THE
UNIVERSITY
OF UTAH

DEPARTMENT OF
MECHANICAL AND
INDUSTRIAL ENGINEERING

MEB 3008
SALT LAKE CITY, UTAH 84112

November 29, 1984

Cecil O. Thomas, Chief
Standardization & Special Projects Branch
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir:

The enclosed materials are submitted in support of the University of Utah's application for renewal of its TRIGA Nuclear Reactor Operating License, R-126, Docket No. 50-407.

The applicant (University of Utah) formally submits the following:

1. Responses to specific questions posed by the NRC on the facility, operations and assessments.

2. Re producible figures required for preparation of the Safety Evaluation Report.

3. Corrections and amendments to the applicant's Safety Analysis Report.

The applicant has requested that the NRC grant a 20 year operating license for the TRIGA Reactor Facility.

Sincerely yours,

Gary M. Sandquist

G. M. Sandquist
Reactor Supervisor

Gary DeVries

K. L. DeVries
Reactor Administrator

GMS:ffc
Enclosures

J Broadbent
Notary Public
29 Nov 84
Salt Lake, Utah

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UNIVERSITY OF UTAH TRIGA REACTOR (UUTR)

1. What are the principal uses of the UUTR? What is the current use in megawatt-hours per year?

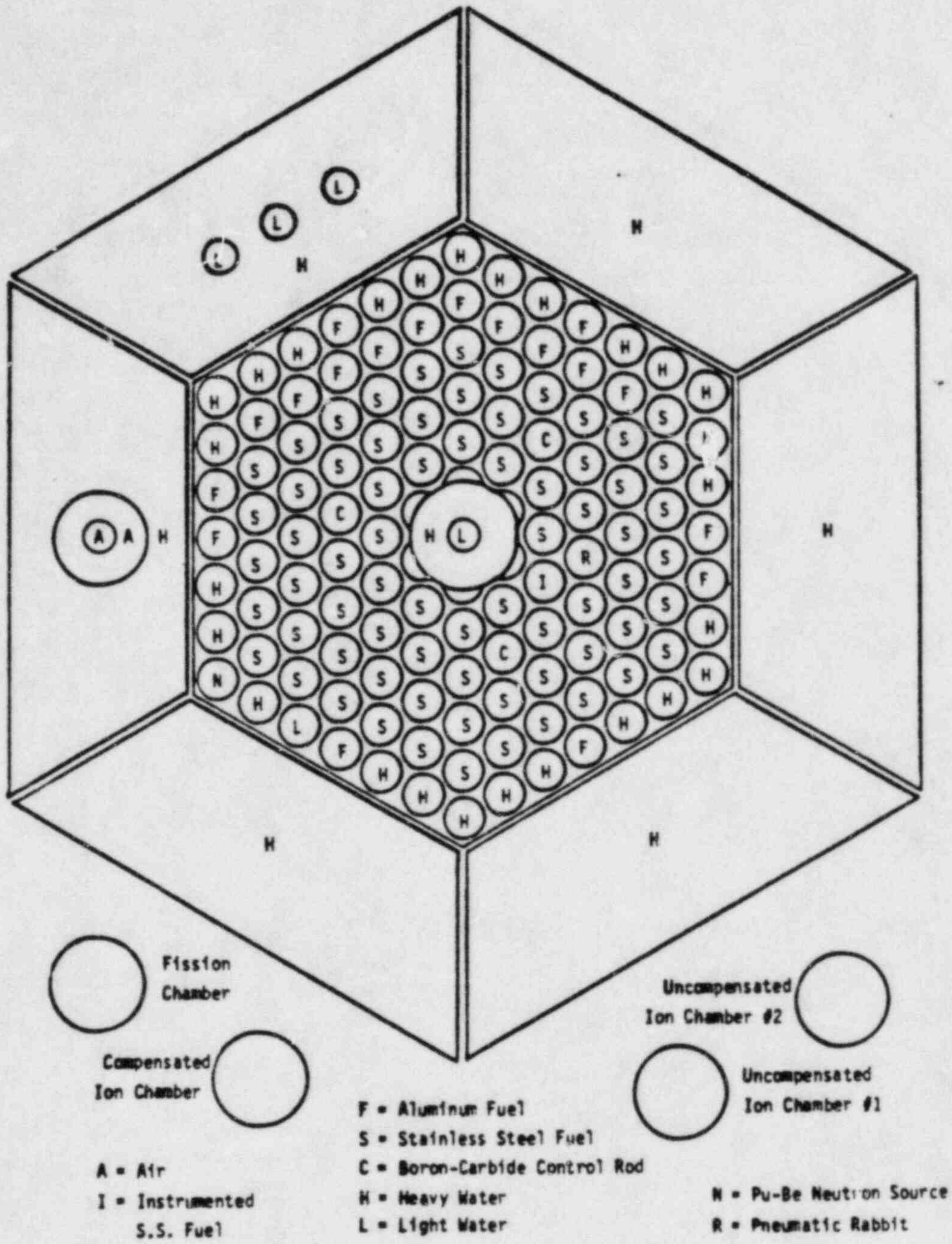
   The University of Utah's TRIGA Reactor (UUTR) is principally used in support of its educational program in Nuclear Engineering and for research. Current use averages about 10 Mw·hrs per year.

2. Provide a plan view of the current core configuration showing the number and locations of the fuel elements (differentiate between stainless-steel and aluminum elements), the control rods, the D_2O reflector elements, the D_2O reflectors tanks, the experimental tubes, the irradiation facilities, the neutron detectors, and the startup source. Also indicate how many and where (if any) the instrumented fuel elements (aluminum or stainless-steel) are in the current core and what type of thermocouples are used.

   The Figure enclosed provides a plan view of the current core configuration. The single instrumented fuel element in position C3 is stainless steel clad. Three thermocouples (chromel-alumel) provide channel output for fuel/fuel temperature sensing.

3. What administrative limits or requirements are placed on fuel loadings?

   The UUTR Technical Specifications limit fuel loading within the core by reactivity and control requirements (Sec.3.1) and by cladding and fuel type (Sec.4.1). Fuel loading operations require the approval of the Reactor Supervisor and presence of a senior licensed reactor operator as specified by the Nuclear Engineering Laboratory Operations Manual for the UUTR.



