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COLLEGE OF ENGINEERING
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Department of Mechanical Engineering
Nuclear Engineering Program
512—471-5136

May 2, 1983

Mr. Cecil O. Thomas, Chief
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Thomas:

The enclosed information provides response to the question on license renewal for Docket No. 50-192. In reply to questions presented by NRC letter dated April 15, questions 4, 5, 6 and 16 have been answered by providing appropriate update pages (enclosed) for the University of Texas TRIGA Reactor Safety Analysis Report. The remaining questions are addressed by the enclosed attachment.

Sincerely yours,

Thomas L Bauer

T. L. Bauer
Reactor Supervisor
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Dale Klein, Director
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Enclosures
TLB:DK:mgm

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FORMAL REVIEW QUESTION RESPONSE

I. Reactor Parameters (questions 1, 2, and 3)

From periodic measurements, values of specific University of Texas TRIGA Reactor parameters are routinely updated to represent current reactor core configurations. Values for reactivity of the UTTR core on April 1, 1983 indicated a core excess reactivity of $2.73 (1.91%). Control rod worths at the same time indicated values of $2.07 (1.45%) for transient rod, $3.02 (2.11%) for shim rod and $1.36 (.95%) for the regulating rod.

Reactivity loss at 250 kW power is $1.56 (1.09%) and typical fuel element temperature in the B ring is 215°C. Temperatures in the B ring following a $2.00 pulse insertion are typically 275°C with maximum values of about 325°C. The peak temperature depends on the period of time that the transient rod is removed from the reactor core (<15 secs) and reaches a peak value between 5 to 10 secs following a 2 second pulse insertion.

II. Radiation Protection (questions 8-14)

Daily activities of the radiation protection program at The University of Texas TRIGA Reactor facility are the responsibility of the Reactor Supervisor. Two university committees, the College of Engineering Nuclear Reactor Committee and University Radiation Safety Committee provide policy and review of activities. Administration of laboratory operation is through the Dean of the College of Engineering by way of the Chairman of the Mechanical Engineering Department and Director of the Nuclear Engineering Teaching Program. The Nuclear Reactor Committee monitors reactor safety and other laboratory functions of USNRC licenses R-92, SNM-180 and

TDH state license 6-485. The broader responsibility for the university's general state license for radioactive materials is monitored by the University Radiation Safety Committee. Administration of the state license is provided by way of the President of the University, Vice President for Business Affairs, and the Director of the Physical Plant through the University Safety Office. Daily activities of the university's radiation programs are the responsibility of the Radiation Safety Officer.

Equipment is maintained by the UTTR facility for detection of α, β, γ and neutron radiations, for performing area radiation surveys, counting radioactive contamination swipes and for personnel dosimetry measurements. The Reactor Supervisor implements appropriate measurements as situations indicate and assures that routine periodic measurements and equipment calibrations are maintained. Instruction, training, or supervision is provided as necessary to laboratory personnel, experimenters and students. Although the laboratory maintains the ability to calibrate instruments and perform routine or periodic radiation surveys, activities of the Radiation Safety Officer supplement laboratory activities by administering a vendor supplied film badge dosimetry program, performing semiannual swipe tests of sealed radiation sources, taking samples of reactor coolant water, counting swipes for area surface contamination, providing monitoring of transported radioactive materials and provising for disposal of low level radioactive wastes. The radiation safety officer also maintains a calibrated radiation source for survey instrument calibrations and periodically monitors the laboratory facility boundary.

Individual knowledge of the goal and principle of the "as low as reasonably achievable" concept is a major factor of the radiation protection program in the UTTR facility operation. The educational nature of the

facility and its small staff allows ready review or evaluation of specific radiation exposure situations. All personnel dosimetry reports are reviewed by the reactor supervisor. For unusual situations nonroutine situations or particularly hazardous situations, the university Radiation Safety Officer is consulted.

General procedures for radiation safety require supervision or appropriate knowledge on instruction of the potential hazards, proper handling techniques, and detection methods required for various radioactive materials. Personnel frequenting the laboratory are instructed of areas of radioactive material storage, use of radiation survey equipment and laboratory radiation sources. Routine surveys of reactor coolant water, area radiation levels and surface contamination levels are performed at least quarterly, but frequently supplemented with additional surveys to monitor particular laboratory activities. Any deviations from typically measured values or acceptable working area limits are reviewed and activites curtailed if levels exceed anticipated values. All materials removed from the reactor pool are routinely monitored. Wipe tests for removable contamination are taken when appropriate. Dose levels of removed samples that are greater than 100 mr/hr at the surface are evaluated for leakage and removed for storage unless the activity is of very short half-life. Action levels for emergency situations from an unknown source are detection of 5000 cpm/hr increase of air filter particulate activity or 100 mr/hr measured on any area radiation monitor.

