily to assess the impact of such activities as waste packages removal from the Reactor Room. The survey will proceed much as the building survey except that alpha measurements are not considered necessary.

3. Soil and Water Sampling

a. Soil Sampling

The exact number, extent and analysis of soil samples will be determined based on the results of the bata-gamma measurements in the outdoor survey.

Random samples from the outdoor survey blocks as well as samples from "hot spots" identified during the scanning survey will be taken and analyzed.

b. Water Sampling

Water samples will be taken from each potentially contaminated source of water on the site.

7.2 Termination Radiation Survey Report

A report will be written and submitted to the NRC on the Termination Radiation Survey as required by Regulatory Guide 1.86.

The report will include a description of the survey methods, instruments, analyses, and an evaluation of the results. The report is expected to conclude that the site is suitable for release to unrestricted use.