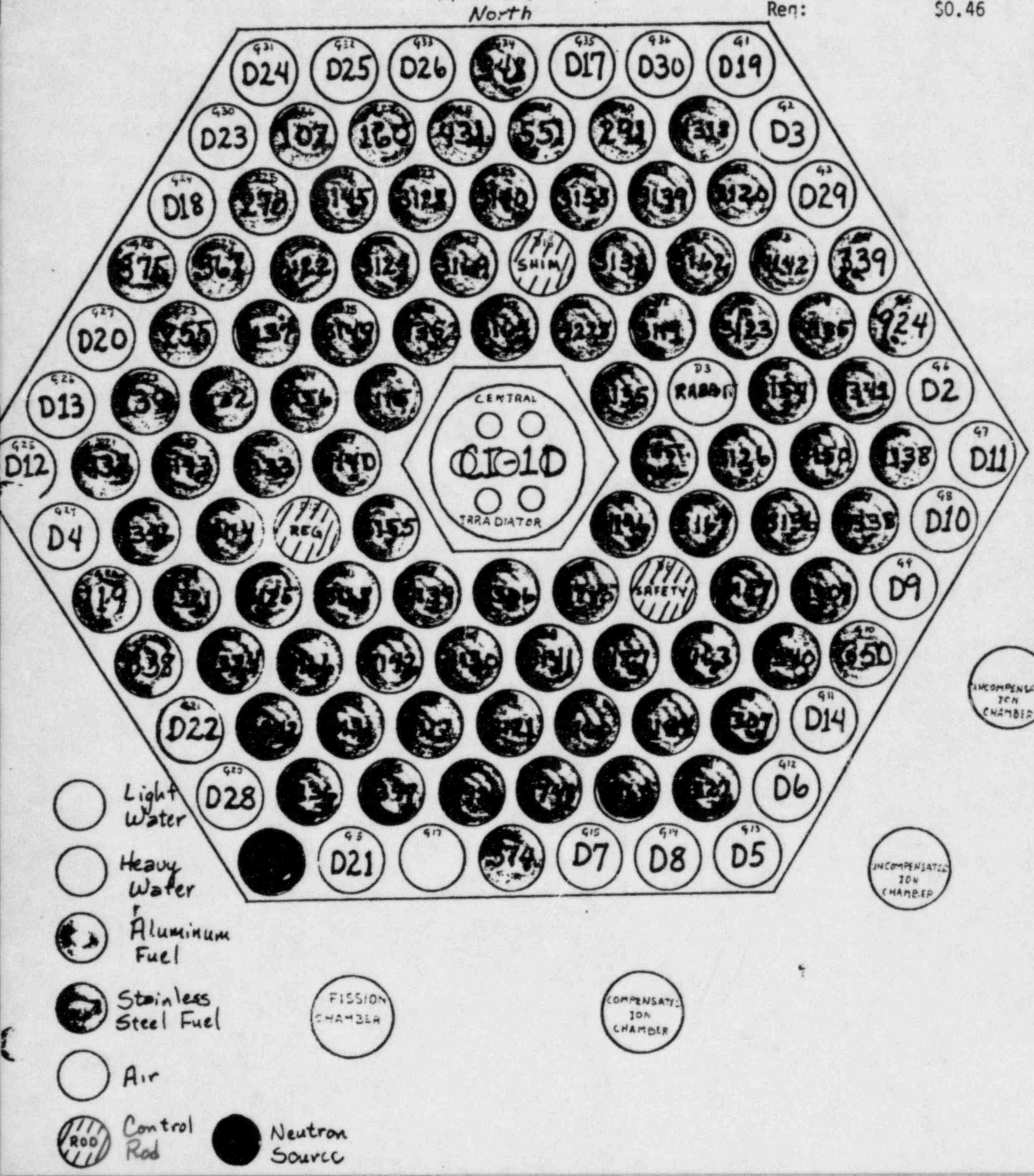
Core Configuration

UNIVERSITY OF UTAH

Core Configuration #18

April 25, 1984

Shut down :	$0.83
Excess React:	$1.18
Safety:	$1.76
Shim:	$1.55
Reg:	$0.46



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

U-235 content in core is 3.348 kg. Startup source is a 5.0 Ci Pu-Be source.

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

Excess reactivity $1.18, Safety rod $1.76, Shim rod $1.55, Regulating rod $0.46, Shut down margin is $0.83.

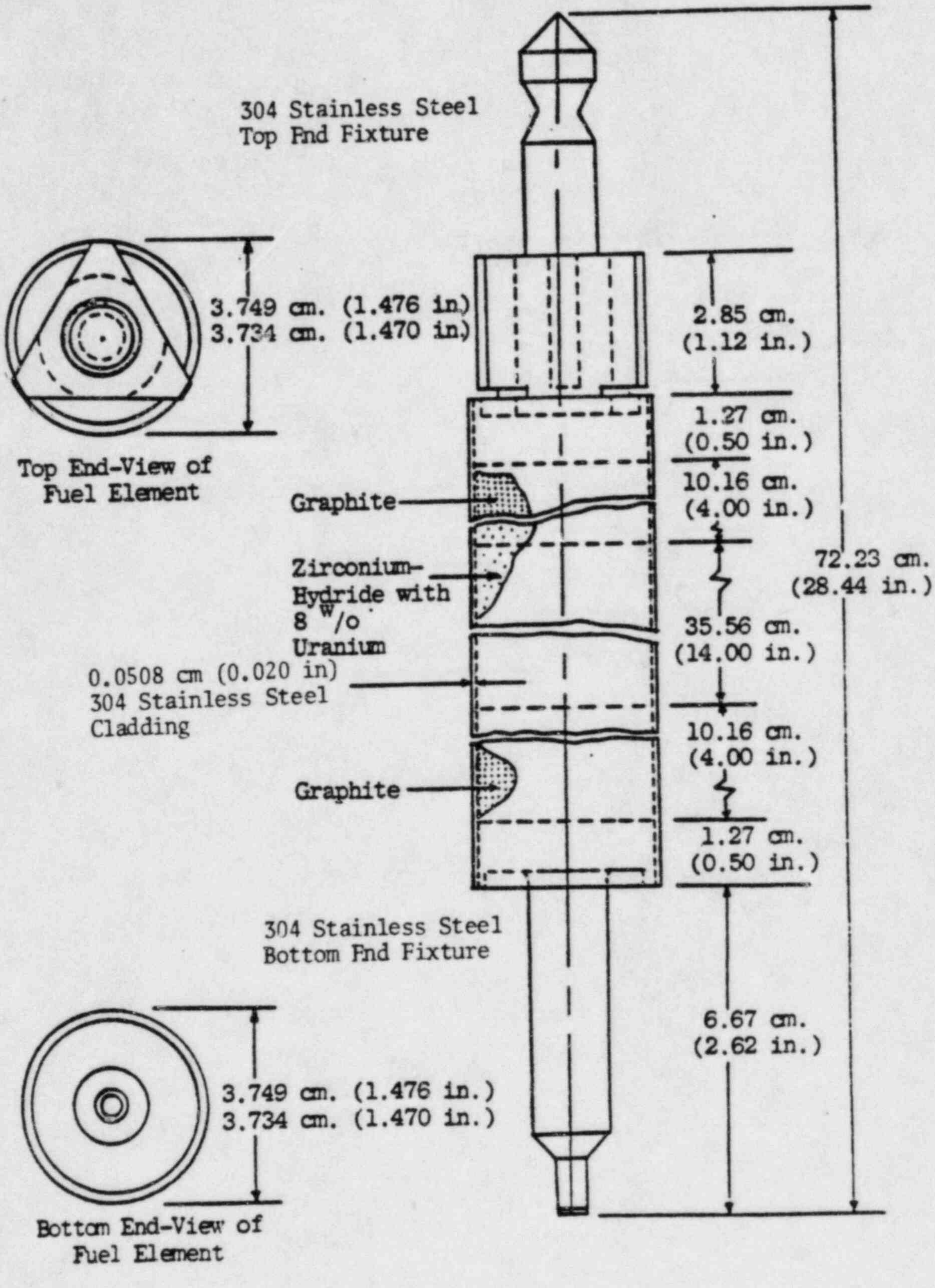
6. What is β-effective for the UUTR current core configuration?