An area radiation monitoring system composed of 4 fixed location GM tube detectors provides gamma ray detection with logarithm readouts from .01 mr/hr to 100 mr/hr. Another GM tube detector monitors activity of pool water with a readout at the reactor console. The readout has

low and high sensitivity calibrated to readout 25 mr/hr and 2.5 mr/hr at full scale or the estimated equivalent of .0125 μCicc or .00125 μCicc, respectively, of water activity. Both the water monitor and the area radiation monitors have visual and audible alarms. A continuous air monitor also functions with a GM tube to detect radioactivity deposited on an air particulate filter. Readout scale is logarithmic from 50 to 50,000 cpm with system operation sensitivities of approximately 90% filter retention and 10% activity detection. Calibration of the fixed GM tube detectors is performed semiannually with a standard calibration source (~10 mCi Co^{60}) by checking a point on each decade of the area radiation monitor readout, and at least 2 points on each range of the water monitor and particulate monitor. Each week a small source is used to check the alarm setting and indication of each GM tube monitor.

Two portable instruments are routinely maintained for radiation surveys or radioactivity detections. Both utilize GM tube detectors with thin windows that allow detection of β's, scale range multiples of 1, 10, 100 and 1000, and detection sensitivities of 4000 cpm for 1 mr/hr dose measurements. Calibrations are performed on all ranges every six months by measurement of a standard radiation source at one or more points on each scale. The x1000 has limited accuracy because of counter dead times. Each survey instrument has an additional scintillation type detection device available for special purpose measurements. One detector is designed for neutron detection and the other for alpha detection. No routine calibrations of these detectors are performed but functional operation checks are done as necessary. Another instrument, an ionization chamber, is available to read doses from 0-300 mr/hr and although not routinely calibrated is frequently checked for correct readout indications.

All parameters in the reactor area that are alarmed with visual and audible indications are to provide information that directly or indirectly effect radiation safety. The only alarm not located at the console is the continuous air monitor that provides a warning or evacuation alarm for detection of high air particulate activity (1000 cpm and 5000 cpm). Each of the four area radiation monitors are alarmed with settings at 5 mr/hr. The water radiation monitor system is alarmed at calibrated indications of 2.5 mr/hr and 25 mr/hr. An additional alarm on the water system indicates undesired conditions of pool coolant level (with $\pm$ 3" of 16' above core), temperature (>100°F), or differential pressure (<4 psid) with the secondary cooling system. A visual indication at the console identifies which water parameter caused the alarm.

III. Reactor Component and System Design Life (question 7)

Protection of the reactor tank, structure and coolant system components is primarily achieved by system design and corrosion control. System design consists of the original fabrication procedures and selection of material types. An active evaluation of component failures occurs during all repair activities to maintain equivalent or improved system functions. Corrosion control depends on water quality and procedural review of potential experimental impacts on the coolant system. Maintenance of water conductivities generally average one third or less of the 6 μmho/cm specification for an average monthly value. The twenty years of operating experience at the UTTR have not indicated to date any major deterioration of any structures, components or systems associated with the reactor tank. Numerous expected minor problems occur as expected by any aging system and efforts are made to

review conditions that might indicate impending system problems. Radiological damage to systems and components represents a minor effect because of the relatively low burnup history (19.5 MW days) and average duty cycle (220 hours/year) of the reactor. Experience with periodic surveillance required by license specifications or facility implementation have been sufficient to identify changes in reactor systems to date.

Fuel elements, control rods, and electrical components of the reactor control system are frequently inspected or checked against specifications. Experience indicates that periodic inspections each two years is sufficient to identify significant chemical or mechanical changes to the exterior of control rods and fuel elements. More frequent semiannual inspections are made of reactor console systems to monitor the status of reactor operation and protection systems. Reactor operating personnel are instructed to recognize deviations from operating or design specifications of console equipment so that appropriate steps can be initiated. An active program exists to replace, when possible, reactor console systems or components that have become less reliable. Goals of the program are to update some console systems with more updated versions and increase redundancy of functions to provide alternate equipment when some repairs are required. The major part of the program to date was the installation of a wide range channel that provides full range measurement of reactor power.

Observations over the past years of operation do not indicate that any major reactor system repair is expected although it is anticipated that leaks to the coolant containment system, failures of neutron measurement detectors and other problems are possible if not likely. However, routine repairs, surveillance programs, and operation staff observation should be sufficient to assure safe reactor operation and timely identification of potential problems.

IV. Five Year Radiation Exposures (question 14)

A record of radiation exposures during the past 5 years is evidence of an effective radiation control program. The table includes personnel doses and area doses.

