MICHIGAN STATE UNIVERSITY

VICE PRESIDENT FOR FINANCE AND OPERATIONS AND TREASURER
412 ADMINISTRATION BUILDING
EAST LANSING · MICHIGAN · 48824

June 11, 1984

Mr. Cecil O. Thomas, Chief
Standardization and Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Docket 50-294, Formal Review Questions--Your Letter Dated May 11, 1984

Dear Mr. Thomas:

Pursuant to your request, I am enclosing 3 copies of the replies to your questions regarding the relicensing of the Michigan State University Reactor, R-114. Additionally requested material (Revisions to the Safety Analysis and revised Technical Specifications) are being prepared and will be submitted shortly.

Very truly yours,

Stephen H. Terry

Stephen H. Terry
Assistant Vice President for Finance

SHT/bbc/0398F

cc: Bruce Wilkinson

Enclosures

STATE OF MICHIGAN)
)ss.
COUNTY OF INGHAM)

Subscribed and sworn to before me, a Notary Public, this 11th day of June, 1984.

Virginia Stewart

Notary Public, Ingham Co., Mich.
My Commission Expires March 6, 1985.

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Michigan State University
Docket 50-294 Formal Review Questions, MSTR
(Reference: NRC Letter Thomas → Wilkinson 5/11/84)

1. What are the principal uses of the MSTR? What is the current use in megawatt-hours per year? how frequently is the reactor pulsed?

   The three principle uses for MSTR are neutron activation analysis, isotope generation and engineering training. Between January 1, 1983 and December 31, 1983 the total operation of the reactor was 5166 kilowatt-hours. The reactor is pulsed semi-annually with at least 1.0% $\Delta k/k$.

2. Provide a plan view of the current core configuration showing the number and location of the fuel elements (differentiate between 8.5 and 12 weight-per cent elements), control rods, graphite reflector assemblies, experimental tubes, and the startup source.

   Figure 1 shows the current core configuration. All positions are filled with fuel elements except those marked "G", representing graphite reflector elements, the three control rod positions, marked reg. shim, and pulse, and the source position marked source. Currently there are no experiment tubes in the core grid plate except the central thimble, marked C.T.

3. What is the total ^{235}U content in the core? What is the current startup source and what is its strength?

   The total ^{235}U content of the core is 2528 grams. The startup source is an Americium-Beryllium source of 1.88 curies.

4. What is the fuel to moderator ratio for the current core?

   2528_g ^{235}U in core

   148706_g ^{91}Zr in core $[Zr = \frac{2528/0.2}{0.085} = 148706]$

   6.4759×10^{24} Atoms ^{235}U $[U = \frac{2528}{235} \times 6.02 \times 10^{23} = 6.4759 \times 10^{24}]$

   9.8374×10^{26} Atoms ^{91}Zr $[Zr = \frac{148706}{91} \times 6.02 \times 10^{23} = 9.8374 \times 10^{26}]$

   1.67×10^{27} Atoms H in core Fuel $[9.8374 \times 10^{26} \times 1.7 = H]$

   core volume $= \frac{\pi D^2}{4}$ (H) $= 4.127.64$ in^3 $\begin{bmatrix} D = 19.385 \\ H = 14 \text{ in} \end{bmatrix}$

   water volume = core volume - Fuel volume

   $$= 4.127.64 - 91 \left(\frac{\pi(1.5)^2}{4} \right) 14$$

5/84

$= 4.127.64 - 2251.34 = 1876.3 \text{ in}^3 = 30747 g\ H_2O$

2.057×10^{27} Atoms H in core water $[\frac{30747}{18} \times 6.02 \times 10^{23} \times 2 = H]$

Total core H $= H_{Fuel} + H_{water}$

$= 1.67 \times 10^{27} + 2.057 \times 10^{27} = 3.73 \times 10^{27}$

Fuel to moderator ratio =

^{235}U in core ÷ H in core $= 6.4759 \times 10^{24} \div 3.73 \times 10^{27}$

$= 1.736 \times 10^{-3}$

5. What are the excess reactivity and control rod worths in the current core?

   When k = 1 the reactivity inserted in \$3.83 or 0.02681 Δk/k. Total reactivity of the current core is \$6.83 or 0.04781 Δk/k. Therefore excess reactivity is \$6.83 - \$3.83 = \$3.00 or 0.021 Δk/k.

   The individual control rod worths are:

   Reg = \$1.85 = 0.01295 Δk/k
   Shim = \$3.03 = 0.02121 Δk/k
   Pulse = \$1.95 = 0.01365 Δk/k

6. Provide schematic drawings of the primary cooling system and the water purification system.

   Figu.e 2 shows the cooling system and water purification shunt. These two systems are interdependent.

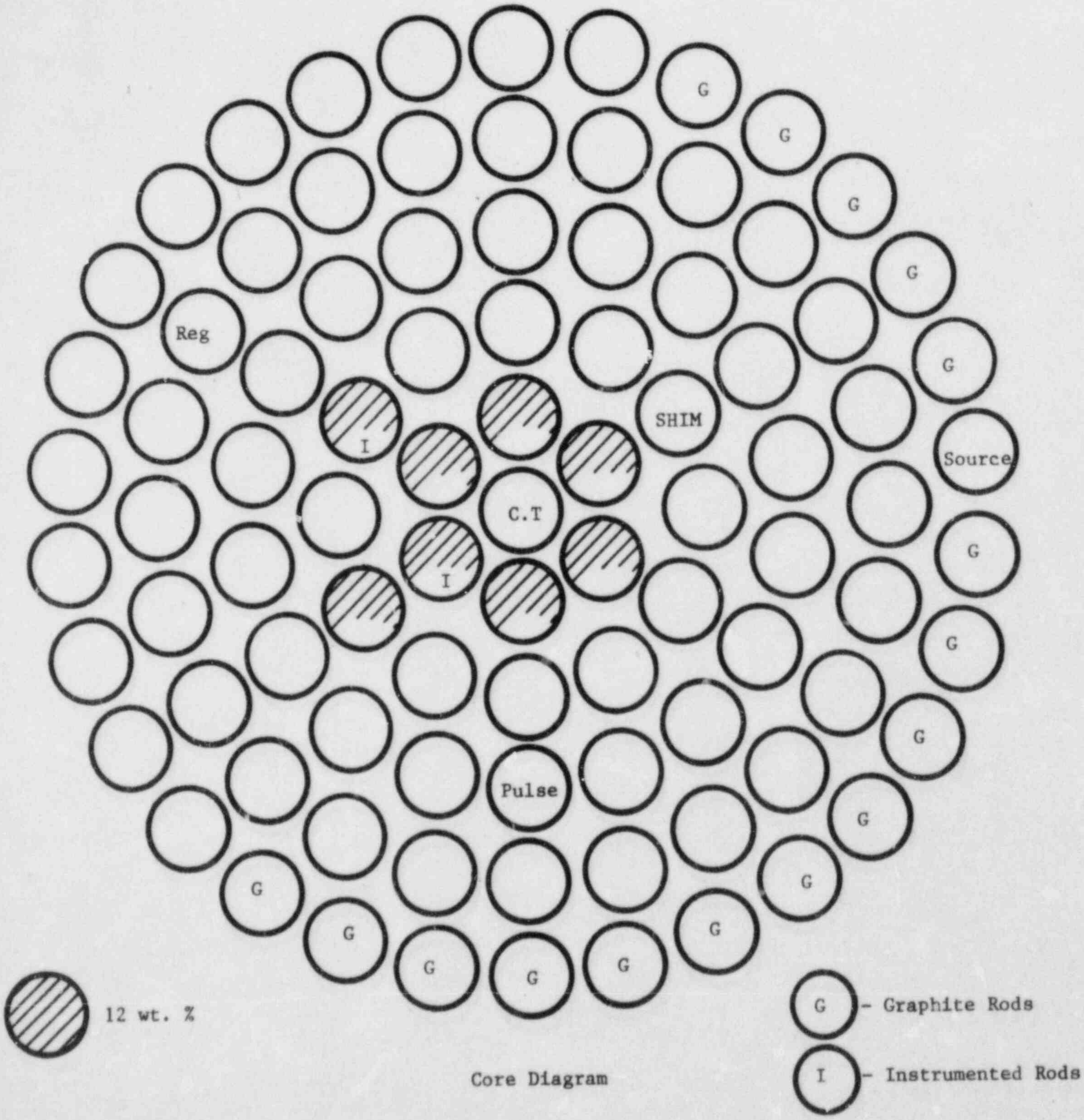
7. What isotopes are found normally in the reactor pool water and in what concentrations?

   The pool water analysis of February 2, 1984 shows that gross β exclusive of tritium is 2.52×10^{-9} μ Ci/cc. The tritium analysis on that date shows 1.1×10^{-4} μ Ci/cc.

8. What are the normal evaporation losses from the pool? Describe the water make system.

   Normal evaporation loss equaled 51.6 gallons per month averaged between September 1983 and March 1984. Distilled water is brought into Room 184 in a 50 gallon carboy. The water is drained manually to the pool.

9. Is there more than one conductivity monitor in the coolant purification system? Where are the readouts for the conductivity monitors? How often are the conductivity monitors calibrated?



Core Diagram

Figure 1.

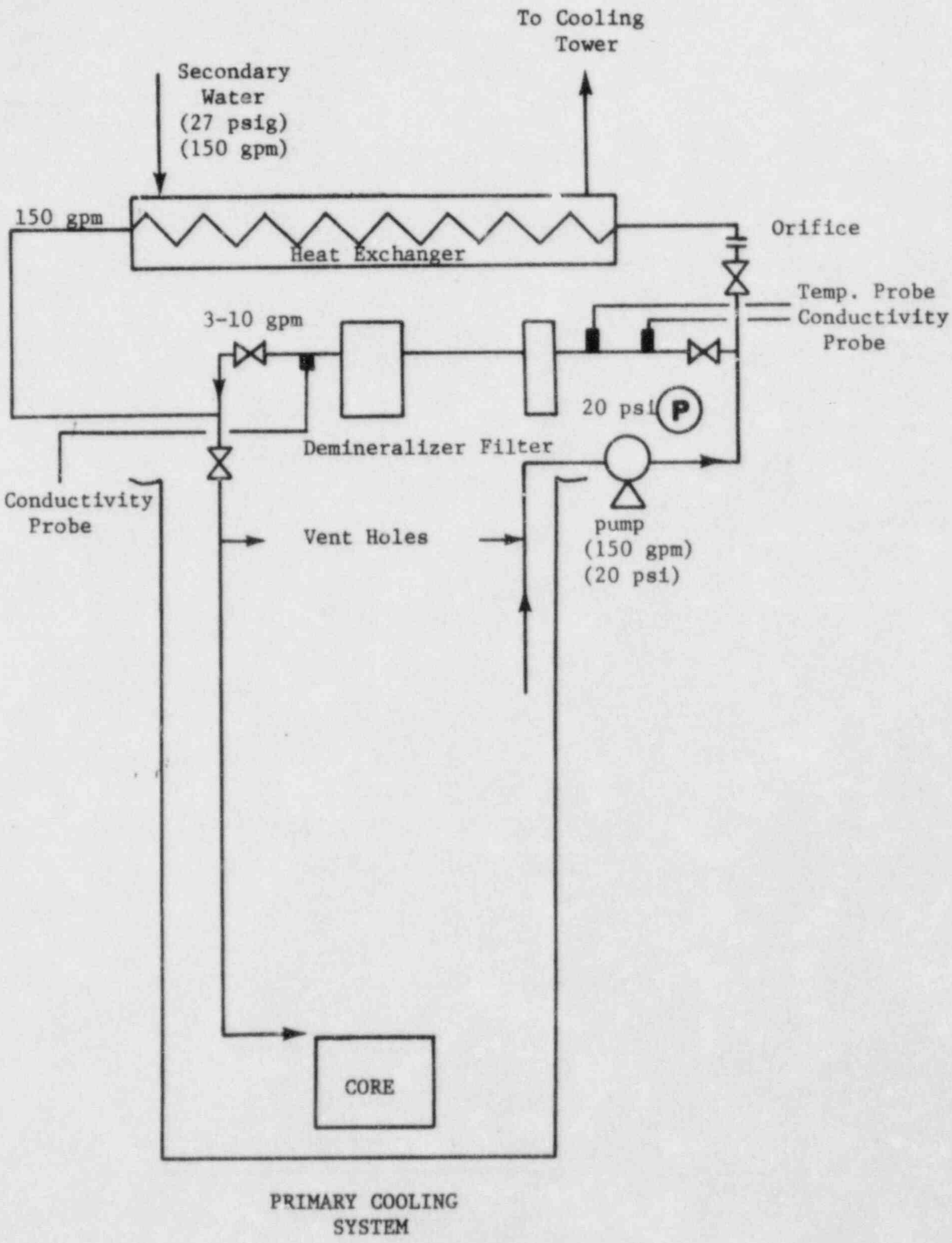

To Cooling Tower
Secondary Water (27 psig) (150 gpm)
150 gpm
Heat Exchanger
Orifice
Temp. Probe
Conductivity Probe
3-10 gpm
20 psi
P
Demineralizer Filter
Conductivity Probe
Vent Holes
pump (150 gpm) (20 psi)
CORE
PRIMARY COOLING SYSTEM

Figure 2.

There are two conductivity cells, one before and one after the demineralizer on the filtration shunt of the primary water system. The monitor readouts are at the console. The cells are compared with each other as part of the daily check list. The cells are not calibrated unless readings are suspect.

10. Provide a description of the heating system at the MSTR.

    There are two sources of heat in Room 184. Building air is brought in through the console room and outside air is brought in through the unit ventilator on the east wall of Room 184. In the ventilator, the air moves through thermostatically controlled heating coils, past automatic dampers and into the room.

11. Provide a schematic drawing of the ventilation system by-pass through the absolute filter. How are the monitoring system and the damper function monitored and controlled from the console?

    Figure 3 is a schematic of the ventilation by-pass. Damper function is monitored and controlled at the console. The operator has a button marked "vent" with dual lights to indicate damper position.

12. What kind of fire protection systems are available at the reactor facility? Which fire departments respond to fire-related emergencies at the reactor facility? Who is responsible for maintaining fire extinguishers at the facility?

    Fire extinguishers are on site. A fire hose is available in the hallway. The City of East Lansing operates a fire station 3 blocks west of the Engineering Building. The emergency plan contains a letter of agreement with the City of East Lansing Fire Department to provide fire protection support. The Department of Public Safety through the Office of the Campus Safety Engineer maintains the fire extinguishers.