Beta effective = 0.007.

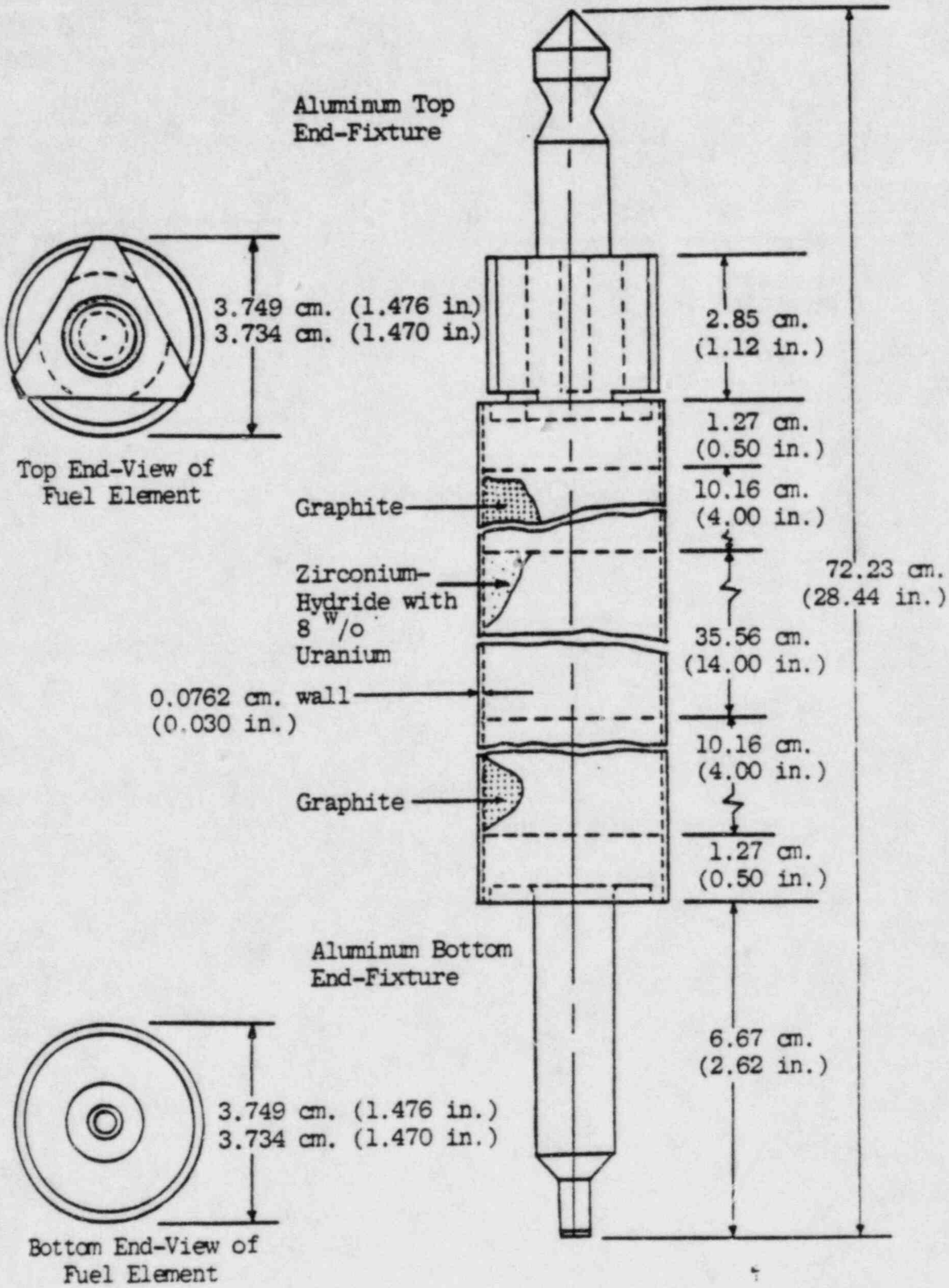
7. How many spare fuel elements are there? How many have been irradiated? Where are they stored? Describe the fuel handling tools used and fuel storage facilities at the UUTR. Give details of the current inventory of fuels in these storage areas.