Five Year Dose Distributions

	personnel					area				
rem	1	2	3	4	5	1	2	3	4	5
Non-measurable	13	31	16	49	39	2*	1*	0	1*	1*
0.0 - 0.1	7	4	25	7	4	1	1	1*	0	0
0.1 - 0.25	0	0	0	0	0	0	0	0	1	0
0.25 - 0.5	0	0	0	0	0	1	0	2	0	3
0.5 - 0.75	0	0	0	0	0	0	1	0	0	0
0.75 - 1.0	0	0	0	0	0	0	0	0	1	0
1.0 - 2.0	0	0	0	0	0	0	1	0	0	0
2.0 - 3.0	0	0	0	0	0	0	0	0	1	0

*Includes control badge

drive housing. As the ball-nut is rotated by a worm gear, the cylinder moves up or down depending on the direction of worm gear rotation. A mechanical indicator driven by the worm shaft provides a monitor of the position of the cylinder and the distance the transient rod will be ejected from the reactor core. Operation for pneumatic cylinder positioning is controlled by a crank inserted at the rod drive. The crank is inserted into the drive only when position changes are to be made.

Attached to and extending downward from the transient rod drive housing is the rod guide support, which serves several purposes. The air inlet connection near the bottom of the cylinder projects through a slot in the rod guide and prevents the cylinder from rotating. Attached to the lower end of the piston rod is a flanged connector that is attached to the connecting rod assembly that moves the transient rod. The flanged connector stops the downward movement of the transient rod when the connector strikes the damp pad at the bottom of the rod guide support. A microswitch is mounted on the outside of the guide tube with its actuating lever extending inward through a slot. When the transient rod is fully inserted in the reactor core, the flange connector engages the actuating lever of the microswitch and indicates on the instrument console that the rod is in the core.

In the case of the safety-transient rod a scram signal de-energizes the solenoid valve which supplies the air required to hold the rod in a withdrawn position and the rod drops into the core from the full out position in about 1 second.

3.4.7.3. Evaluation of Control Rod System. The reactivity worth and speed of travel for the control rods are adequate to allow complete control of the reactor system during operation from a shutdown condition to full power. The scram times for the rods are quite adequate since the TRIGA system does not rely on speed of control as being paramount to the safety of the reactor. The inherent shutdown mechanism of the TRIGA prevents unsafe excursions and the control system is used only for the planned shutdown of the reactor and to control the power level in steady-state operation.

For the accident considered here, it is assumed a fuel element in the region of highest power density fails in water and that the peak fuel temperature in the element is less than 300°C. At this temperature, the long-term release fraction would be less than 1.5×10^{-5}. For the purpose of this analysis it is also assumed that 100% of the noble gases and 50% of the halogens are released from the highest power density fuel element in which 2.6% of the total power is generated.

It is important to note that the release fraction in accident conditions is characteristic of the normal operating temperature and not the temperature during the accident conditions. This is because the fission products released as a result of a fuel clad failure are those that have collected in the fuel-clad gap during normal operation.

Other assumptions concerning estimated accident scenario doses are:

1. Assume an element fails in air such that all noble gases (100%) and halogens (100%) in the gap are effectively released.

2. After the failure a ventilation rate of 4 air changes per hour is maintained with no air filtration.

3. There is no plate-out of any release fission products.

4. Doses are calculated for releases external to the building (dilution factor 0.42 sec/m^3, wind velocity 1.0 m/sec).

5. Doses are calculated for releases internal to the building (dilution factor 1.667 sec/m^3, wind velocity 0.2 m/sec).

The net effect of these assumptions is that for the accident condition, the fraction of the noble gases released from the building is:

$$f_{NG} = 1.5 \times 10^{-5} \times 1.0 \times 2.6 \times 10^{-2} = 3.9 \times 10^{-7} \quad ,$$

and of the halogens:

$$f_H = 1.5 \times 10^{-5} \times 1.0 \times 2.6 \times 10^{-2}$$
$$= 3.9 \times 10^{-7} .$$

8.1.3. Downwind Dose Calculations

The minimum roof level dilution factor was calculated, in Section 5.4 , to be 4.2×10^{-2} sec/m^3. This is based on mixing in the lee of the building when the wind velocity is 1 m/sec.

The calculation of whole body gamma doses and thyroid doses downwind from the point of release was accomplished through the use of the computer code GADOSE (Ref. 2). In this code the set of differential equations describing the rate of production of an isotope through the decay of its precursors and the rate of removal through radioactive decay and removal by the ventilation system is integrated for each member of the chain. The release rate q_i to the environment for the ith isotope at time t_i, in hours is:

$$q_i(t) = g_i \, Q_i(t) \; (\ell/V)/3600 \quad ,$$

where $Q_i(t)$ = the release of the ith isotope in Ci,

ℓ/V = the building leakage rate in $(m^3/hr)/m^3$,

$g_i = 1 - \varepsilon_i$,

ε_i = the filter efficiency for the ith isotope.