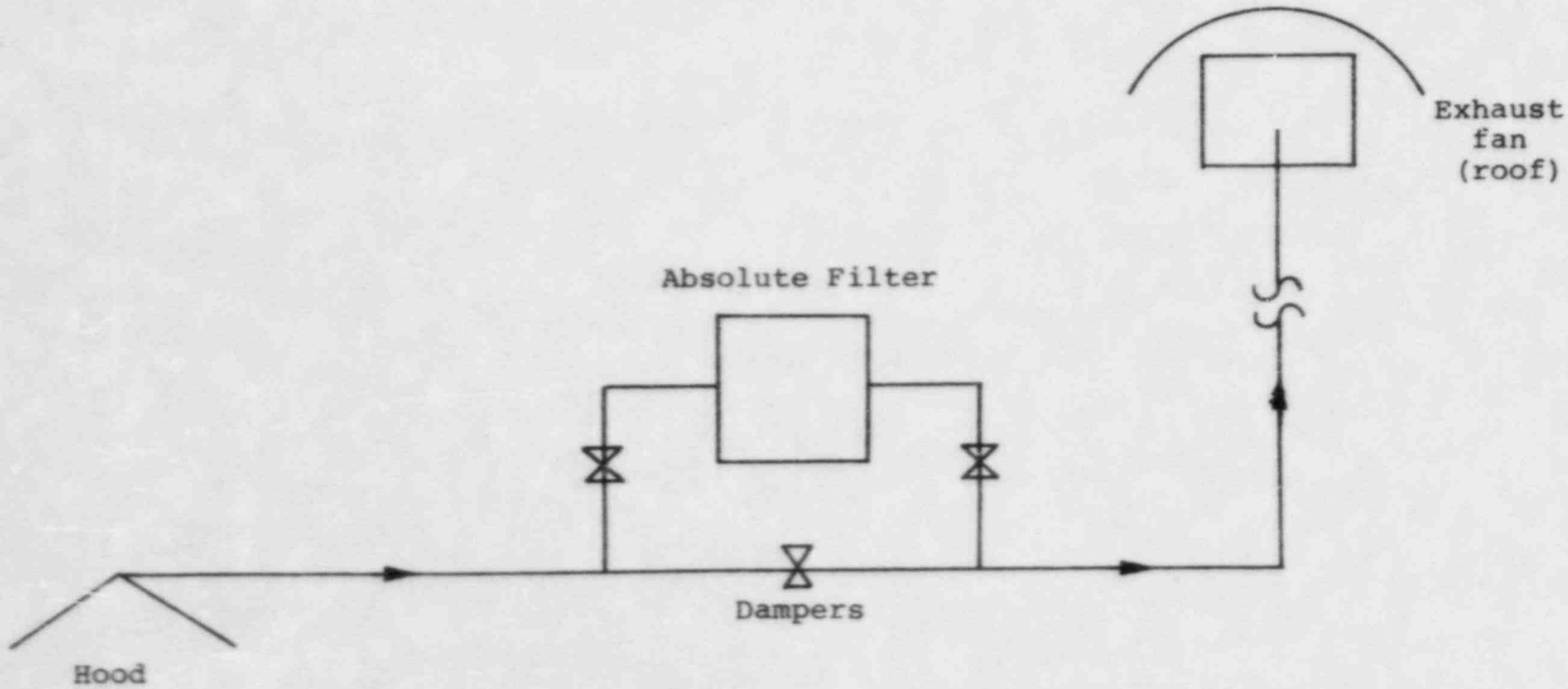
13. What kinds of communication systems (intercom, internal telephone system, commercial telephone services, and so on) are available at the reactor facility?

    Commercial telephone system is the only communication system at the reactor facility.

14. Describe the compressed air system(s) at the reactor facility, including details of installation and uses.

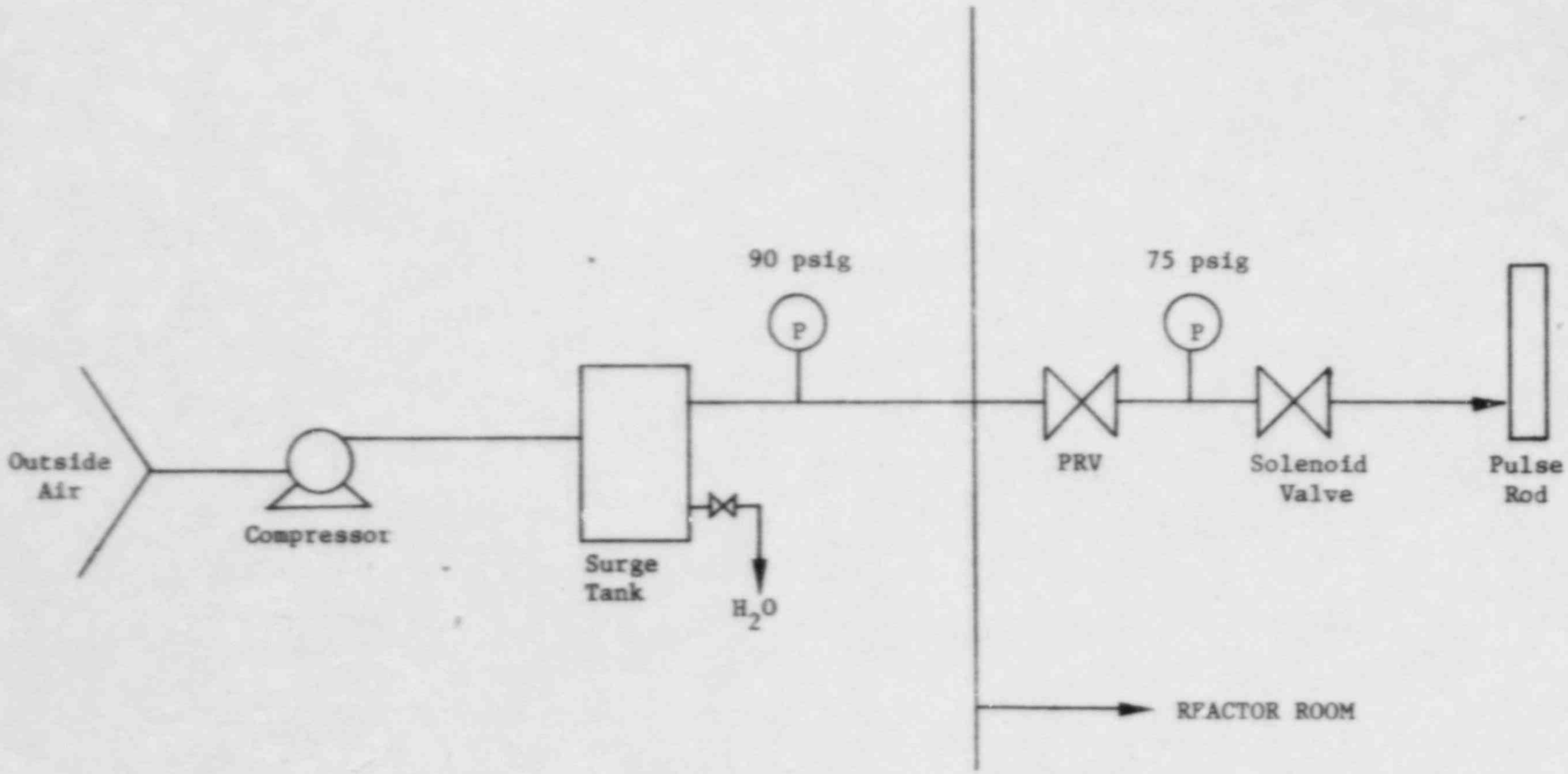
    Figure 4 shows a schematic of the compressed air system to Room 184. The only use for compressed air in Room 184 is to drive the pulse rod.

15. How many spare fuel elements are there? How many have been irradiated? Where are they stored?



Ventilation By-pass

Figure 3.



Compressed Air System

Figure 4.

There are 14 total spare stainless steel fuel elements in reactor pool storage racks. The 14 elements are stainless steel clad, 8.5% by weight. In addition, 12 aluminum clad elements are stored in the reactor pool storage racks (these elements will not be used in the core). All spare fuel has been irradiated.

16. What volume of spent ion exchange resins are generated annually from the coolant maintenance operations? What kind of radioactivity levels have been observed in these materials in the past?

The ion exchange resin was last replaced in 1973. Our records indicated very low radiation levels at that time. Because we use distilled make-up water we do not foresee changing the exchange resin the near future. There is approximately 4 cubic feet of resin in the ion exchange resin canister.

17. Describe the liquid radwaste management program.

There is no significant radioactive waste produced except occasional liquid irradiated samples.

18. Describe the solid radwaste management program.

All solid radwaste from room 184 is accumulated and held at the reactor facility until time of shipment. At the time of shipment a DOT approved 55-gallon steel drum, lined with 4 mil poly liners, is brought to the reactor facility. The drum is packed, monitored, labeled and loaded on the commercial waste hauler's truck and sent to a burial site at Richland, Washington.

19. Summarize the quantities of liquid and solid radioactive waste resulting from reactor operations for the last 5 yr (total activity of each physical form at times of release or shipment for each year).

There were 3 pickups of solid radwaste from the reactor facility over the past 5 years. There were no liquid pickups. The solids consisted of rubber gloves, paper and used sample holders. The total activity of each pickup was < 0.05 μCi/gm of material.

20. Describe the facility electrical power system and list all controls and instrumentation that are provided with emergency back-up power.

Normal electrical power is 110/220/440 V line power. The following systems are on an emergency back-up power system as well:

a) Intrusion alarm
b) Area alarm
c) Particulate alarm
d) Argon-41 monitor

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    c) Particulate alarm
    d) Argon-41 monitor

21. Provide definitions for the following terms as used in your Technical Specifications.
    (1) Secured experiments
    (2) Nonsecured experiments
    (3) Movable experiments
    (4) Irradiations

<u>Secured Experiment</u>

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulics, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

<u>Nonsecured Experiment</u>

A non-secured experiment is one which does not comply with the requirements set forth for a secured experiment.

<u>Movable experiment</u>

A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the core while the reactor is operating.

<u>Irradiations</u>

Exposure of samples to neutrons and/or gamma radiation in either rotary specimen rack, central thimble or other experimental assembly.

22. Describe the administrative organization of the radiation protection program, including the authority and responsibility of each position identified.

The administration organization of the Office of Radiation, Chemical, and Biological safety is listed below as a flow chart of authority.

Director, and Safety Officer

↓

Health Physicist

↓

Radiation Monitor

The authority and responsibility of each position is detailed as follows:

Radiation Safety Officer

Evaluate exposure records and determine required changes in procedures, equipment, and shielding to reduce hazards.

Serve as the University's radiological safety officer.

Determine that radiation usage is in compliance with the U.S. Nuclear Regulatory Commission and the Michigan Department of Health regulations.

Approve all procedures and changes in University policies that may involve personnel radiation exposure.

Supervise all aspects of University radiation protection and measurement activities such as personnel monitoring, x-ray monitoring, maintenance of exposure records, and sealed source leak testing as required by the U.S. Nuclear Regulatory Commission license and state regulations.

Inspect areas where radiation is used to identify radiological health and safety hazards.

Prepare and disseminate information on radiation safety practices and procedures to staff and students.

Develop and supervise an environmental radioactivity monitoring program for the University.

Supervise the radiation testing and monitoring laboratory.

Organize and train radiation monitoring teams which are responsible for radiation monitoring and control in the event of an emergency.

Act as liaison with University architects and engineers for the purpose of including radiologically safe construction in all new and remodeled buildings in which radiation usage is planned.

Consult with equipment manufacturers and suppliers regarding the design and installation of radiation facilities.

Develop and supervise a radioactive waste collection and disposal system.

Develop control systems and records of receipt, transport, testing, and disposal of radioactive materials.

Develop and administer program for obtaining U.S. Nuclear Regulatory Commission licenses and State of Michigan authorizations for the use of radiation facilities and supplies.

Participate in radiological health and safety programs and conferences.

Prepare and submit budget needs for the radiological health and safety functions.

Authorize the purchase of equipment and supplies.

Health Physicist

Assist in the development and coordination of a radiological health and safety program.

Assist in the development of an expanded bioassay program, covering a comprehensive range of frequently used radionuclides on-campus.

Coordinate nuclear emergency planning and procedures with appropriate University units.

Coordinate daily operational activities for control, receipt, monitoring, transport, testing, and disposal of radioactive materials.

Develop radiation monitoring procedures for University facilities, including medical and research.

Maintain liaison with computer personnel in the development and implementation of computerized systems/services.

Act as a resource authority regarding administrative unit policies and procedures.

Train support staff in new or revised work methods and procedures.

Interview and recommend employment of support staff.
Evaluate and review performance of support staff.

<u>Radiation Monitor</u>

Survey radiation personnel, facilities, and work areas to monitor radiation levels and detect radioactive contamination.

Distribute, collect, and record data from personnel monitoring equipment such as various film badges and dosimeters.

Calibrate radiation measuring instruments and develop new methods of calibration.

Advise University employees using radioactive materials, equipment, and areas.

Collect samples and analyze and maintain records on air, water, and population radiation exposure.

Inspect incoming radioisotope shipments and notify delivering carrier and U.S. Nuclear Regulatory Commission of contamination.

Perform bioassays to determine internal disposition of radioactive compounds and institute corrective measures when necessary.

Conduct training and lecture programs for radiation workers.

Maintain up-to-date knowledge of U.S. Nuclear Regulatory Commission and other regulations.

Serve on the Radiation "On Call" lists for campus radiation emergencies.

Prepare and maintain appropriate computer and other records.

23. Describe any radiation protection training for the non-Health Physics staff.

A slide presentation using a narrative cassette is used by all police and fire fighting personnel to inform them of emergency procedures. A visit to the reactor facility by the same personnel is used to familiarize them with the area. Slides, videotape and lectures are available to them and other non-emergency staff for general radiation training. The presentations are one hour in length to allow for questions, answers and general discussion after the planned presentation. On going training, specifically for police and fire departments includes an annual tour of the reactor lab.

24. Summarize your general radiation safety procedures. Identify the minimum frequency of surveys, action points (levels), and appropriate responses.

Radiation surveys are done as needed on an emergency basis. Regular radiation surveys take place every other week. Action is required on any area that registers two times background-action varies from complete clean-up to explanation of higher reading (i.e., sample storage area).

25. Describe your program to ensure that personnel radiation exposure and releases of radioactive material are maintained at a level that is "as low as reasonably achievable" (ALARA). Identify steps taken to implement the ALARA principle.

Routine bi-weekly surveys are conducted of the reactor area, control room and associated labs to ensure there is no spread of contamination. Area and particulate monitors as well as environmental monitors have been installed. Personnel film badges are changed and read biweekly. The reactor is shielded to minimize exposure in the office, control room and surrounding passages.