There are 3 stainless clad and 45 aluminum clad fuel elements stored in the reactor tank outside the core or locked fuel storage pits. All have been irradiated (see fuel allocation table.) The fuel handling tool is the standard GA flexible cable with manual actuator which locks on the chamfered top of the element:

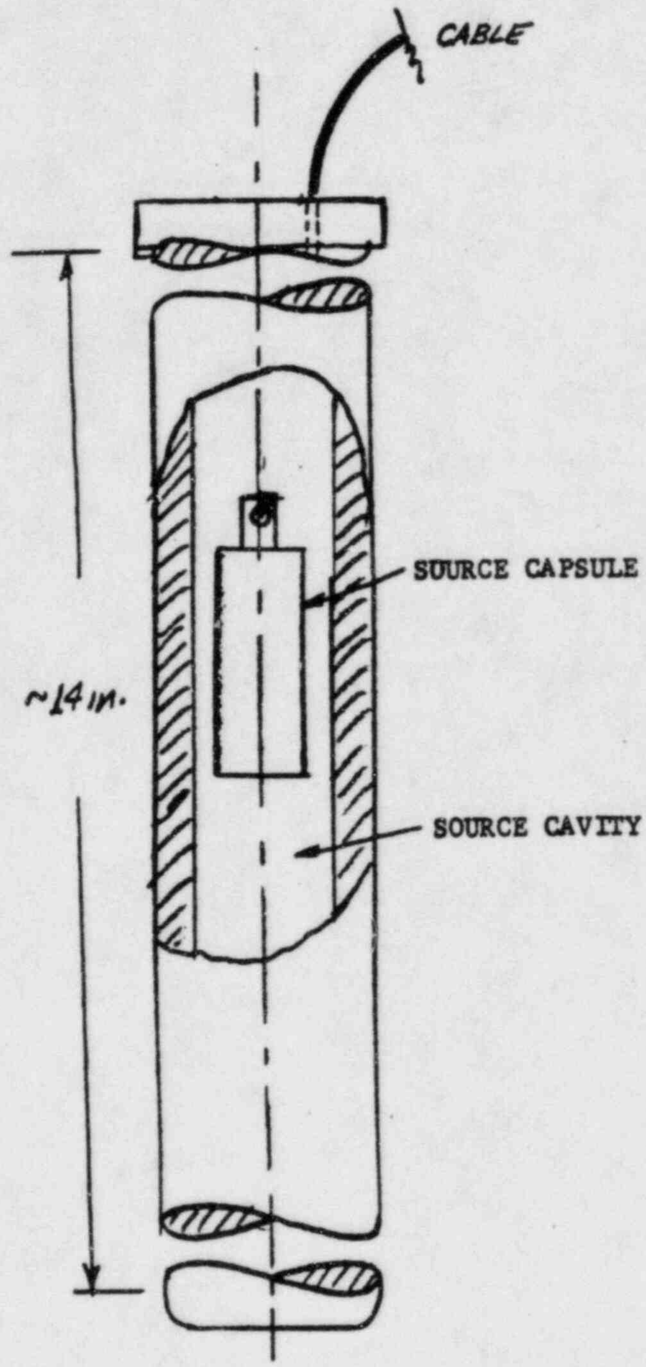
Reactor Tank	Fuel Storage Pits
43 A2 Clad	2 A2 Clad
3 SS Clad	0 SS Clad



Stainless-Steel Clad Fuel Element

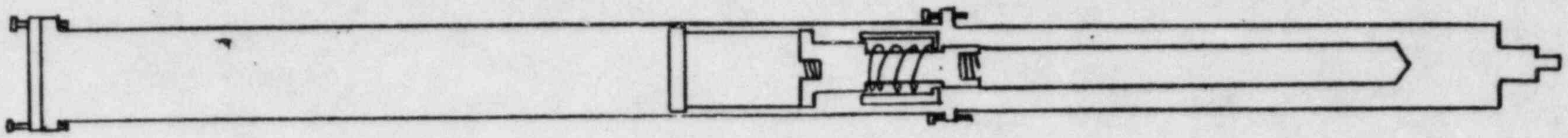


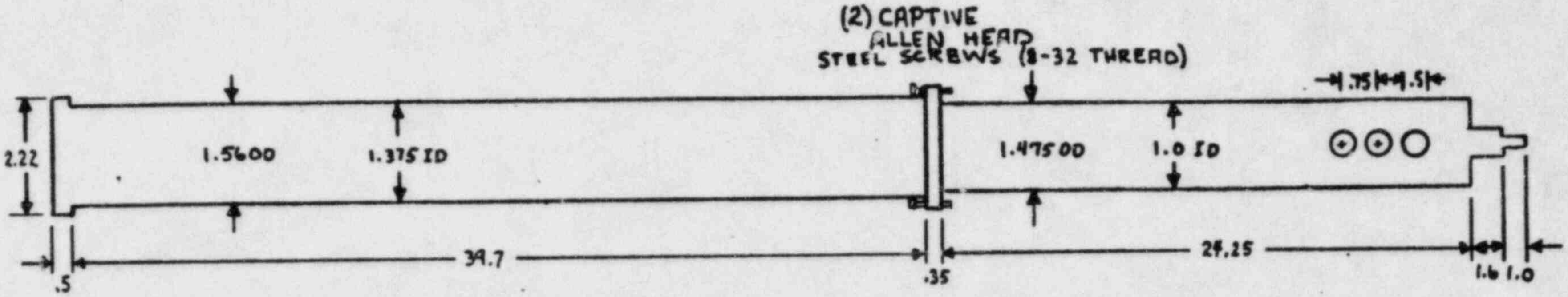
Aluminum-Clad Fuel Element



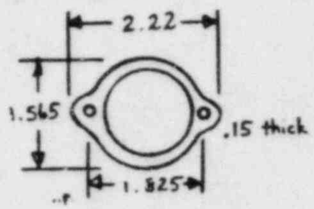
Typical Source Holder

Figure 5.3.7.6

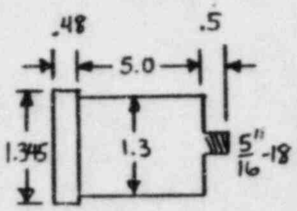




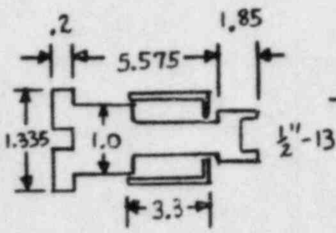
ALUMINUM GUIDE TUBE



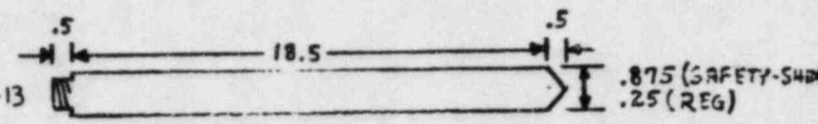
RETAINING COLLAR



ARMATURE AND LEAD WEIGHT



SHOCK ABSORBER AND DAMPING ASSEMBLY



ALUMINUM CLAD - BORON CARBIDE CONTROL ROD

DECEMBER 28, 1983 ALL MEASUREMENTS IN INCHES.	CONTROL ROD MECHANISM UNIVERSITY OF UTAH TRIGA REACTOR NUCLEAR ENGINEERING LABORATORY	KEVAN CRAWFORD SENIOR REACTOR ENGINEER