The quantity $Q_i(t)$ is the amount of the ith isotope in the discharged air at the time, t. This quantity is given by

$$Q_i(t) = f_i Q_i(0) e^{-(\lambda_i + \ell/V)t}$$

where $Q_i(0)$ = the inventory of the ith isotope as found in Table 8-1,

λ_i = the decay constant for the ith isotope, and

f_i = the release fraction to the reactor hall.

The concentration downwind at a distance x for the ith isotope is calculated from

$$Q_i'(t,x) = q_i(t-\tau) \cdot \psi(x) e^{-\lambda_i \tau} \quad ,$$

where τ = the transit time from the release point to the dose point, hr,

ψ = the dilution factor at the distance x, sec/m^3.

The whole body gamma ray dose rate for the ith isotope, D_{w_i}, at the distance x and time t is calculated, assuming a semi-infinite cloud, through the expression:

$$D_{w_i}(t,X_i) = 900\ \overline{E}_i Q_i'(t,x) \quad ,$$

where $\overline{E}_i$ = the average gamma ray energy per disintegration, MeV, and the constant includes the attenuation coefficient for air as well as the conversion factors required.

Internal dose rates, in this case the dose rate to the thyroid, are calculated by:

$$D_{th_i}(t,x) = 3600\ B \cdot Q_i'(t,x) K_i \quad ,$$

where B = the breathing rate, m^3/sec, and

K_i = the internal dose effectivity of the ith isotope, rem/Ci.

The values for the breathing rate are given in Table 8-2 and are taken from USAEC Regulatory Guide 4.

The average gamma ray energy per disintegration and the internal dose effectivity for each isotope considered are given in Table 8-3.

The decay products of these isotopes are also included in the calculation; however, their contribution to the dose rates are small and therefore the data for these isotopes were not included in the table.

8.1.4. Downwind Doses

The whole body gamma dose and thyroid dose in the lee of the building are shown in Table 8-4. These doses are trivial in nature.

TABLE 8-2
ASSUMED BREATHING RATES

Time (hr)	Breathing Rate (m^3/sec)
0 to 8	3.47×10^{-4}
8 to 24	1.75×10^{-4}
Over 24	2.32×10^{-4}

TABLE 8-4

DOSES FROM FISSION PRODUCT RELEASE

(single element release in air)

Location	Distance	0.1/1.0/8.0 hours Whole Body Gamma (mrad)	Thyroid (mrad)
Outside	0	.25/.61/.62	16/47/48
Hallway	0	10.1/24.4/24.7	640/1878/1911

8.2 LOSS OF REACTOR COOLANT

8.2.1 Summary

The reactor will operate at a calculated maximum power density of 6 kW/element when the reactor power is 250 kW and there are 63 elements in the core, all of which are standard TRIGA fuel. If the coolant is lost immediately after reactor shutdown, the fuel temperature (see Fig. 8-1) will rise to a maximum value of ~275°C. The stress imposed on the fuel element clad by the internal gas pressure (see Fig. 8-2) is about 1200 psi when the fuel and clad temperature is 275°C and the yield stress for the clad is about 37,000 psi. Therefore, it can be concluded that the postulated loss-of-coolant accident will not result in any damage to the fuel, will not result in release of fission products to the environment, and will not require emergency cooling.