26. Describe any gaseous and air particulate effluent sampling equipment with respect to sampling location, sampling rates, and probe geometry.

Argon-41 monitoring System

probe geometry: side window G-M-Tube
location: Probe - in room 198
monitor readout - console, room 184
stack flow: 700 CFM
sample rate: several cfm

Air particulate monitoring system

probe geometry: end window - G-M-Tube
location: probe - room 184
monitor readout - console, room 184
sample rate 2.5 cfm

27. For the fixed-position radiation and effluent monitors, describe the generic types of detector and their efficiencies and operable ranges, also specify the methods and frequency of instrument calibrations and routine operational checks.

Monitor	Type of Detector	Ranges	Efficiency Calibration and Frequency
Area	Ion chamber	0.1 to 10,000 MR/HR	Known source (Co^{60}), annual
Air-particulate	G-M Tube	20 to 200,000 counts/min	NBS standard source, annual
Argon-41	G-M Tube	1 to 999,000 counts/30 sec	known source (Ar-41), annual
Criticality	Ion chamber	1 to 15 mR/HR	known source (CS-137) annual

Operational Checks

Type of Check	Frequency	Monitor(s)
Alarm check	daily check list	air and area
Source calibration	daily check list	area
H.V. operation check	daily check list	air
Background count check	daily check list	Ar-41, air, area
Alarm check, source cal.	weekly check list	criticality

28. For the radiation monitors that are alarmed, specify the alarm set points and indicate the required staff response to each alarm.

Alarm	Set Point	Response (for all alarms)
Air particulate	10 K cmp	1. Scram reactor
Area	100 mR/hour	2. shut off primary pump
Criticality	15 mR/hour	3. pull building evacuation alarm
		4. evacuate room, locking doors
		5. meet members of the DPS, Fire Department and Reactor Safety Committee at the northwest corner of the Engineering Building for re-entry procedure.

29. Identify the generic type, number, and operable range of each of the portable Health Physics instruments routinely available at the reactor installation. Specify the methods and frequency of calibration.

Type	Quantity	Range	Calibration & Frequency
G-M Tube pulse rate meter	4	5 to 500,000 counts/min	NBS Source - semi annually
Ion chamber	2	0.1 to 199,000 MR/HR and 0.1 to 100,000 MR/Hr	Known source (CS 137) semi-annually

30. Describe your personnel monitoring program.

Film badges are the only in-house personnel monitoring system. They are changed bi-weekly and supplied by R.S. Landauer Co., Chicago, IL.

31. Provide a summary of the reactor facility's annual personnel exposures [the number of persons receiving a total annual exposure within the designated exposure ranges, similar to the report described in 10 CFR 20.407(b)] for the last 5 yr of operation.

	Number of Individuals in Each Range					
Whole-Body Exposure Range (Rems)	1979	1980	1981	1982	1983	Total
No measureable exposure	79	49	125	57	69	375
Less than 0.1	1	8	1	2	0	10
0.1 to 0.2	0	7	0	0	0	7
0.2 to 0.3	0	1	0	0	0	1
more than 0.3	0	0	0	0	0	0

32. Describe your environmental monitoring program; summarize the results for past 10 yr and compare recent measurements with any performed before the initial reactor criticality. Provide your analysis of the environmental monitoring results.

Environmental Radioactivity Data

Environmental radioactivity data have been collected on a continuing basis for the past several years. Included are air, water and population exposure measurements. The latter measurements are obtained by the use of a new calcium sulfide thermoluminescent (TLD) dosimetry system which enables precise determination of doses for comparison with governmental control reference data. The long term record enables an evaluation of contributions of radioactive material releases by campus nuclear facilities. The university is required to estimate the population dose resulting from nuclear facilities for which it is licensed.

Summary of past 10 years:

Year	Airborne Particulate (μCi/cc)	Water (μCi/cc) gross α	gross β	Quarterly Average[1] TLD (mR)	Quarterly Average Background TLD (mR)
1983	5.58E-15	1.36E-10	1.46E-8	35.28	23.52
1982	1.42E-15	2.08E-10	2.35E-8	42.51	21.12
1981	2.56E-15	2.70E-10	2.50E-8	34.48	20.62
1980	3.61E-15	3.68E-10	4.34E-8	29.35	20.78
1979	2.37E-15	4.50E-10	3.45E-7	45.92	16.26
1978	1.91E-14	1.8 E-10	3.14E-7	28.56	14.28
1977	5.77E-13	1.15E-11	1.12E-8	27.14	19.97
1976	1.76E.13	4.08E-11	2.54E-8	9.85	12.11
1975	9.18E-15	6.08E-11	4.26E-8	13.60	16.55
1974	4.06E-14	3.44E-11	2.61E-8	11.48	11.10
1973	1.74E-14	4.3 E-11	3.4 E-7	7.60	6.48

The date of initial reactor criticality is: March, 1969.

In 1976, the Office of Radiation, Chemical, and Biological Safety re-calibrated and repaired the TLD measuring equipment which explains the sudden rise in TLD readings.

33. Comment on the ability of the reactor components and systems to continue to operate safely and withstand prolonged use over the term of the requested license renewal. Include the potential effects of aging on fuel elements, instrumentation, and safety systems.

Reactor components have withstood approximately 20 years of exposure with little failure. The thermocouples in the fuel typically fail after several years and this necessitates the installation of backup thermocouples or the replacement of instrumented fuel assemblies. The distortion of fuel has been minimal at the present operating conditions both at MSU and at other TRIGA facilities. Reactor instrumentation is relatively old and experiences some component failures. The design of reactor instrumentation is such that components can be readily removed and replaced. Redundancy in safety systems assures that a system safety

[1] Includes background. Measured at reactor exhaust stack.

is maintained even with component failure. Mechanical safety systems have experienced essentially no failure during the MSU experience and it is believed that continued operation will be possible.

34. What is the measured temperature in a B-ring fuel element immediately following a 1.4% $\Delta k/k$ (2.00$) pulse? What is the maximum measured temperature in the element and how long after the pulse does this maximum temperature occur?

On January 26, 1984 we pulsed the reactor with a $1.98 reactivity insertion (1.39% $\Delta k/k$). The fuel temperature meter showed temperature of just over 250°C. The temperature was sustained for approximately 2 seconds.

35. What is β effective for the MSTR current core configuration?

β effective is 0.007

36. In the steady-state mode, why is the interlock that prevents the application of air to the safety transient rod unless the "safety transient rod cylinder" is full "in" not listed in the proposed Technical Specifications? Justify.

In the steady state mode, the transient rod cannot be withdrawn unless the reg & shim rods are fully "in." Thus the reactor is subcritical. Since the maximum reactivity insertion by the transient rod occurs when the rod is withdrawn from the full "in" position, any reactivity insertion from a partially withdrawn position will be less than the situation normally encountered (i.e. startup with transient rod fully "in.") Thus, no interlock is necessary.

37. The proposed Technical Specifications state the control rod drop time for the slowest rod as 2 s. The old Technical Specifications give a rod drop time of less than 1 s (except for the pulse rod). Justify this longer time interval.

The proposed 2 second time is a simplification of the original requirement of 1 second for Reg and Shim rods or 2 seconds for pulse rods. The pulse rod has always been the slowest one and this was defined as the limiting case. We will revert to the 1 and 2 second base as requested. There has been no problem with compliance under the current license.

38. What are the actual numbers of 8.5 and 12 weight-per fuel elements loaded into the core? Where are they positioned? What administrative limits or requirements are placed on fuel loadings?

There are eight 12% by weight fuel elements in the core. Six in the B-ring and one in the C-9 position and one in the C-11 position. The other 62 elements in the core are 8.5% by weight.

The administrative requirements of fuel loading are as follows:

1. All core loading must be done under the supervision of a licensed Senior Reactor Operator.

2. During core loading, the normal procedures of reactor operation must be observed, including:
   a. Daily and weekly check lists must be completed with no abnormal circumstances.
   b. Room must be isolated (doors and windows closed).

3. The pulse and shim rods must be in an up (or cocked) condition during critical loading or movement of more than one element to serve as a safety system.

4. A licensed reactor operator must be at the console during core loading to determine critical core configuration when more than one element is being moved.

5. No aluminum clad fuel will be loaded into the core without prior safety analysis written and approved by the Reactor Safety Committee and the NRC.

6. There are no special restrictions on the location of 12 wt.% fuel in the core.

39. What is the annual release rate of ^{41}Ar from the reactor facility to the environment? What is the normal ^{41}Ar concentration in the experimental areas?

The annual release rate varies from year to year depending on the amount of time the reactor is running. A typical release rate is 400 µCi/year. The "normal" A^{41} concentration in the reactor room when operating at full power for extended time is approximately 4×10^{-8} µCi/cc.

40. Specify assumptions and provide calculations of the expected radiation levels at the following locations.
   a. Beyond the limits of the reactor facilities (as a result of all airborne releases of radioactive materials from the reactor facility)
   b. In the reactor room (as a result of ^{41}Ar concentrations when such concentrations are a maximum)
   c. Above the reactor pool (as a result of the maximum ^{16}N releases possible)
   d. In the reactor room (because of continuous exposure to ^{16}N and ^{41}Ar for some specified period)

(a) Typical A^{41} release rate = 400 µCi/yr = 20 µCi/Full Power Operation hr

$$A^{41} \text{ conc, then} = \frac{20}{700 \times 60 \times 7.49 \times 3785} = 1.6 \times 10^{-8} \ \mu\text{Ci/cc}$$

= 40% of MPC to unrestricted area

If MPC = 500 mr/yr or $\frac{500}{8760}$ = 0.09 mr/hr

Then dose rate to public = 0.4 x 0.09 = 0.036 mr/hr

(b) Max A^{41} conc = 2×10^{-6} (insurance limit)
= 5000 mr/2080 hrs = 2.5 mr/hr

(c) N-16 in pool water shielded from operating area by 10 ft of stagnant water.

N-16 release rate in outgassing: (assume same outgas rate as A^{41})

Atom Ratios A/O/N = 1/40/158 in dissolved air

Fast ϕ = Thermal ϕ

$$\frac{\text{rate of prodn } N^{16}}{\text{rate of prodn } A^{41}} = \left(\frac{40}{1}\right)\left(\frac{0.017 \times 10^{-3} \text{ barns}}{0.62 \text{ barns}}\right) = 1.1 \times 10^{-3}$$

$$\frac{\text{rate of decay } N^{16}}{\text{rate of decay } A^{41}} = \frac{T_{1/2} \text{ of } A^{41}}{T_{1/2} \text{ of } N^{16}} = \frac{1.83 \times 3600}{7.13} = 924$$

∴ Emmission rate of N16 ≈ that of A41 if outgassing occurs without delay. (delay likely and will reduce N16 due to decay)

∴ Max release rate $N^{16} = 1.6 \times 10^{-8}$ μCi/cc air

This is 53% of MPC to uncontrolled areas and equivalent to 0.048 mr/hr

(d) Sum of $A^{41} + N^{16} = 0.036 + 0.048 = 0.084$ mr/hr

41. Provide justifications for not having a reactor pool water level alarm.

Loss of pool water very unlikely due to facility construction and location. Small (few feet) loss of pool water would not generate safety hazard. Radiation monitors over pool give ample warning of radiation hazard.

42. What controls are placed on the spare aluminum elements currently located in the reactor pool storage racks surrounding the reactor core to prevent them from accidentally being loaded into the core?

Administrative record keeping to assure that proper fuel loading is being performed. All fuel elements are numbered for identification.

43. Provide a block diagram of the nuclear instruments control and set points. It should include the following information.
(1) Type of channel
(2) Detector
(3) Range of detector (not meter readout)
(4) Automatic controls
(5) Set points (minimum and maximum where applicable)