8. Provide a summary of the fuel element specifications for the stainless-steel-clad elements that are used in the UUTR. Include a schematic drawing of a stainless-steel-clad fuel element assembly.

   See Attached Drawings.

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

   The reactor console power is provided through an isolation transformer from building line power. The reactor console powers all reactor operations not including auxiliary functions such as lighting, overhead crane, recirculation pump, and facility security systems.

   The emergency power supply system powers facility security and lighting. The system (12 volts DC) consists of a series of large liquid acid batteries continually under recharge while line voltage is available.

   In the event of line power failure, the emergency power supply continues to provide power to the security system and lighting. If the reactor is operating during AC line power loss, the control rod magnet power is also lost, automatically scramming the reactor. However, the 12 volt DC emergency power supply continues operation, maintaining the radiation monitoring system and other emergency equipment.

10. Provide information on the dimensions of the neutron source holder and the manipulation of the neutron source during startup and power operation.

    See attached drawing. The neutron source is always located in the reactor core except when the power level of the core exceeds 1 watt of thermal power. Then the neutron source is removed from the core via a stainless steel cable attached to the source and stored during power operation in the tank wall storage rack.

11. What is the type and form of neutron poison contained in the control rods? What are the dimensions and what is the vertical travel length of the control rods?

    See figure attached. Neutron poison in control rod is boron-carbide, vertical rod travel is 15 inches.

12. What is the withdrawal speed of the control rods and what is the accuracy of the rod position indication?

Control rod withdrawal speed is 3.75 inches per minute or 4 minutes to drive a 15 inch rod. Accuracy of rod position repeatability is about 2% or ± $0.03 for the shim rod and ± $0.01 for the regulating rod.

Control Rod	Worth	Avg. Rate	Max. Rate
Safety	$1.76	$0.44/min	$0.91/min
Shim	$1.55	$0.39/min	$0.91/min
Regulating	$0.46	$0.12/min	$0.29/min

13. Provide information on the scrams and interlocks associated with the four neutron channels. Include set points and automatic actions (if any).

Scams and interlocks are specified in Section 3.3 of the Technical Specifications.

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Scams and interlocks are specified in Section 3.3 of the Technical Specifications.

Specification

The reactor shall not be operated unless the safety systems described in the following table are operable.

Safety System or Measuring Channel	Minimum Number Operable	Function
Fuel element temperature	1(a)	Scram at or below Limiting Safety System Setting
Reactor power level	2	Scram at 120 per cent of full licensed power
Manual console scram button	1	Manual scram
Magnet current key switch	1	Manual scram
Console power supply	1	Scram on loss of electrical power
Reactor tank water level	1	Scram at one foot below normal operating level
Startup count rate interlock	1	Prevent control rod withdrawal when neutron count rate is less than 2/sec.
Control rod withdrawal interlocks	all control rods	Prevent manual withdrawal of more than one control rod simultaneously.

(a) For periods of time for maintenance to the standard thermocouple fuel element, the reactor shall be in the shutdown condition with all control rods fully inserted, and, power to the control-rod magnets and actuating solenoid has been switched off and the key removed.

14. What type of signal initiate scrams? Indicate if the meters open circuits, relays, or initiate other actions.

Signal: high power signals will trip set points on the percent power channel, high log power channel, and linear power channel.
Action: relays cut magnet current and connect scram indicator lights.
Signal: high temperature signal will trip the set point on the fuel temperature circuit.
Action: relay cuts magnet current and connects scrams indicator light.
Signal: high radiation will trip set points on the area radiation monitor.
Action: Relay cuts magnet current, connects scram indicator light, and trips security relay to warn campus polich.
Signal: loss of high voltage power supply to core power detectors will trip relay.
Action: relay cuts magnet current and connects scram indicator light.

15. Can the UUTR reactor scram on a loss of high voltage signal to the neutron detectors and/or on loss of power to other individual instruments?

Loss of high voltage to the neutron detectors trips a relay in the power supply unit which trips a relay in the scram chassis and thus scrams the reactor.

Other individual instruments, such as the area radiation monitor and fuel temperature system have low level scrams, while the conductivity meter and ventilation have no power loss scrams.

16. Are there any rod withdrawal interlocks other than the source level and two "up" switches depressed at the same time? List those, if any, associated with the mode selector switch or the armature or lower rod limit switches.

The Shim/Safety and Regulating rod are interlocked with each other and the safety rod. Only when the safety rod is fully withdrawn and latched can the shim/safety or regulating rods be withdrawn. Upon a scram signal from any source to the scram buss, magnet power to all rods is interrupted, dropping the rods and the rod carriages are driven into their fully down position. Only one control can be withdrawn at a time.

17. Are there any provisions in the scram logic circuitry and interlocks to exclude a loss of function on a single component failure?

Scram of the UUTR is realized by AC power interruption to the control rod magnets. The standard TRIGA circuitry and interlocks are designed so that no single component failure can prevent scram capability. This is accomplished through a common buss for all scram signal inputs.

18. What provisions are used to prevent ganged rod withdrawal?

The solitary selector switch for rod withdrawal can only be set to one rod selection at a time.

19. Can the control rods be scrammed individually?

The control rods can be scrammed individually by interrupting magnet power to a given rod from the console. This process is used to determine individual control rod reactivity worths.

20. How is the pool bulk coolant temperature monitored and what is the frequency of temperature monitoring during reactor operation?