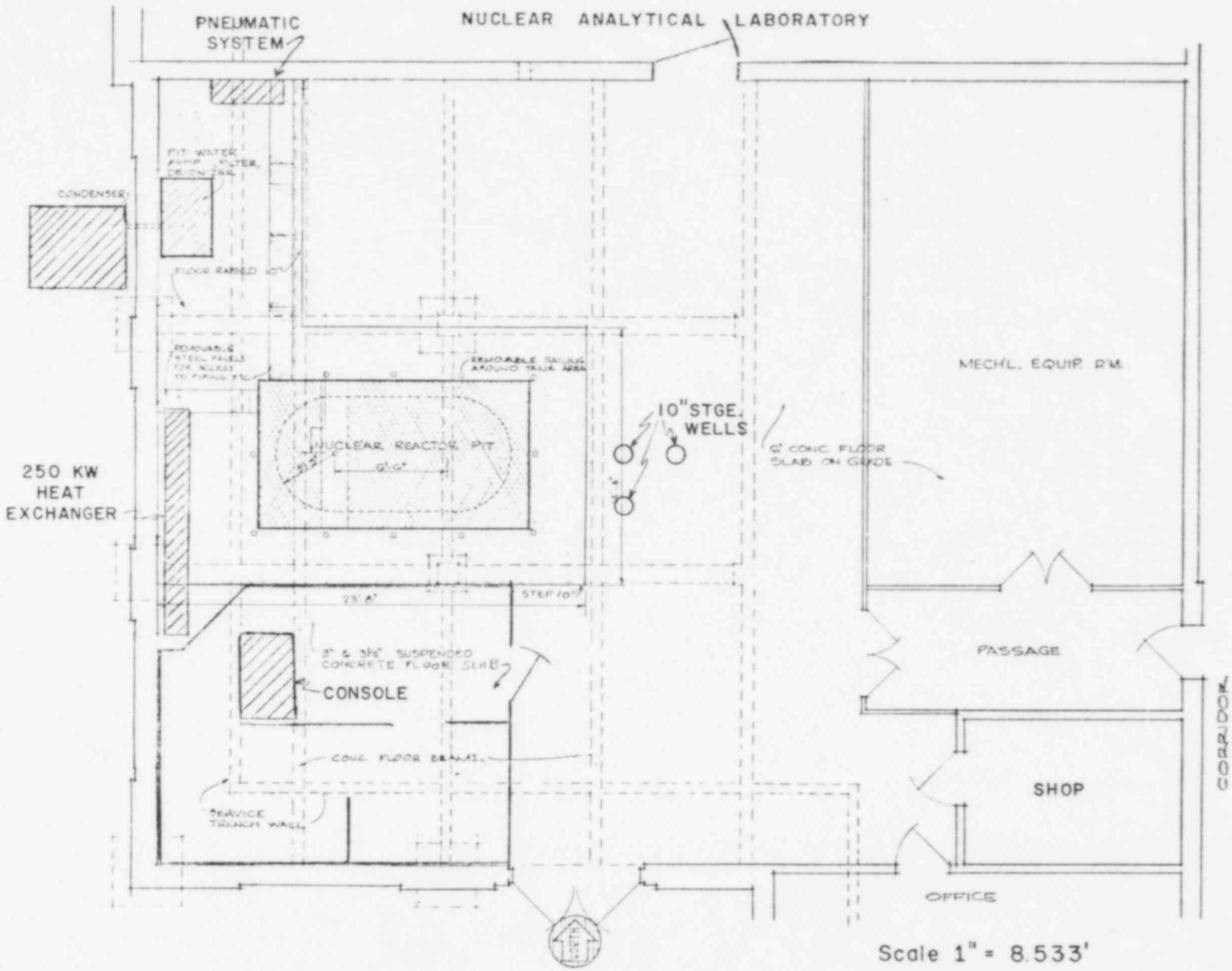

Fig. 5-1. TAYLOR HALL 131 FLOOR PLAN

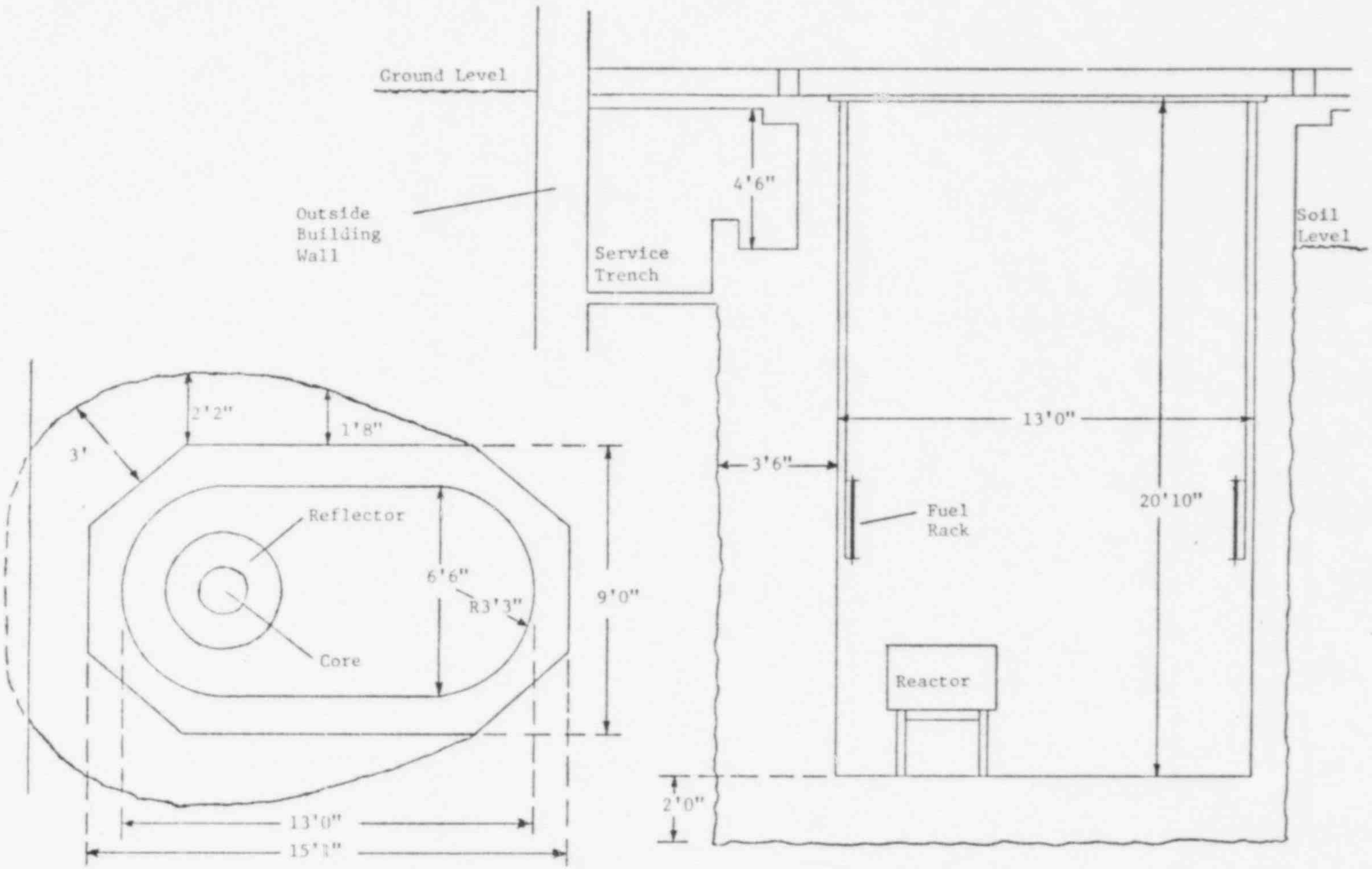

Fig. 5-2 Reactor Tank and Shield Structure

to the reactor room and create a restricted access entry passage to the laboratory (see Figure 5-3).

The ceiling height in the main part of the laboratory is about 16 feet to the bottom of the steel girders and about 17 feet between the girders. The ceiling (roof) is supported by steel girders and consists of 4.5-inch gypsum board and fill, one inch of rigid insulation, water proofing, and built of roofing. The main floor area of room 131 with 17 foot ceiling height consists of 1445 ft^2 floor area with approximately 24,500 ft^3 of air volume. A plaster board and partition structure in the room encloses 2400 ft^3 as a console operation area. Shop and work areas on the east side of the laboratory add, respectively, 147 ft^2 and 298 ft^2 of floor space with volumes of 2495 ft^3 and 4470 ft^3. The ceiling construction varies from 20 ft to 12 ft heights, fabricated from concrete or truss and board design. Adjacent room space compromising the mechanical equipment room, entry way, and shop area totals about 750 ft^2 of floor space and about 12000 ft^3 of air volume.

All doors to the reactor laboratory are of solid core construction with intrusion alarms. Two entrances may be electronically controlled. Each entrance, the utility access trench and roof area are provided with visual warning signals of reactor operation.

5.3 ROOM ISOLATION

Although the original analysis by General Atomic based on experimental data had shown that no special requirements for a building were required to install a TRIGA reactor, several modifications have been made to assure reasonable control of potential radioactive material releases. The immediate area of the reactor room contains a closed ventilation system with both heated and cooled air generated in the reactor room by supplies of steam and chilled water respectively. Other utility supplies to the reactor room include heat exchanger chilled water, high pressure air, and electrical power.