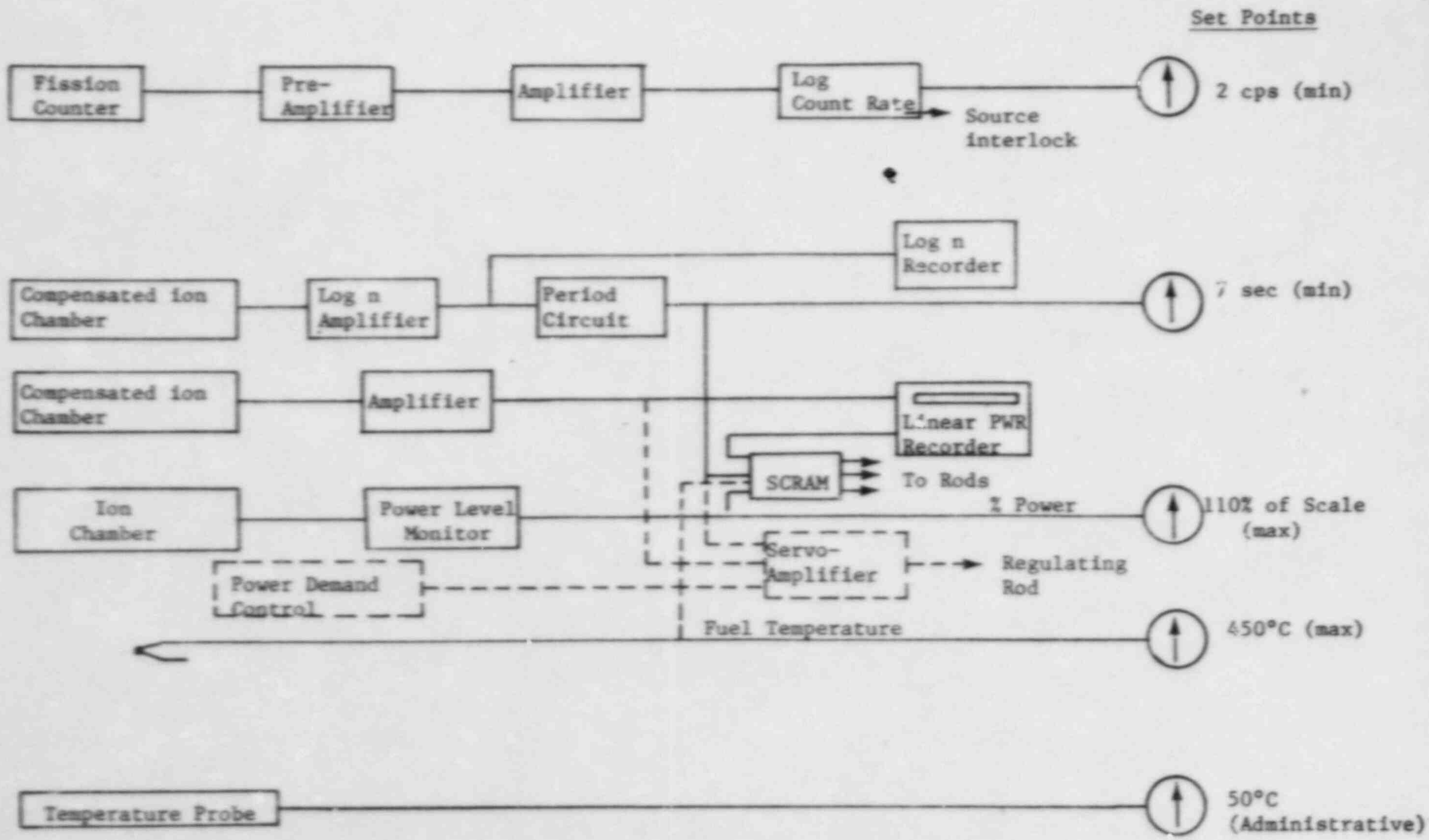

Figure 5A. Block Diagram of Reactor Instrumentation for Steady-State Operation

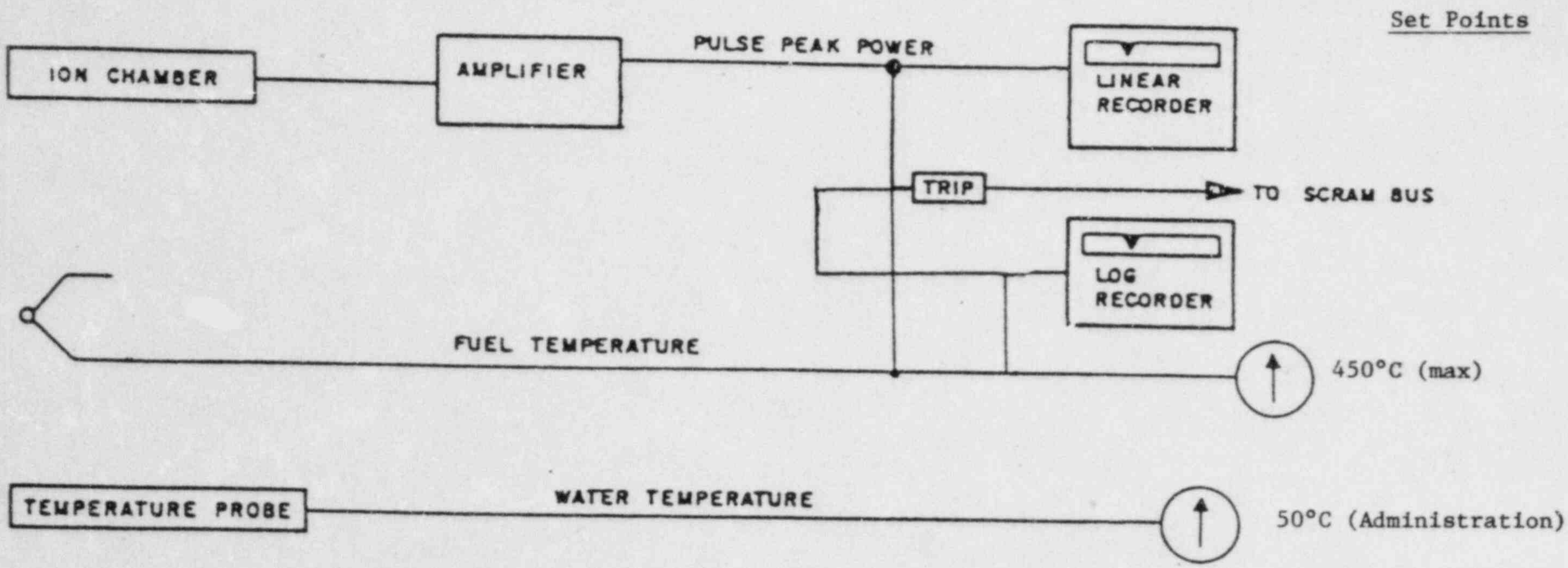

Figure 5B. Block Diagram of Reactor Instrumentation for Pulsing Operation

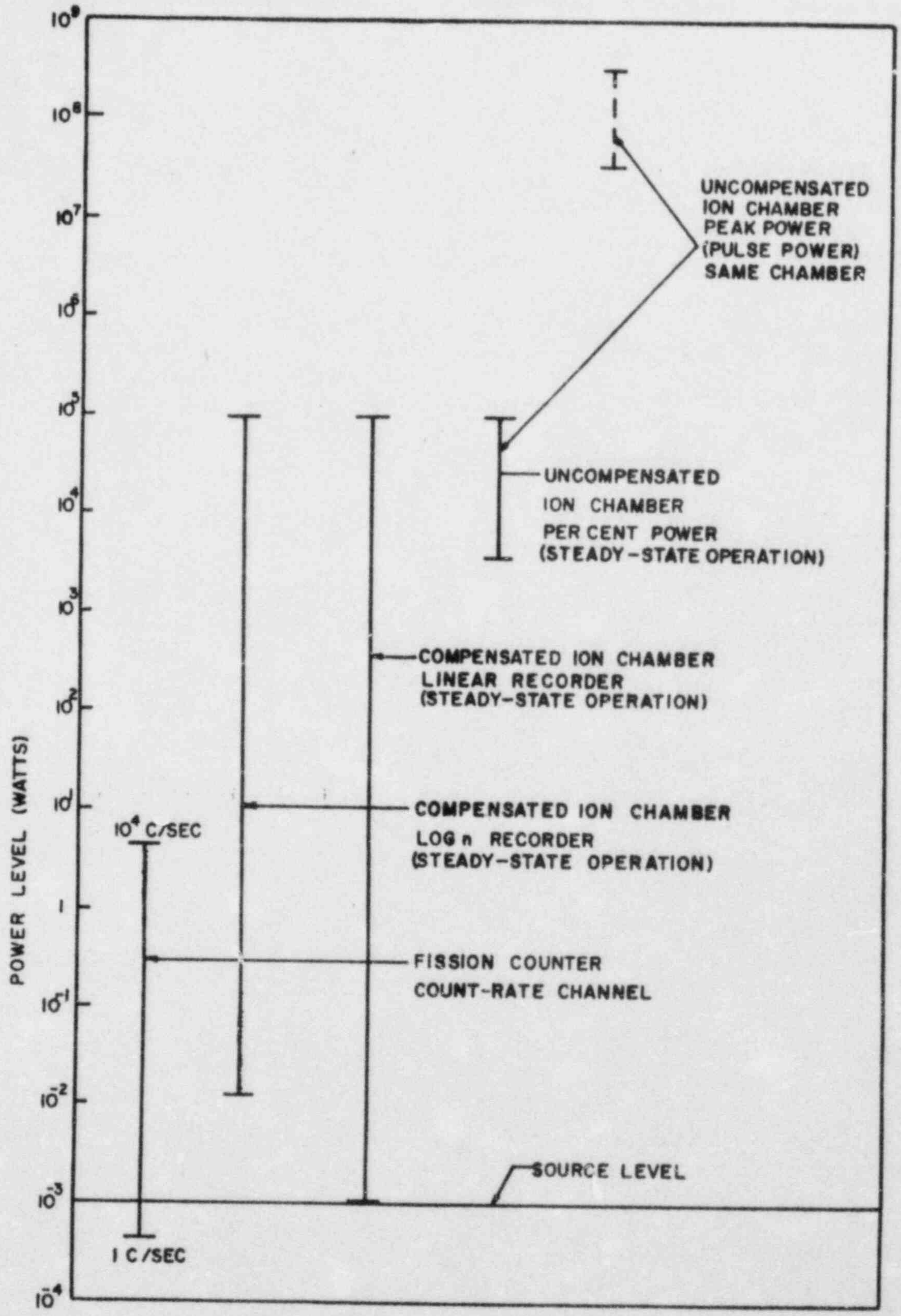

Figure 6. Detector Ranges

Figure 5A + B are block diagrams of the instrumentation during steady state operation and pulsing respectively.

Figure 6 is a graph of the detector ranges for the MSTR.

44. Provide a bar chart drawing of the operating ranges of the in-core nuclear detectors.

    Figure 6 is a bar chart of the operating ranges of the detectors.

45. Please address the following accidents; include all assumptions made, calculative methods used, and accident scenarios.
    (1) Loss-of-coolant accident
    (2) Fuel element failure in air
    (3) Rapid insertion of reactivity (step nuclear excursion)
    (4) Mechanical rearrangement of fuel (consequences of dropping 600-lb lead cask into core).
    (See attached Safety Analyses p. 34-48)

46. What are the flow rates of the coolant through the heat exchanger system and the demineralizer loop?

    The present flow rate of the primary water through the heat exchanger is 150 gallons per minute. Flow through the deminerlizer loop is 3 gallons per minute.

47. Describe the ^{16}N diffuser system at the MSTR.

    The design of the primary water return from the heat exchanger provides for ^{16}N diffusion. The cold water return is directed across the top of the core. The primary cooling system is always on during normal operation.

48. What is the current use of the storage vault and storage pipes in the pit next to the reactor in room 184?

    The storage pipes are not used at this time but would provide emergency storage of fuel elements. The vault is used to store a small amount of tritiated water. It provides extra radioactive sample storage space.

49. Describe any equipment and services shared by the reactor facility with other parts of the University.

    Facilities shared with other parts of the building include:

    1. Pressurized air - building
    2. Electricity - line power - campus
    3. Steam - for heat - campus
    4. Water - to sinks - campus

    There is no gas service to room 184.

50. Describe the primary power supply to the campus and to the reactor facility.

Power plant no. 65 is owned and operated by MSU. It provides power and steam to the University. It has a 40 MW_e capacity and generates 850,000 lbs of steam/hour. The plant is on intertie with Consumer's Power Co. with which it exchanges power during peak demand periods.

51. Provide detailed geographic data for the site, including location with respect to area features, nearby industry, transportation, military and manufacturing facilities.

The following is a list of major geographic points and their direction and distance from the reactor lab. Note that East and South of the reactor lab is mostly farm land.

I-496: 1 1/2 miles West
I-96: 2 1/2 miles South
I-69: 1 1/2 miles North

Red Cedar River: 1400' (North)
East Lansing downtown area: 4200' (North)
Grand River junction: 4 miles (West)
Lansing downtown area: 4 miles (West)
Major industrial (Oldsmobile): 4 miles (West)
C&O railroad: 3500' (South)
Grand Trunk railroad: (leased to passenger) = 1400' (South)
Lansing Capitol Airport: 7 1/2 miles (North-West)
Abrams airport - Air National Guard post: 16 1/2 miles (North-West)
State Capitol: 4 miles (West)

52. Discuss the ability of the Engineering Building to withstand the most severe credible tornado. What would be the effects of tornado damage on the reactor room and on any safety-related equipment?

The Engineering Building was constructed to withstand winds to 88 mph. Data received from T.T. Fujeta at the University of Chicago indicates that although East Lansing is in the southern half of lower Michigan, an area of high tornado frequency, East Lansing has been historically free of tornados. From data collected between 1930 and 1974 all tornados passed North and West or South and East of East Lansing. East Lansing is on the Northern edge of a pocket of relative calm. The storms to the North and West have all been of the F_2orF_3 intensity on the Fujeta scale (113 to 157 mph). Only one tornado of the F_4 or F_5 intensity was reported between 1930 and 1974. This was about 10 miles North of the reactor site. It is estimated that, in the case of several (> 88 mph) winds, the non-supporting curtain walls of the building would collapse but the floor/ceiling structure of the building would survive. Damage to the below ground reactor core would be limited to debris falling into the pool. Since the building is not designed as a containment structure for the reactor, loss of curtain walls would not be serious. Monitoring for radioactivity would be done by battery-operated instruments since electric power would certainly be disrupted.

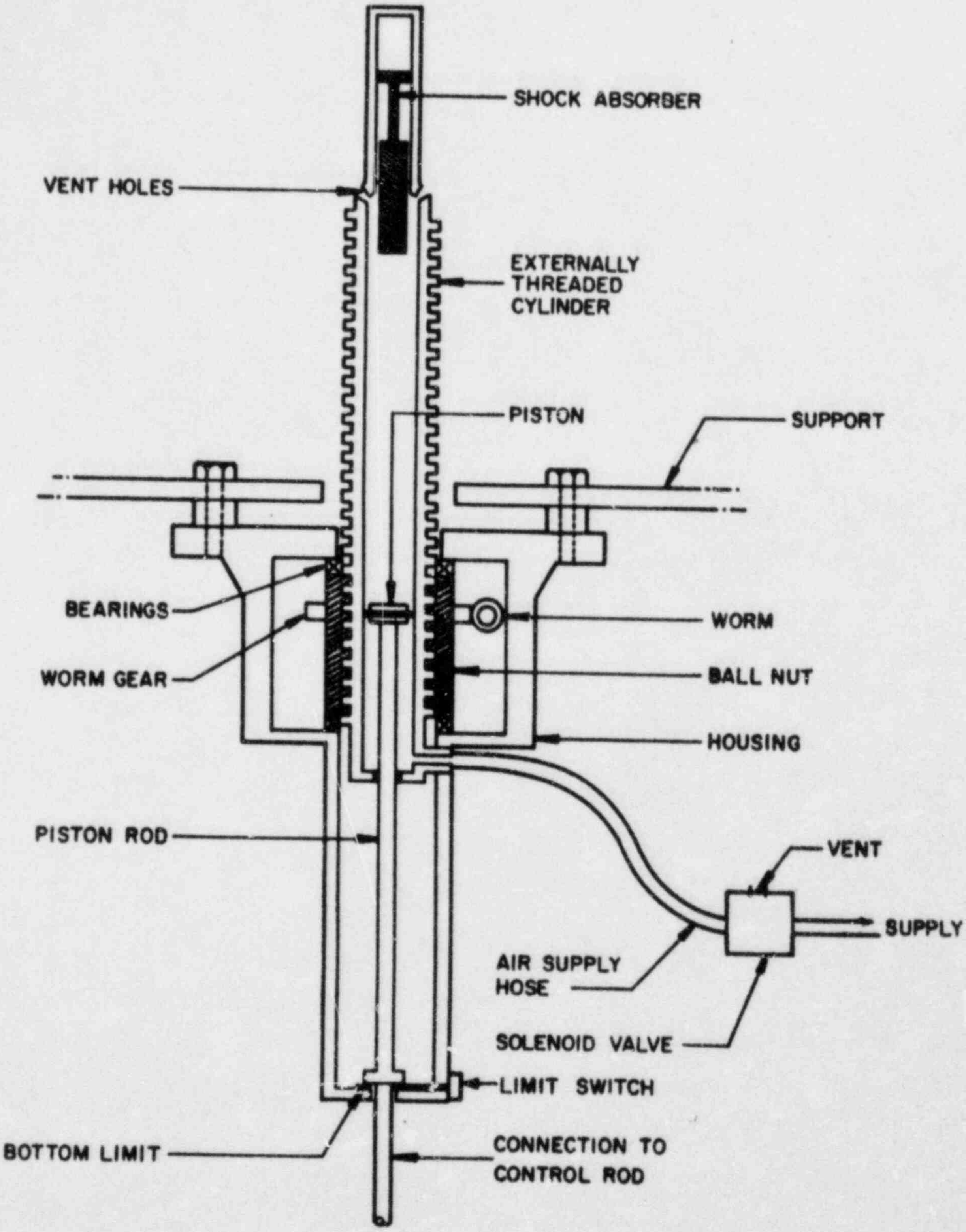

Fig. 7. Transient-rod drive mechanism

53. Provide more detailed information on the geology of the site and surrounding areas. Are there any faults in the area? What is the history of seismic activity in the area?