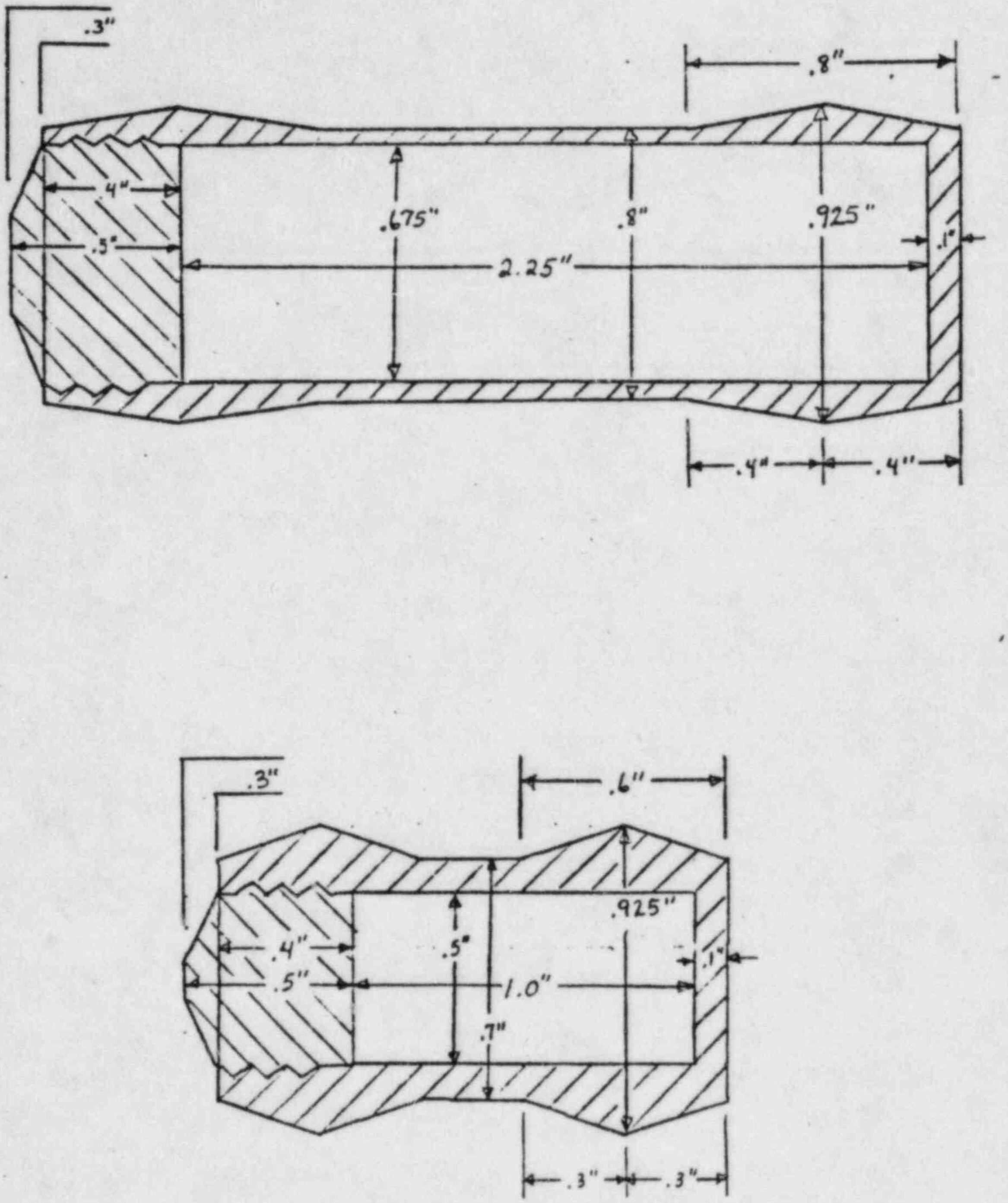
A thermocouple which can be rear at the console is located in the primary coolant circulation system. Also thermo couples which are located in the reactor tank and used for thermal power calibration, can be used to monitor tank water temperature. The primary coolant temperature is read during each reactor power operation as required by the checkout and operational logs. An alcohol thermometer situated in the reactor tank is used to verify the accuracy of thermocouples.

21. What is the maximum fuel temperature that has been attained for the UUTR? Indicate in what ring the instrumented element was located and whether the center irradiation facility replacing the B ring elements was installed.

The highest fuel temperature attained was 112°C while operating at 90% power (90 kw(th)). The instrumented fuel element was located in the C-ring with the D_2O filled central irradiator filling fuel positions in the A and B rings.

22. What restrictions are placed on the samples that are irradiated in the pneumatic transfer system? Describe the pneumatic transfer irradiation facility (materials of construction, dimensions, terminals, propulsion method, and controls).

All samples irradiated in the pneumatic transfer system must satisfy technical specification 3.6 and are always encapsulated and sealed in a polyethelene sample vial and then inserted in a polyethelen pneumatic transport vial (see figure). A detailed figure of the pneumatic rabbit is enclosed. All construction materials are aluminum except the damper spring which is stainless steel. Air is used to propel the sample in and out of the core. The control system for insertion and removal is electronic employing a time and air- value relay system. Fissile samples are not permitted to be irradiated in the pneumatic system under the provisions of the Nuclear Engineering Laboratory Operations Manual for the UUTR.