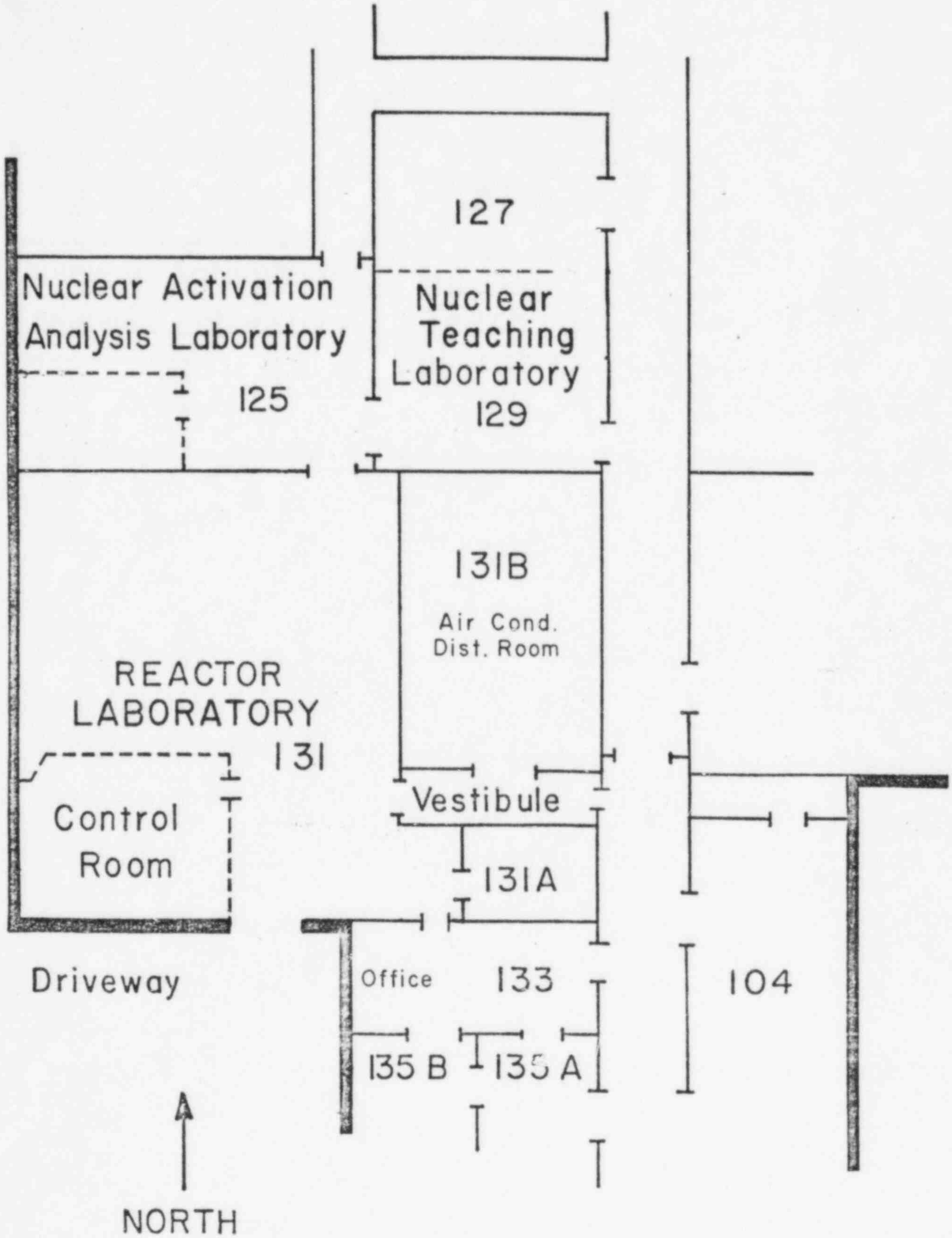

Fig. 5-3 TAYLOR HALL FLOOR PLAN
ADJACENT ROOMS TO 131

The ventilation air in the reactor room area (~31,500 ft^3) is restricted to the room by recirculation. Leakage of air from the reactor room directly to the environment through the emergency exit to the south is limited by weatherstripped doors that will remain closed during reactor operation. Other potential direct leakage paths to the building through doors or penetrations are either weatherstripped or sealed. Besides the emergency exit, two laboratory staff offices with a volume of 4,000 ft^3 are located to the south. Leakage from the north side of the reactor room is into a controlled access laboratory area of approximately 12,400 ft^3 (plus an adjacent room with a volume of ~25,000 ft^3).

Isolation of the laboratory to the east is provided by a buffer zone consisting of the entry way and adjoining ventilation equipment room. The entry way provides two doors to restrict air leakage to building corridors. The ventilation system, whose intake is located in the corridor outside the entry way, has a capacity of 15,600 ft^3/min. and ventilates adjacent offices, laboratories, classrooms, and corridors consisting of about 78,400 cubic feet.

A high volume exhaust system for the reactor room air has been installed. The system has a capacity of 1500 cubic feet per minute through a prefilter, high efficiency particulate filter, pneumatic damper and exhaust stack. The exhaust system is not to be operated concurrently with reactor operation and the damper is routinely maintained closed.

Other low volume room air leakage occurs through a glove box and sample pass through port. Operation of the glove box is manually operated and exhausts room air through absolute filters. The sample port is interlocked so that inner and outer doors are not opened at the same time.

The concentration downwind from the point at which the activity is discharged from the building is

$$A_D = Aq\ \psi(x) \quad ,$$

where ψ = the dilution factor at the distance x, (sec/m^3),
A = the activity concentration in the discharge (Ci/m^3),
q = the building exhaust rates (m^3/sec).

The dilution factor in the lee of the building (x=0), if it is assumed that the discharge is at the roof line, is given (Ref. 3) by:

$$\psi(0) = \frac{1}{csu}$$

where c = a constant (0.5),
s = building cross-sectional area normal to the wind direction (m^2),
u = wind velocity (m/sec).

The whole body gamma ray dose rate to a person immersed in a semi-infinite cloud of radioactive gases can be approximated by

$$D = 900\ EA_D$$

where E = the photon energy.