The North Western corner of Ingham county lies near the center of the Michiganian basin. This is an interacratonic basin developed on precambrian crust. The basin is filled with primarily paleozoic sedimentary rocks approximately 15,000 feet thick.

In the Lansing area, Bedrock, which consists of shales and sandstone of the Saginaw formation and Pennsylvanian age, are covered with a layer of glacial drift approximately 75 feet thick. Bedrock is essentially underformed and faults do not off set younger formations.

There is one recordable event in recent history. August 9, 1947 in Coldwater, Michigan on earthquake occurred. It could be felt in Lansing at intensity IV and is the largest event reported in Michigan. This event occurred along a fault line in the Coldwater area. It is estimated that this fault line could produce earthquakes of magnitude IV. Coldwater is approximately 50 miles from East Lansing.

54. Describe in detail the transient rod pneumatic drive system and the system operation in the pulse, scram and manual modes. Address the absence of an electromagnet.

Figure 7 details the rod drive mechanism for the pulse rod. Operation of the pulse rod is controlled from the console.

The drive is mounted on a steel frame that bolts to the center channel cover plate. Two steel covers keep the mechanism clean. From zero to a maximum of 15 inches of rod may be withdrawn from the core; however, administrative control is exercised to restrict the travel so as not to exceed the maximum permissible step insertion of reactivity (2.00 or 1.4% $\Delta k/k$).

The transient-rod drive is a single-acting peneumatic cylinder with its piston attached to the transient rod through a connecting rod assembly. The piston rod passes through an air seal at the lower end of the cylinder. Compressed air at 75 pounds per square inch is supplied to the lower end of the cylinder from an accumulator tank mounted beneath the center channels when a three-way solenoid valve located in the piping between the accumulator and cylinder is energized. The compressed air drives the piston upward in the cylinder and causes the rapid withdrawal of the transient rod from the core. As the piston rises, the air trapped above it is pushed out through vents at the upper end of the cylinder. At the end of its travel the piston strikes the anvil of an oil-filled hydraulic shock absorber, which has a spring return, and which decelerates the piston at a controlled rate over its last inch of travel. When the solenoid is de-energized, the valve cuts off the compressed air supply and relieves the pressure in the cylinder, thus allowing the piston to drop by gravity to its original position with the transient rod fully inserted in the reactor core. No electromagnet is present to hold the rod in position as in the reg and shim rods. Instead pressurized air holds the transient rod up. The operator reads solenoid position at the console, not electromagnet contact.

Provision is made by raising or lowering the cylinder, thereby controlling the distance the piston travels. This adjustment determines how far the transient rod is withdrawn from the reactor core, and thus determines the amount of excess reactivity inserted. The cylinder has external threads running most of its length, which engage a series of ball bearings contained in a ball nut mounted in the drive housing. As the ball nut is manually rotated by a crank-driven worm gear, the cylinder, which is prevented from rotating, moves up or down, depending on the direction the crank is turned. An extension of the crank drives a mechanical revolution counter that indicates the position of the cylinder and also indicates the distance the transient rod will be ejected from the reactor core. As soon as the cylinder has been positioned, the crank is removed from the mechanism and placed in storage to prevent unauthorized operation.

Attached to, and extending downward from the drive housing, is the rod guide support, which serves several purposes. A bar attached to 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 damper 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 flanged connector engages the actuating level of the microswitch, and indicates on the instrument console that the rod is in the core.

Pulse and normal steady state operation is the same for pulse rod. Rod interlocks are provided in steady-state mode to limit upward travel of the pulse rod only when both reg and shim rod are in the fully inserted position.

All scrams apply to the pulse rod as do to the reg and shim in both pulse and steady state mode. However, in the pulse mode a 15 second timer will insert the pulse rod is well.

55. Provide a brief description of the Radioisotope Committee and the Office of Radiation, Chemical and Biological Safety, including membership and general responsibilities.

The Radioisotope Committee is an advisory committee to the University sanctioned to help the University meet federal and state regulations as regards radioactive materials. This committee gives approval to qualified individuals and their laboratories to use radioactive materials on campus. The committee also develops policy and guidelines for the university in the use, transport, receipt and shipping of radioactive materials. Current membership on the committee includes faculty members from the departments of Biochemistry, Cyclotron, Food

Sciences, Animal Science and Biology as well as the campus Radiation Safety Office, and Health Physicist. All of these people have had extensive experience with the use of radioactive materials. Committee membership is appointed by the MSU Provost.

The Office of Radiation, Chemical and Biological Safety (ORCBS) membership was discussed in Question 22. This office is the enforcement arm of the radioisotope committee. The office maintains the university broad license as well as the license from the State of Michigan to operate ionization equipment and cyclotrons. The office provides radiological expertise to the committee, as well as other data that will be helpful to the committee in granting approval to individuals and laboratories for the use of radioactive materials. ORCBS also provides health physics support to the reactor lab in the form of expertise and back-up emergency equipment.

56. Describe selection and training of reactor and senior reactor operators.

Nuclear reactor operators are personnel who have basic technical education (physical science or engineering) and an interest in nuclear reactor operations. Most of the operators are undergraduate students in engineering who are employed as part time student assistants.

Senior operators are generally people with at least a BS degree in a physical science or engineering who appear to have the qualifications and aptitude to serve in a more advanced position. Senior operators are generally full time employees of the University (rather than students).

Training of operators involves self study of suitable text material and one-on-one tutoring by a Licensed Operator or the Reactor Supervisor and the ORCBS. Wherever possible, formal training by MSU courses on Reactor Theory or in courses conducted at the University of Michigan is used to supplement the individualized instruction.

57. Please provide clear, reproducible MSTR figures for the following:

1. Reactor Facility Layout
2. Current Core Loading Diagram (See Question 2)
3. MSTR Fuel/Moderator Element Showing Construction and Dimensions
4. Cutaway View of MSTR Installation.
5. Schematic Drawing of Primary Coolant and Purification Systems (See Question 6)
6. MSTR Demineralizer Loop
7. Ventilation Bypass System (See Question 11)
8. Transient-rod Pneumatic Drive
9. Reactor Instrumentation Block Diagram for Steady-State Operations
10. Reactor Instrumentation Block Diagram for Pulsing Operations
11. Block Diagram of Nuclear Instrumentation and Setpoints (See Question 43)
12. Figure 6. Engineering Building Floor Plan (From 1967 SAR)
13. Figure 20. Block Diagram of Reactor Instrumentation (from 1967 corrected SAR)

14. Figure 25, Loading Diagram (from 1967 SAR)
15. Figure 1. East Lansing Census Tracts (from March 1984 SAR)
16. Figure 2. MSU Campus map (from March 1984 SAR)

See:

1. Figure 8
2. Figure 1
3. Figure 9
4. Figure 10
5. Figure 2
6. Figure 2
7. Figure 3
8. Figure 7
9. Figure 5A
10. Figure 5B
11. Figure 5A and B
12. Figure 11
13. Figure 5A
14. Figure 2
15. Enclosed 1 copy. Figure 12.
16. Enclosed 1 copy. Figure 13.

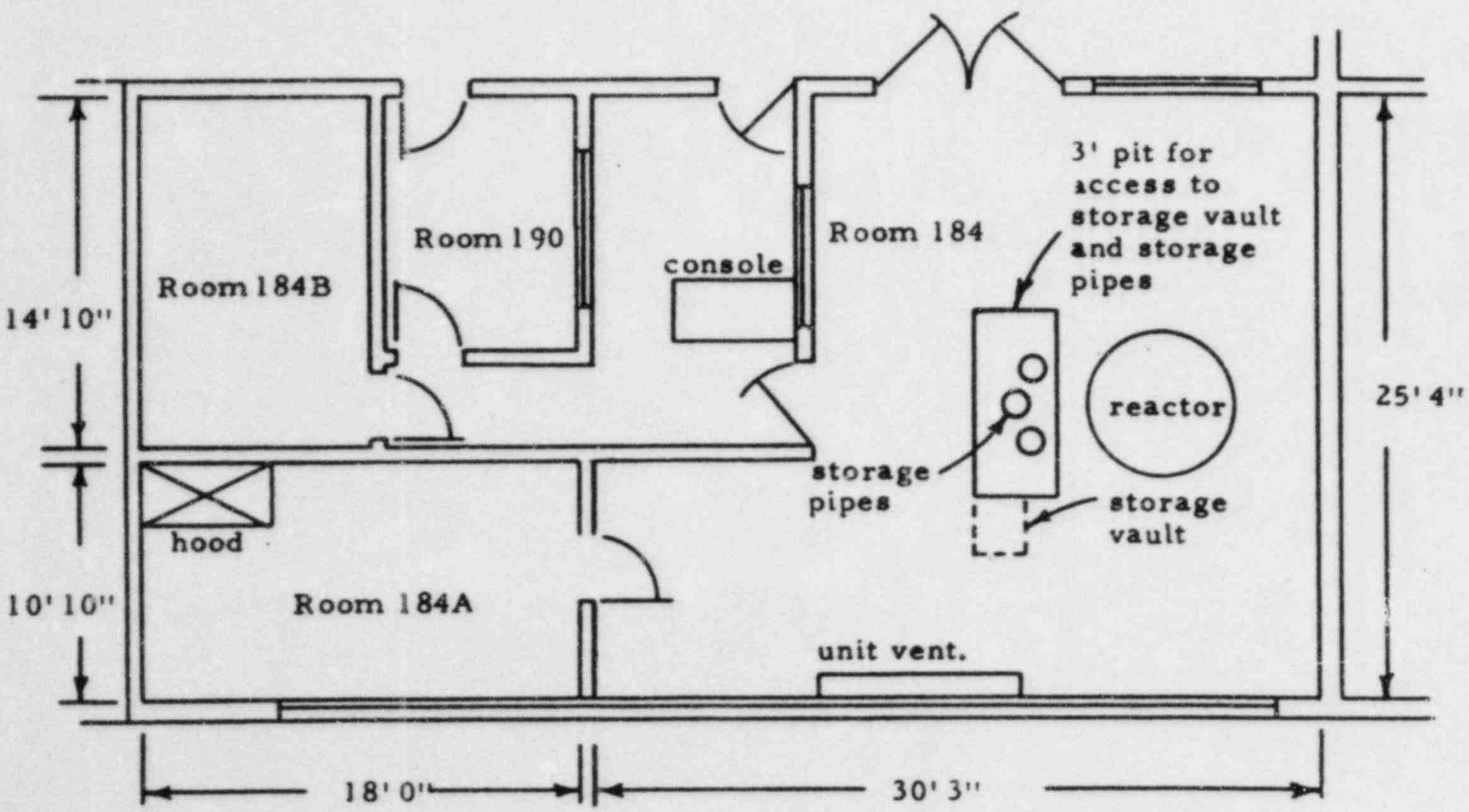

Figure 8.   Floor Plan for Reactor Room, scale 1/8" = 1 foot.

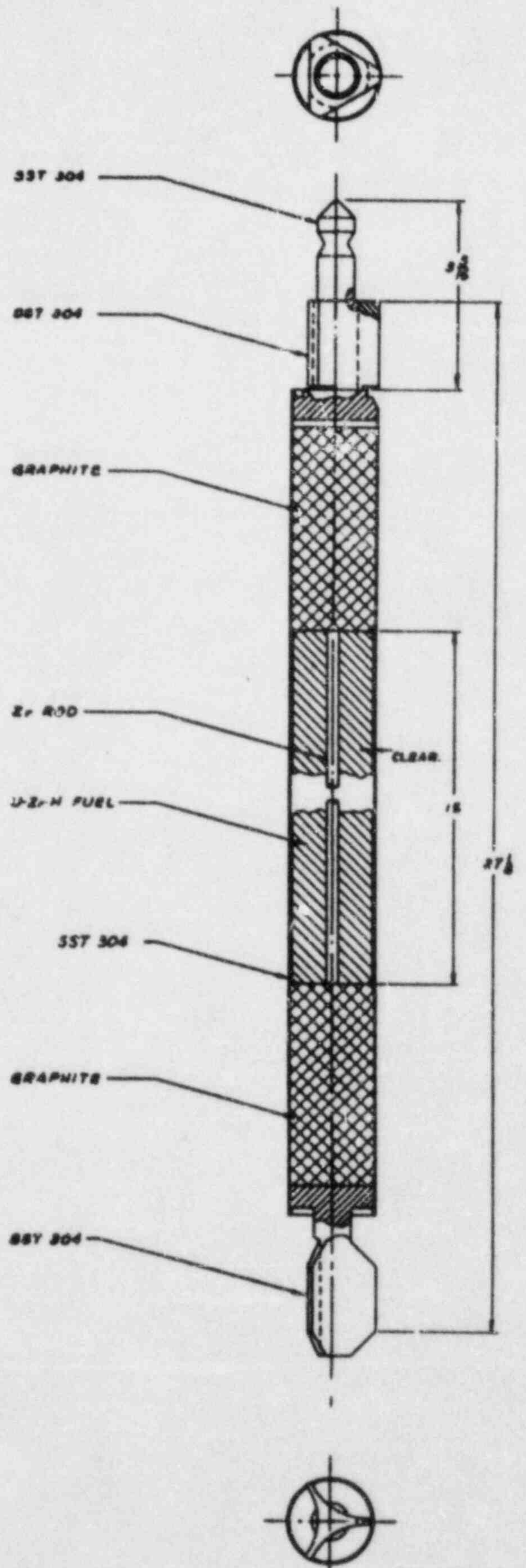

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Fig. 9. **Typical TRIGA fuel element**