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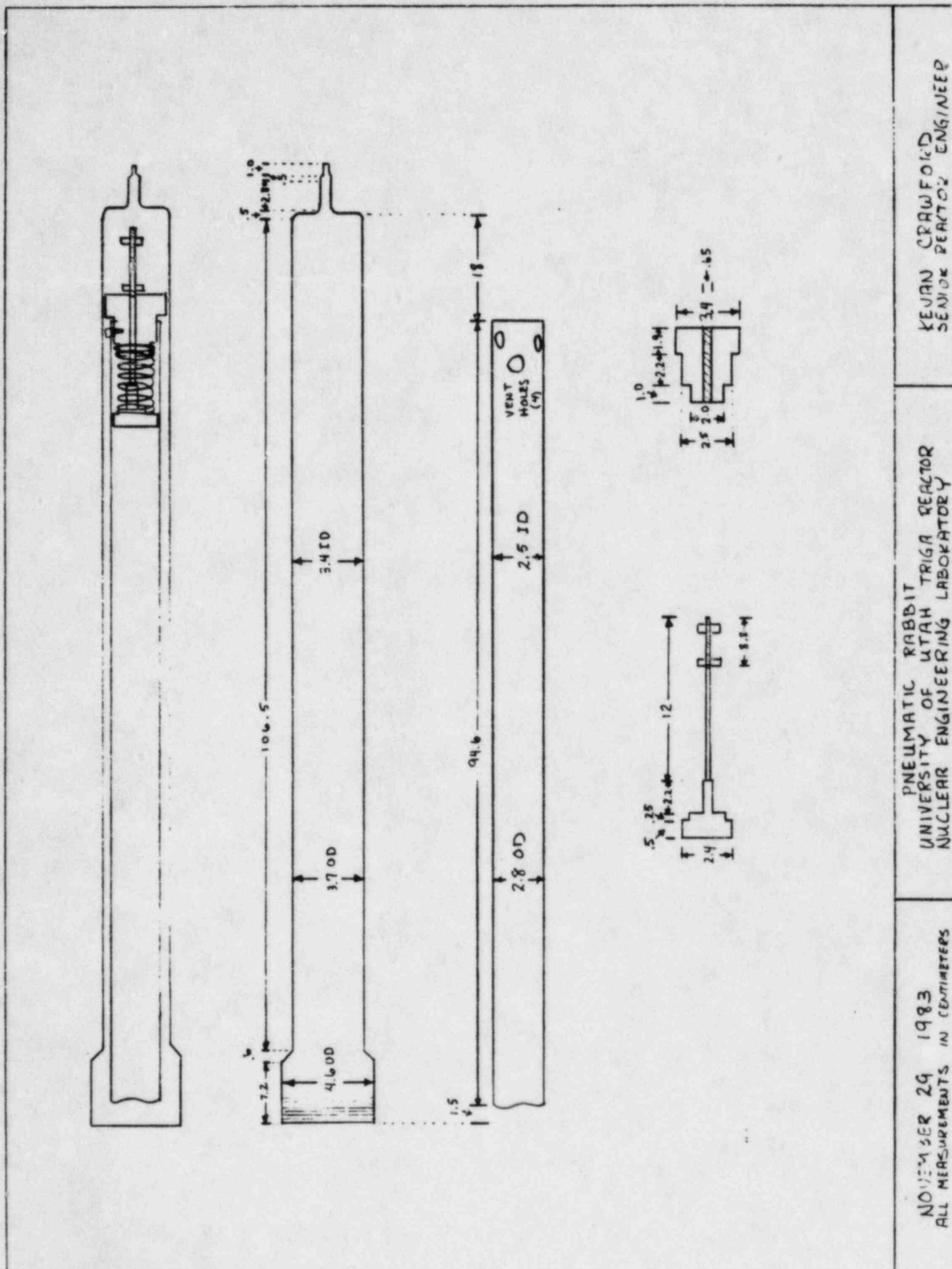
Polyethelene Irradiation Vials



In-Core Pneumatic Table.