Leakage of laboratory air to adjacent areas of the building without a laboratory ventilation system operating would be diluted by the high flow rate of the adjacent area ventilation unit. Assuming 100% leakage of laboratory air to adjacent areas at a rate of 2 air volume exchanges per hour the previous analysis yields a leakage source term of 132 argon-41 atoms/cm^3 or 3.78×10^{-7} μCi/cm^3. The activity discharge rate to adjacent areas is

$$Aq = 3.78 \times 10^{-7} \times 3.78 \times 10^5 = 1.43 \times 10^{-7} \text{ Ci/sec}$$

The recirculation of adjacent area air results in a possible maximum concentration of

$$(24000/78000) \times 3.78 \times 10^{-7} = 1.17 \times 10^{-7}\ \mu c/cm^3 \quad .$$

Averaged over one year of 250 kw full power operation 5 days/week, 8 hrs/day the minimal averaged concentration is

$$1.17 \times 10^{-7} \times (5/7) \times (8/24) = 2.79 \times 10^{-8}\ \mu c/cm^3 \quad .$$

The average dose rate represented by the leakage would be

$$D = 900 \times 1.3\ (2.79 \times 10^{-8}) = 3.26 \times 10^{-5}\ \text{rads/hr},$$

or an average of 32.6 μrad/hr, and a peak of 137 μrad/hr.

The ventilation flow rate of adjacent building areas provides a dilution factor, ψ, of $(7.08\ m^3/sec)^{-1}$. Argon-41 concentration in leakage air is

$$Aq\ \psi = 1.43 \times 10^{-7}/7.08 = 2.02 \times 10^{-8}\ \mu Ci/cm^3 \quad .$$

For operation of a laboratory ventilation system at $7.08 \times 10^5\ cm^3/sec$ (3.75 volume exchanges per hour) the previous analysis predicts a room activity of $2.18 \times 10^{-7}\ \mu Ci/cm^3$ and the activity discharge rate would be

$$Aq = 2.2 \times 10^{-7} \times 7.1 \times 10^5 = 0.16\ \mu Ci/sec = 1.6 \times 10^{-7}\ Ci/sec \quad .$$

The minimum cross-sectional area is $47.6\ m^2$ (32 x 16 ft at laboratory roof level) and, for a wind velocity of 1m/sec,

$$\psi(0) = \frac{1}{0.5 \times 47.6 \times 1.0} = 4.2 \times 10^{-2}\ \frac{sec}{m^3} \quad .$$

The maximum argon-41 concentration outside the building would be

$$A_D = 1.6 \times 10^{-7} \times 4.2 \times 10^{-2} = 6.7 \times 10^{-9}\ Ci/m^3 \quad .$$

Thus, the maximum downwind dose rate resulting from discharge of argon-41 produced in the reaction is

$$D = 900\ EA_D$$
$$= 900 \times 1.3\ (6.7 \times 10^{-9})$$
$$= 7.8 \times 10^{-6}\ \text{rads/hr} = 7.8\ \mu\text{rad/hr} \quad .$$

The actual effect of argon-41 releases from the reactor pool would be substantially less than those estimated as a result of the various conservative estimates in the calculation. Among the major conservative assumptions are the transfer amounts of argon from the pool surface, period of full power operation, release rates and volumes.

5.4.1.2. Nitrogen-16 Activity in Reactor Room. The cross-section threshold for the oxygen-16 (n,p) nitrogen-16 reactions is 9.4 MeV; however, the minimum energy of the incident neutrons must be about 10.2 MeV because of center of mass corrections. This high threshold limits the production of nitrogen-16 since only about 0.1% of all fission neutrons have an energy in excess of 10 MeV. Moreover, a single hydrogen scattering event will reduce the energy of these high-energy neutrons to below the threshold. The effective cross-section of oxygen-16 (n,p) nitrogen-16 reaction averaged over the TRIGA spectrum is 2.1×10^{-29} cm^2. This value agrees well with the value obtained from integrating the effective cross section over the fission spectrum.

The concentration of nitrogen-16 atoms per cm^3 of water as it leaves the reactor core is given by

$$N_2 = \frac{N_1\ \sigma_1\ \phi_v}{\lambda_2} \left(1 - e^{-\lambda_2 t}\right) \quad ,$$

where N_2 = nitrogen-16 atoms per cm^3 of water,

ϕ_v = neutron flux (0.6 - 15 MeV) = 3.0×10^{12} n/cm^2-sec,

N_1 = oxygen atoms per cm^3 of water = 3.3×10^{22} atoms/cm^3,

σ_1 = (n,p) cross section of oxygen = 2.1×10^{-29} cm^2 (averaged over 0.6 - 15 MeV),

λ_2 = nitrogen-16 decay constant = 9.35×10^{-2} sec^{-1},

t = average time of exposure in reactor.