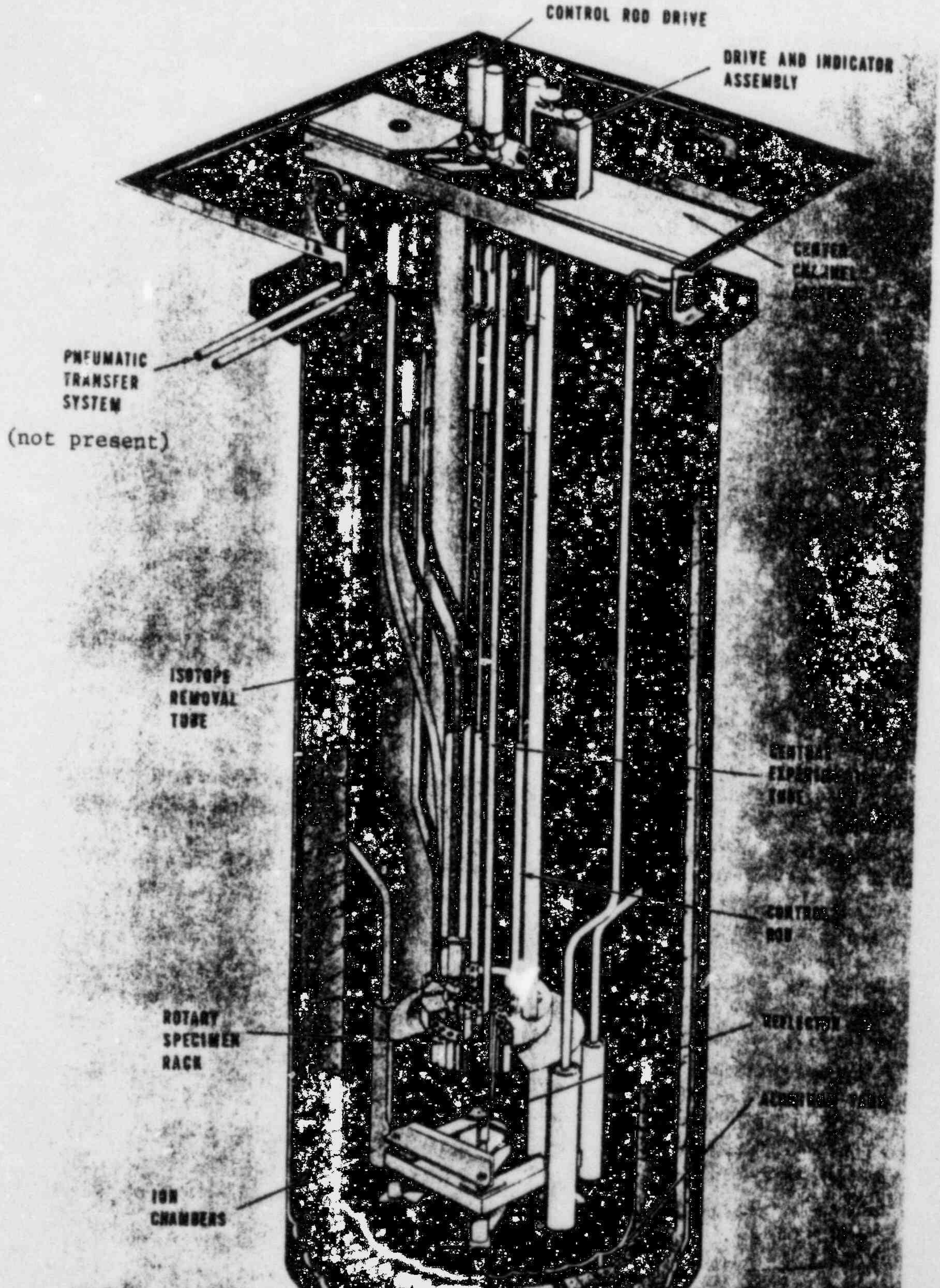


Figure 10. TRIGA Mark I reactor

MSU Engineering Building (Southeast Wing)

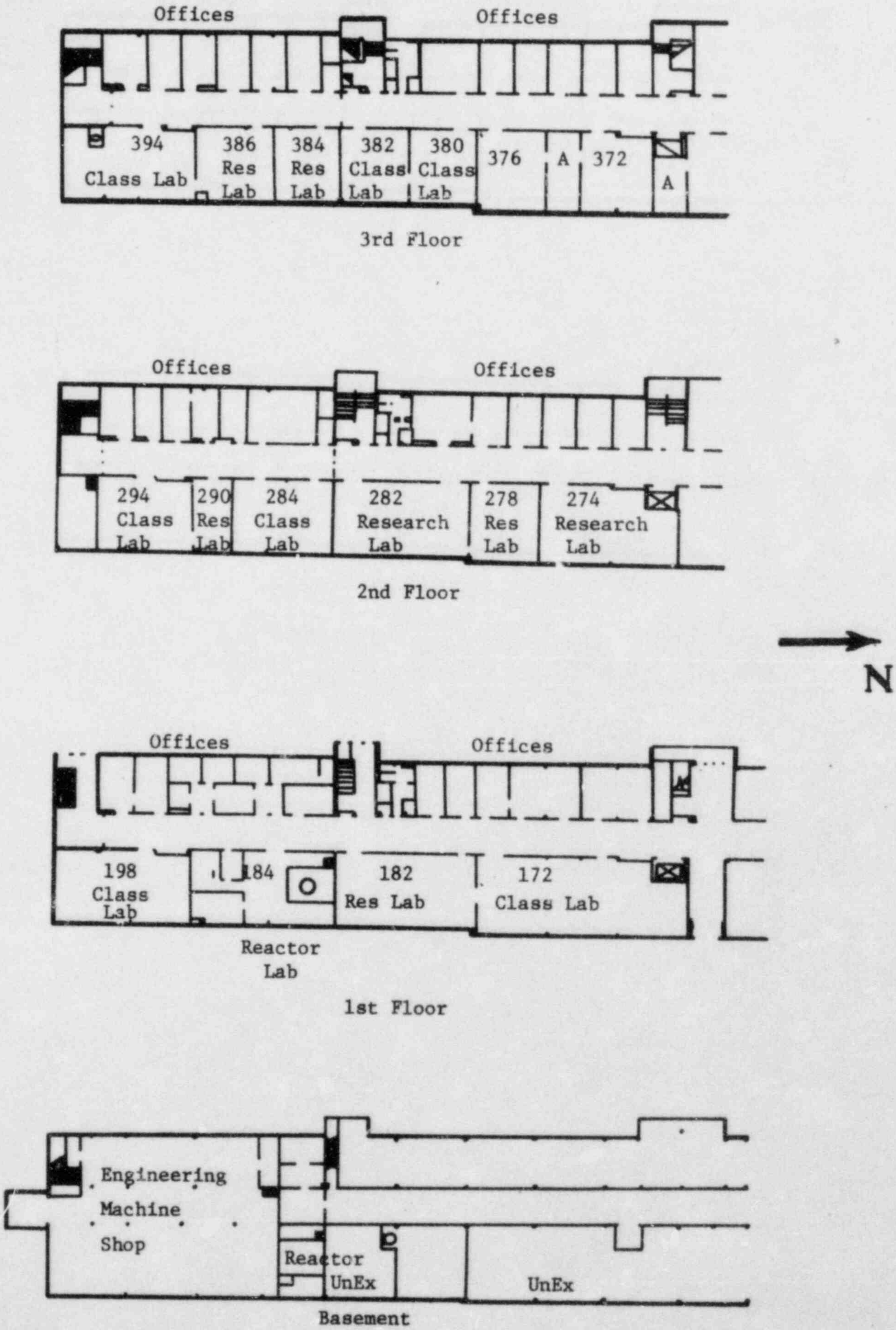


Figure 11.

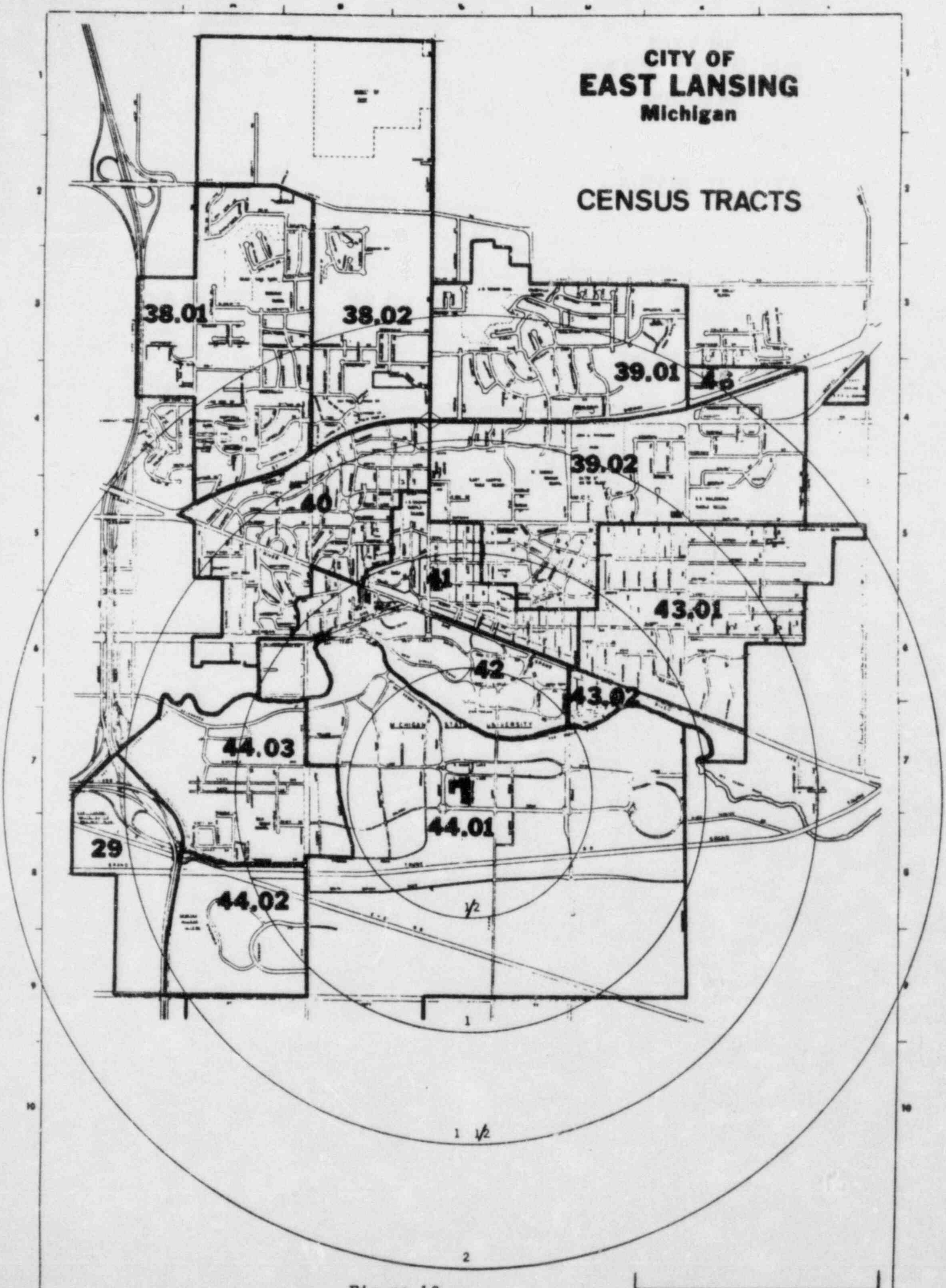

CITY OF
EAST LANSING
Michigan
CENSUS TRACTS
38.01
38.02
39.01
39.02
40
41
42
43.01
43.02
44.01
44.02
44.03
29
1/2
1
1 1/2
2
1 mile

Figure 12

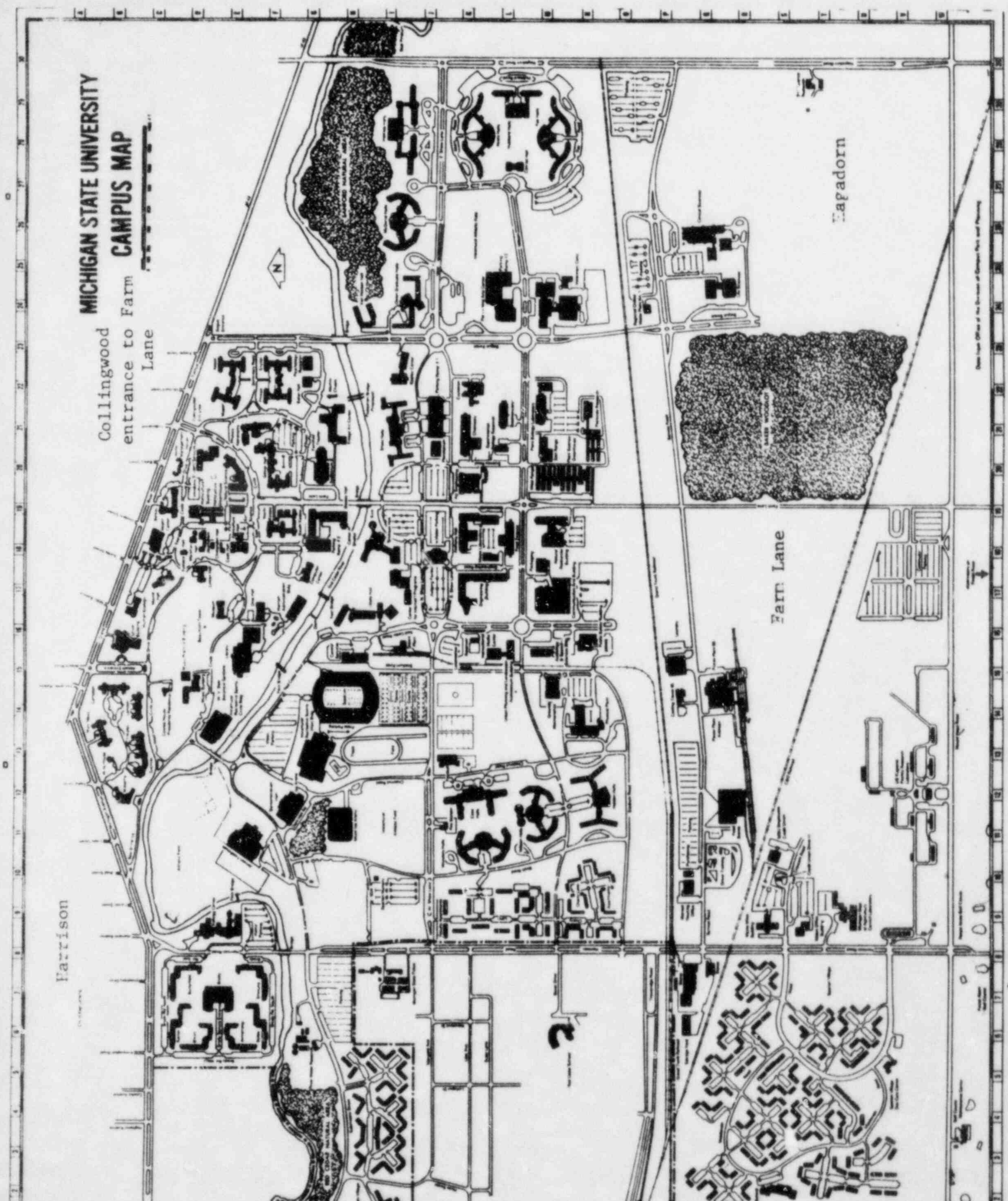

MICHIGAN STATE UNIVERSITY
CAMPUS MAP
Collingwood entrance to Farm Lane
Harrison
Farm Lane
Hagadorn

Figure 13

SAFETY ANALYSIS 45-1

LOSS OF COOLANT ACCIDENT

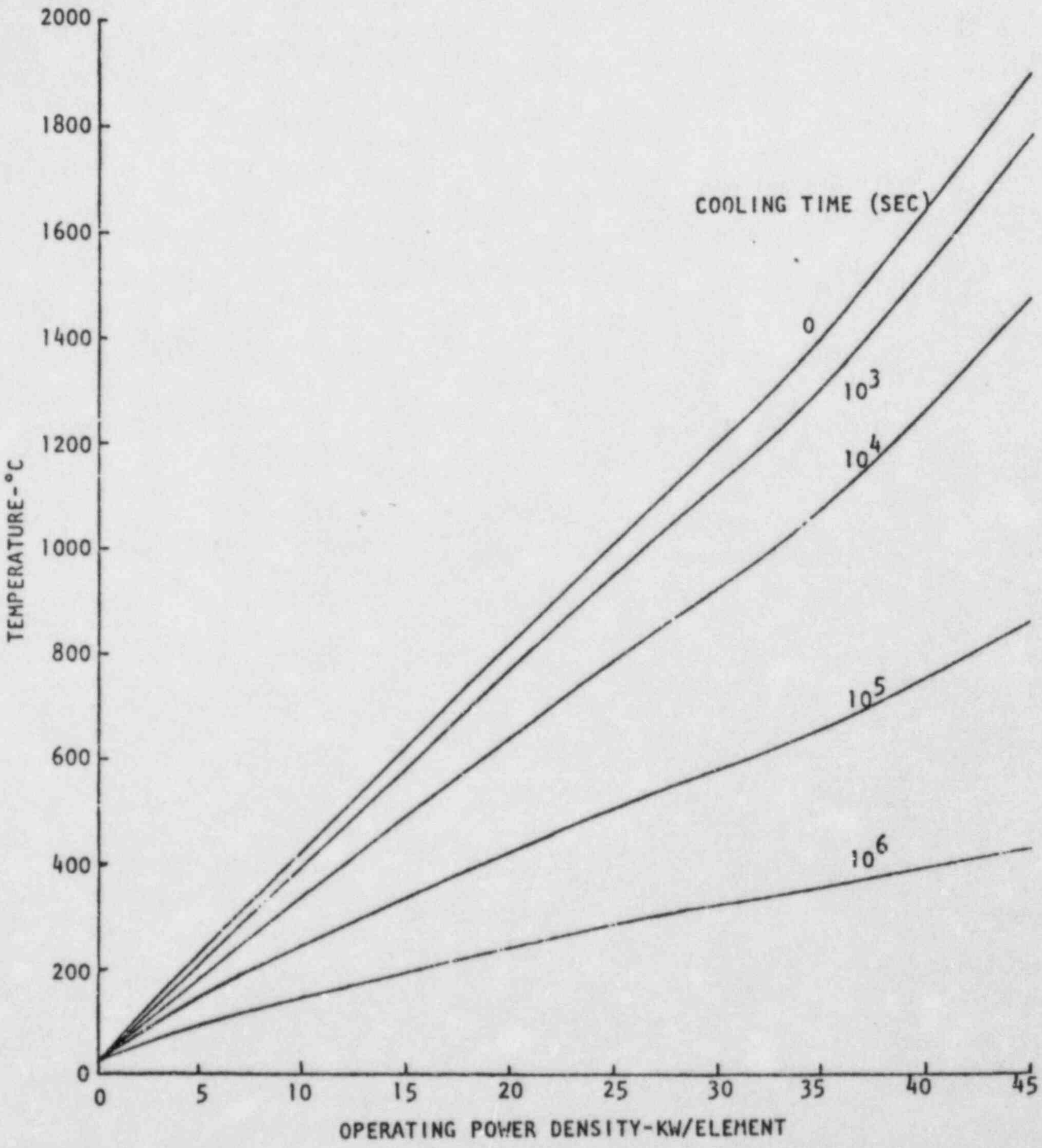
The MSTR operates at a maximum of 250 KW_{th} and the present core contains 70 elements. The average power density, thus, is 250/70 = 3.57 KW_{th} per element. A portion of the fuel contains 12 wt% uranium and, thus, is operated at a 41% higher power density than the remaining 8.5% uranium fuel. Thus, the maximum power density would be 3.57 x 1.41 or 5.03 KW_{th} per element. Under conditions speculated to involve the complete loss of power operation, it is estimated that the fuel temperature would reach approximately 225°C (see Fig. LCA-1). Under such conditions the estimated stress imposed on the cladding by internal gas is approximately 1200 psi (see Fig. LCA-2). This stress is considerably less than the approximately 37,000 psi yield stress for the stainless steel fuel cladding. It is, therefore, concluded that the loss of coolant would not result in the failure of the fuel cladding and that the fission product containment would not be lost.

Radiation Levels from Loss of Coolant

If it is assumed that the reactor water is suddenly lost, the radiation emitted from the contained fission products would potentially pose a threat to personnel in the reactor room and in Room 284 directly over the reactor. The operating floor of the MSTR is 20 feet above the reactor core and the estimated radiation levels from both direct radiation and scattered from the concrete room ceiling are given below for various decay times immediately following full power operation.

Calculated Radiation Dose Rates
For Loss of Reactor Pool Water

Time	Direct Radiation r/hr	Scattered Radiation r/hr
10 sec	2,500	0.65
1 day	300	0.075
1 week	130	0.035
1 month	35	0.01



EL-1872

Fig. LCA-1. Maximum fuel temperature versus power density after loss of coolant for various cooling times between reactor shutdown and coolant loss.

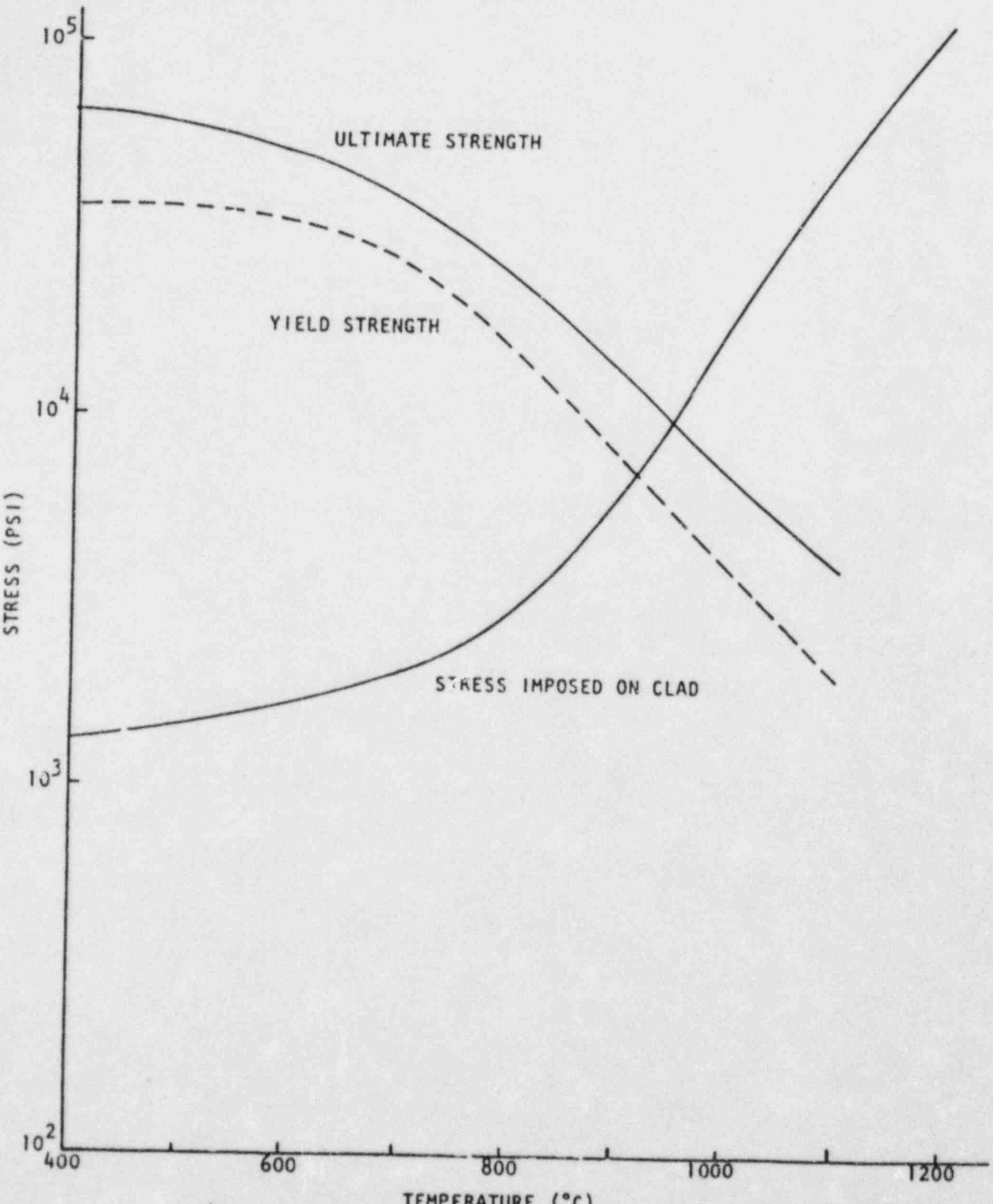


Fig. LCA-2. Strength and applied stress as a function of temperature, U-$ZrH_{1.65}$ fuel, fuel and clad at same temperature.

The table indicates that, except for the direct beam from the core radiation exposure in the reactor room would be high but tolerable even immediately after loss of coolant and that emergency operations could be carried out with limited time of action.

Of greater concern is the radiation levels in the uncontrolled area, Room 284, located directly over the reactor. The distance from the core to the occupied area of this room is 30 feet and the radiation levels calculated above (for 18 feet) will be reduced by inverse square $(18/30)^2 = 0.36x$ and the shielding of the concrete floor (approximately 6 inches) = 0.2x.

On this basis, the direct radiation in Room 284 is estimated to be

Time	Direct Radiation r/hr
10 sec	180
1 day	22
1 week	9
1 month	2.5

These levels of radiation are sufficiently high to necessitate immediate evacuation of the classroom in case of total loss of pool water.

The integrated exposure to personnel in this room as a function of time delay in evacuation is given below.

Time, Min.	Radiation Dose, R
5	14.97
15	44.77
30	89.07
45	132.9
60	176.3

Evacuation of Room 284 following loss of pool water during or shortly after reactor operation (the case assumed above) could most certainly be done in a time much less than an hour since a reactor operator on duty could assure building evacuation in such a catastrophe. If pool water were to be lost during non-working hours, a longer evacuation time might be experienced as the result of the time delay inherent in emergency response. The radiation levels would be reduced in this case however since the short lived fission products would have decayed away. As an example, assume a 6 hour shutdown period followed by loss of pool water and a 1 hour evacuation delay during which time Room 284 is occupied. The exposure to a person in such a situation would be 132 R. Such an exposure, although high, would not be life threatening.

SAFETY ANALYSIS 45-2

FUEL ELEMENT FAILURE IN AIR

In this analysis, it is assumed that one of the 12 wt% Uranium fuel elements is ruptured in air after a long (1,000 day) exposure to full power (250 kw_{th}) operation of the reactor. The radioactive material released to the air consists of 100% of the rare gases and halogens which, during reactor generation, have migrated to the gap between the fuel meat and cladding. No credit is assumed for deposition of the isotopes on walls, in pool or in any other manner.

(a) Exposure to reactor room occupants. In this case, the rare gases and halogens are assumed released to the reactor room, and the ventilation in the room is assumed to be zero. A room occupant is assumed to remain in the room for 10 minutes while evacuation takes place.

The fission product gas concentration in the core operated at 250 kw_{th} for 1,000 days is given below (from Lamarsh, p. 535--scaled to MSTR power output). (Note: core loading 70 elements, power density in 12 wt% fuel = 41% greater than core average.)

Radioisotope	Curies, core	Curies, Single Element, Avg.	Curies, Single Element 12 wt%
Kr^{85m}	2800	40	56.4
Kr^{85}	98	1.4	1.97
Kr^{87}	5,000	71.4	100.7
Xe^{133m}	400	5.71	8.06
Xe^{133}	14,320	204.6	288.4
Xe^{135m}	2,220	31.7	44.7
Xe^{135}	3,900	55.7	78.6
I^{131}	5,850	83.6	117.8
I^{132}	8,270	118.1	166.6
I^{133}	14,300	204.3	288.0
I^{134}	15,200	217.1	306.2
I^{135}	13,550	193.6	272.9

For fuel element ruptures in air, the gaseous fission product inventory in the gap is estimated to be 1.5×10^{-5} fraction of the total gaseous fission products (GA 4314). If all of this fraction is released into the reactor room (volume 2×10^{8}) with no air change rate, the concentration of radioisotopes in the room air is:

$$\text{Curies} \times 10^{6} \times 1.5 \times 10^{-5} / 2 \times 10^{8} = \mu\text{Ci/ml}$$

Assuming a person occupies the room for a total of 10 minutes following fuel rupture (while evacuation takes place) and that a respiration rate of 3.47×10^{-4} m^3/sec (Reg. Guide 4) is experienced, it is possible to project the personnel exposure and isotope ingestion. The following equations were used:

1. Whole Body Exposure $= \dfrac{\text{conc. of radioisotope}}{\text{MPC (Table 1, 10CFR20)}} \times 5000\,\dfrac{\text{mr}}{\text{yr}} \times \dfrac{10\text{ min}}{2000 \times 60\,\dfrac{\text{min}}{\text{yr}}}$

   (Assumes that continuous exposure to 1 MPC is equivalent to 5 rems/yr whole body dose)

2. Isotope ingested = conc. of radioisotope x $3.47 \times 10^{[illegible]}\,\dfrac{\text{ml}}{\text{sec}}$ x 600 sec

The results of such a calculation for a 12 wt% element are given below.

Radioisotope	Conc. µCi/ml	MPC Table 1	10 Min. Dose, mr	10 Min. Ingestion, µCi
Kr^{85m}	4.23×10^{-6}	6×10^{-6}	0.29	0.88
Kr^{85}	1.48×10^{-7}	1×10^{-5}	0.006	0.03
Kr^{87}	7.56×10^{-6}	1×10^{-6}	3.15	1.57
Xe^{133m}	6.04×10^{-7}	1×10^{-5}	0.02	0.13
Xe^{133}	2.16×10^{-5}	1×10^{-5}	0.90	4.50
Xe^{135m}	3.35×10^{-6}	4×10^{-6}(ass)	0.35	0.70
Xe^{135}	5.90×10^{-6}	4.10^{-6}	0.61	1.23
I^{131}	8.83×10^{-6}	9×10^{-9}(S)	409.0	1.83
I^{132}	1.25×10^{-5}	2×10^{-7}(S)	26.0	2.60
I^{133}	2.16×10^{-5}	3×10^{-8}(S)	300.2	4.50
I^{134}	2.30×10^{-5}	5×10^{-7}(S)	19.2	4.78
I^{135}	2.05×10^{-5}	1×10^{-7}(S)	85.3	4.27
			845.0	

Thus, the maximum whole body exposure to a person in the reactor room for 10 minutes immediately following a fuel element rupture in air would be 845 mrems--a value well within the requirements of 10CFR Part 20.

The ingestion of iodine isotopes identified above would result in the concentration of the isotopes in the thyroid with a resultant thyroid exposure. The extent of exposure can be calculated by multiplying the iodine ingested by the corresponding internal dose effectivity factor.

Radioisotope	10 Min Ingestion μCi	Effectivity Factor rem/Ci	Thyroid Dose, rems
I^{131}	1.83	1.486×10^{6}	2.72
I^{132}	2.60	5.288×10^{4}	0.14
I^{133}	4.50	3.951×10^{5}	1.78
I^{134}	4.78	2.538×10^{4}	0.12
I^{135}	4.27	1.231×10^{6}	5.25
Total thyroid dose			10.01 rems

This resultant thyroid dose is reasonable based on the conservative assumptions made.

(b) Exposure to the general public. In case of a fuel element rupture, some of the released radioisotopes would be swept from the reactor room by the ventilation system and discharged through the hood vent into the uncontrolled area. For the sake of the analysis of exposure to the general public, it is assumed that, at the time of the hypothetical fuel element rupture, the ventilation system would be switched to its "emergency" operation in which the exhaust air is diverted through an absolute filter before being vented. Furthermore, the air intakes to the room would be isolated to maintain the reactor room under negative pressure and inhibit radioactivity release except through the exhaust vent. The air flow through the exhaust under these conditions is 150 cfm which represents an air change rate in the room of about 1 change per hour.

The concentration of radioisotopes released from the fuel would be reduced by the air flow through the room. A common engineering calculation for a well mixed vessel assumes essentially complete flush of the vessel in 5 volume changes. However, for the purpose of the present calculations, only one room volume is assumed, and, thus, the concentration of isotopes in the room exhaust would be the same as was calculated above. However, the air is discharged at the roof level, approximately 40 feet above grade. Thus, a dilution due to wind will be encountered.

The Dilution Factor due to the building is defined by Lamarsh as:

$$DB = cA\bar{v}$$

where

c = shape factor, experimentally between 0.5 and 0.67

A = cross section area of the building

$\bar{v}$ = average wind speed

Since the prevailing winds near the MSTR are form the west, the north-south area of the Engineering Building is used, thus

$A = 500 \text{ ft} \times 40 \text{ ft} = 20{,}000 \text{ ft}^2$

$\bar{v} = 10.1 \text{ mph} = 14.8 \text{ ft/sec}$ (from National Weather Service Bureau)

$c = 0.5$

Thus, $D_{AB} = (0.5)(20{,}000)(14.8) = 1.48\times10^5 \frac{\text{ft}^3}{\text{sec}}$

With a stack flow rate of $150 \frac{\text{ft}^3}{\text{min}}$ ($= 2.5 \frac{\text{ft}^3}{\text{sec}}$), the concentration reduction due to wind dilution will be $\frac{2.5}{1.48\times10^5} = 1.69 \times 10^{-5}$. The concentration of nuclides will, thus, be reduced by this amount outside the building. The dose received by a member of the public over an hour[1] of occupancy may be estimated in a manner similar to that used for occupational exposure given above.

1. Whole Body Exposure $= \frac{\text{conc. of radioisotope}}{\text{MPC(Table 2, 10CFR20)}} \times \frac{500 \text{ mr}}{\text{yr}} \times \frac{1 \text{ hr}}{8760\text{hrs}}$

   (Assumes that continuous exposure to 1MPC is equivalent to 500 mr/yr whole body dose)

2. Isotope ingested = conc. of radioisotope $\times\ 3.47\times10^2 \frac{\text{ml}}{\text{sec}} \times 3600$ sec.

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[1] One hour exposure assumed because released activity will be dispersed after this time.

The results of these calculations are given below for the rupture of one 12 wt% fuel element in air with 100% release of halogens and rare gases in the air gap of the fuel.

Radioisotope	μCi/ml	Table 2	1 hr dose, mr	1 hr ingestion μCi
Kr^{85m}	7.15×10^{-11}	1×10^{-7}	4.08×10^{-5}	0.89×10^{-4}
Kr^{85}	2.5×10^{-12}	3×10^{-7}	4.76×10^{-7}	0.03×10^{-4}
Kr^{87}	1.28×10^{-10}	2×10^{-8}	3.6×10^{-4}	1.6×10^{-4}
Xe^{133m}	1.02×10^{-11}	3×10^{-7}	2×10^{-6}	0.13×10^{-4}
Xe^{133}	3.65×10^{-10}	3×10^{-7}	6.9×10^{-5}	4.56×10^{-4}
Xe^{135m}	5.66×10^{-11}	1×10^{-7}(ass.)	3.2×10^{-5}	0.71×10^{-4}
Xe^{135}	9.97×10^{-11}	1×10^{-7}	5.7×10^{-5}	1.25×10^{-4}
I^{131}	1.49×10^{-13}	1×10^{-10}(S)	8.5×10^{-5}	1.86×10^{-7}
I^{132}	2.11×10^{-10}	3×10^{-9}(S)	4.02×10^{-3}	2.64×10^{-4}
I^{133}	3.65×10^{-10}	4×10^{-10}(S)	5.2×10^{-2}	4.56×10^{-4}
I^{134}	3.89×10^{-10}	6×10^{-9}(S)	3.7×10^{-3}	4.86×10^{-4}
I^{135}	3.46×10^{-10}	1×10^{-9}(S)	1.98×10^{-2}	4.32×10^{-4}
Total Whole Body Exposure			0.080 mr	

Similarly, the thyroid exposure from the ingestion of iodines would be:

Radioisotope	1 hr Ingestion μCi	Effectivity Factor Rem/Ci	Thyroid Dose Rems
I^{131}	1.86×10^{-7}	1.486×10^{6}	2.76×10^{-7}
I^{132}	2.64×10^{-4}	5.288×10^{4}	1.40×10^{-5}
I^{133}	4.56×10^{-4}	3.951×10^{5}	2.0×10^{-4}
I^{134}	4.86×10^{-4}	2.538×10^{4}	1.23×10^{-5}
I^{135}	4.32×10^{-4}	1.231×10^{6}	5.0×10^{-4}
Total Thyroid Dose			7.58×10^{-4} Rems

These calculations indicate that the exposure to the general public as the result of a fuel element failure in air after extendod reactor operations would be negligible.

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REACTIVITY INSERTION

The present MSTR core consists of 70 elements, all of which are stainless steel clad U/ZrH with a nominal H to Zr ratio of 1.7. Eight of the elements are 12 wt % Uranium (<20% enriched) while 62 are 8.5 wt % Uranium (<20% enriched). The 12 wt % elements are located in the B and C rings of the core and two of these (one each in the B & C rings) are instrumented with chromel-alumel thermocouples.

Experiments conducted at MSU involving the rapid insertion of 1.39% $\Delta k/k$ into the core described above by means of a pneumatic control rod results in the production of a power pulse of approximately 258 MW_{th} and an integrated power output of about 15 MWsec. The temperature measured in the instrumented fuel elements located in the B and C rings, 12 wt % fuel, is approximately 275°C. It is expected that this fuel concentration in these locations would experience the highest neutron flux and temperature.

By comparison, experiments conducted at General Atomic (GA6216) with a Stainless Steel Core containing 8.5 wt % (<20% U^{235}) Stainless Steel Fuel and subjected to pulses of up to 3.5% $\Delta k/k$ reactivity insertion compared the temperatures produced. The temperatures in both the B and D rings were measured and a linear relationship between the fuel temperature and the quantity ($\Delta k/\beta$-1) was observed. (β = effective delayed neutron fraction = 0.7% for this fuel). Although experiments were not conducted at Δk = 1.37%, the temperatures produced may be estimated by linear extrapolation of the data obtained between 2.1 and 3.5% down to the 1.37% value corresponding to the MSU tests. This yields an estimated peak fuel temperature of 230°C (in 8.5 wt % fuel). This is consistent with the MSTR data. (Note that the MSTR fuel contains 12 wt % U as compared to 8.5% in the GA tests. The higher fuel content would be expected to result in higher power density and a correspondingly higher temperature. This increase might be expected to be approximately 12-8.5/8.5 = 41% to project a 12 wt % fuel temperature in the GA tests of (230-30)(1.41) + 30 = 322°C.)

The same GA report (6216) gives the results of theoretical calculations of peak and integrated power as well as fuel temperature use due to the reactivity insertion. The predicted temperatures compare closely with the observed values for large reactivity insertions. Similar calculations demonstrated that the peak temperature in the fuel occurs not at the midplane between the centerline and cladding where the thermocouple is located. The peak temperature is approximately 21% higher than the midplane temperature which would extrapolate the measured MSTR fuel temperature to (280-30)(1.21) + 30 = 332°C.

The maximum excess reactivity of the MSTR is 2.25% $\Delta k/k$; and, thus, it is theoretically possible to insert this amount of reactivity into the core. Again, based on GA 6216, such an insertion as a step increase would result in a bulk fuel temperature of 450°C. Again, correcting for the peak to bulk calculation and 12 vs 8.5% fuel as above results in a peak temperature of 716°C. This temperature exceeds to the phase transition temperature of low hydrogen (H/Zr≈1.5) fuel (GA7882) but the MSTR fuel with H/Zr≈1.7 shows no such phase transition; and, thus, temperatures well above 1000°C are permissible. Thus, the rapid insertion of the full 2.25% $\Delta k/k$ reactivity into the MSTR is not expected to result in a thermal degradation of the fuel.

High fuel temperatures might be expected to result in increased gas pressures within the fuel due to expansion of the gas between the fuel meat and the cladding as well as the release of fission product gases. Similarly, the $UZr/H_{1.7}$ will exert a partial pressure of hydrogen which is a function of fuel temperature.

Tests reported in GA6216 indicated that the maximum pressure generated in the fuel element was only 7 psig with a reactivity insertion of \$4.20 (2.95% $\Delta k/k$). It is reasonable to assume, then, that the peak pressure in the MSTR reactor with 2.25% insertion will be less than this (note that at \$4.20 insertion, the peak fuel temperature according to this same report would be 610°C (bulk)-30 x 1.21 (peak/bulk) + 30 = 732°C which is consistent with the anticipated MSTR peak of 716°C). A peak pressure of 7 psig would result in a stress of about 1500 psi in the fuel cladding. Report GA6216 gives the tensile strength of 304SS at 650°C to be 44,500 psi. Thus, the pressure generated by a 2.25% $\Delta k/k$ pulse will not even approach the bursting pressure of the cladding.

On the basis of this analysis, it is concluded that the rapid insertion of the full excess reactivity into the MSTR would not result in damage to the fuel.

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MECHANICAL REARRANGEMENT OF FUEL

When it is necessary to remove irradiated fuel from the MSTR, the following procedure is typically followed. A 400 lb lead transfer cask (such as is used with the BMI-1 shipping cask) is lowered into the reactor pool by means of a chain falls and A-frame superstructure. The cask is capable of holding three standard Triga fuel elements which are remotely loaded into the cask by a fuel handling tool. The cask is then removed from the pool and the lead shielding protects personnel from the radiation emitted by the fuel.

During the cask handling process it is possible that the transfer cask could be dropped into the pool. Such an accident might impact upon the reactor and the reflector. It is proposed to analyze the consequences of such an incident.

Several accident scenarios are possible:

1. Deflection of the control rod extensions with the resultant control rod withdrawal. Since the cask will impact in a downward direction, the principle force exerted will tend to drive the control rods into the core (a "safe" configuration). Even if a horizontal force is exerted on the extension rods, the removal of sufficient control rod worth to enable the reactor to go critical is unlikely in light of the shutdown margin inherent in the MSTR.
2. Crushing of fuel. In case of impact of the cask onto the fuel, deflection or crushing of the fuel is the likely result. The fuel cladding of several elements would probably be damaged and the rare gas/halogen fission products in the gas gap between the fuel an the cladding would escape. Since the only situation involving the fuel transfer cask would involve a flooded pool, the fission product release described above (for fuel failure in air) would be reduced by the dissolution of halogens in the pool water. As indicated in the air rupture analysis, the largest radiation is due to the Iodine release. For the present analysis, more than one element might be damaged resulting in the releases of greater amounts of rare gases and Iodine. However, the presence of the pool water would reduce the amount of Iodine released to the room and, thus, compensate for the increased number of elements involved. It is,

thus, estimated that the radiation exposure for this accident postulation would approximate that of the one analyzed previously (Fuel Element Failure in Air).

3. Impact on Reflect or Ion Chambers. The reflector of the MSTR will serve as an impact shield for the reactor core. It will, therefore, reduce the consequences of a falling cask over that described in #2 above. Since the reflector contains little radioactivity and no fuel, damage to it will not result in a significant radioactivity release.

It is, therefore, concluded that the worst case resulting from a cask dropping into the reactor pool would result in no greater consequences than the fuel element failure in air previously analyzed.