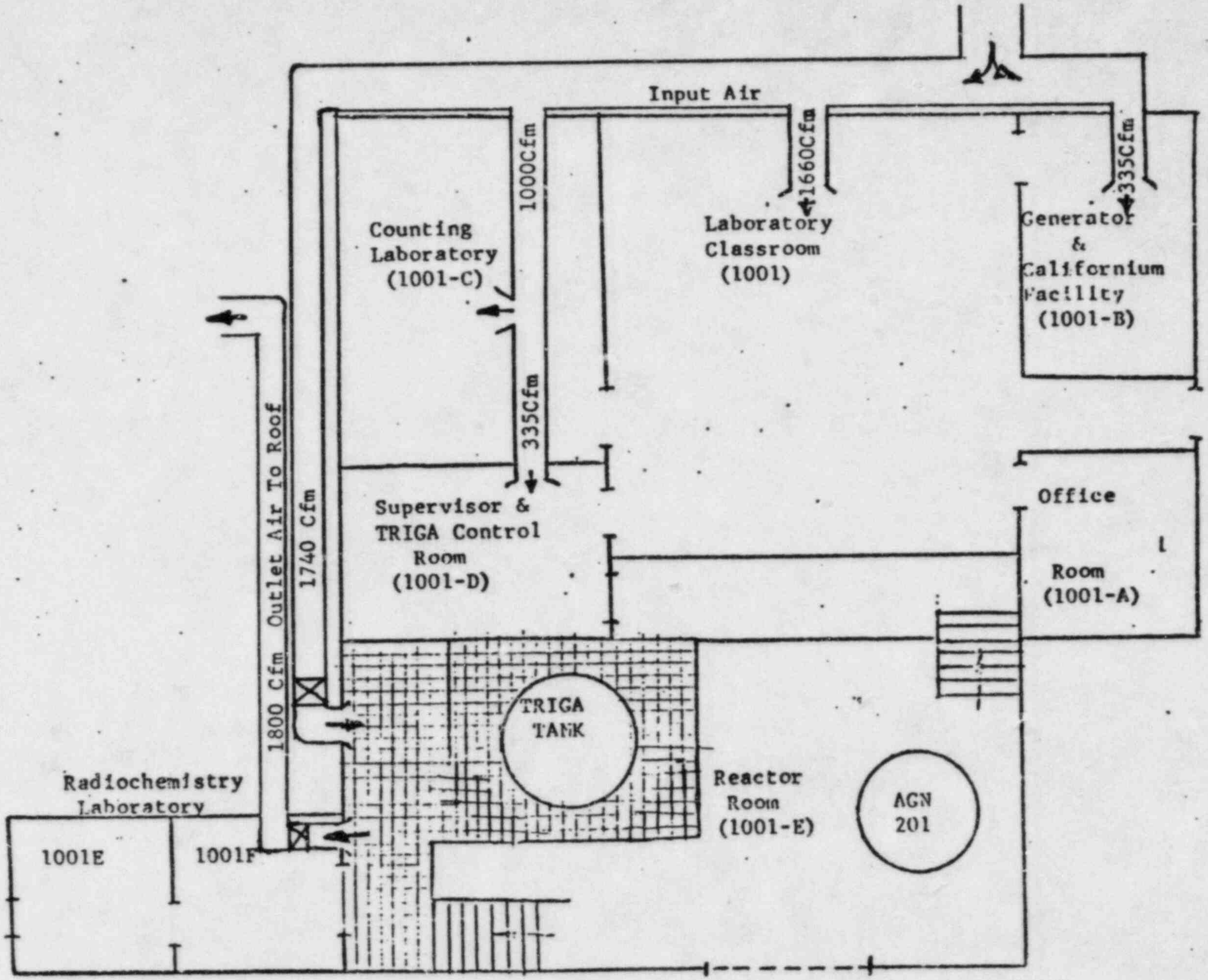
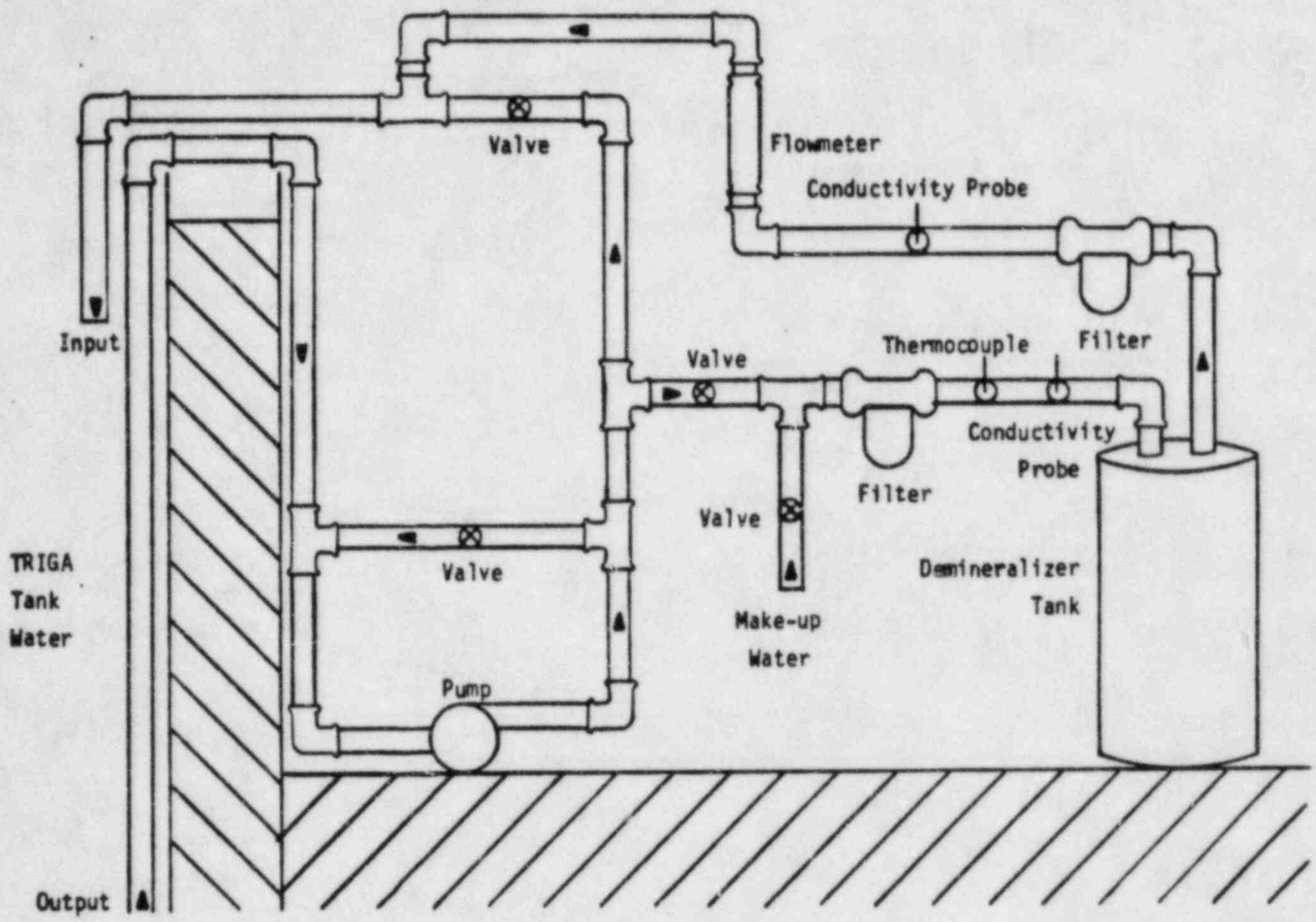


Figure 4.6.1 Ventilation System


Valve
Flowmeter
Conductivity Probe
Input
Filter
Valve
Thermocouple
Conductivity
Probe
Filter
Valve
TRIGA
Tank
Water
Valve
Demineralizer
Tank
Make-up
Water
Pump
Output

23. How is the ^{41}Ar from the irradiation of air in the pneumatic system handled.

Exhaust air from the pneumatic system is vented to the reactor room exhaust system and is monitored by the ventilation system detector. See figure for ventilation system schematic.

24. Please address the following accidents; include all assumptions made, calculative methods used, accident scenarios, and references to any documents.
(1) Fuel element failure in air
(2) Rapid insertion of the maximum allowable excess reactivity ($3.00) (step nuclear excursion)
(3) Loss-of-coolant accident
(4) Mechanical rearrangement of fuel (the consequences of dropping 1000 lb steel cask into core).

The following accidents:

(1) Fuel element failure in air
(2) Rapid insertion of $3.00 reactivity into the core
(3) Loss-of-coolant accident

have all been carefully addressed in the SAR in Chapter 8. The analyses given were very conservative and experimental measurements made on Ar-41, N-16, and other radiation values confirm this conservency. An excerpt from the SAR covering these accidents is provided for review. The fourth accident (i.e. 1000 lb cask dropped into core is addressed) here.

(4) Mechanical rearrangement of core fuel to an dropping of the 1000 lb storage cask into the core tank.

The dropping of the cask into the reactor tank is considered very unlikely due to the administrative and physical constraints imposed upon reactor crane operations. First, the crane is disabled by a locked power supply box except when expressly approved for operation by the reactor supervisor and under his direct supervision. Second, the crane is not to be operated when the reactor is in operation. Third, an additional safety cable is attached (by side body bolts) between the cask and the moving head of the crane whenever the cask is used over the reactor tank.

If the cask were to drop from crane, the heavy, structural beam bridge network that supports the control rod drives would at least deflect the cask so that it would not fall directly on the core. The control rods are bolted into the core assembly and any tilting or tipping of the core would result in the fuel moving from the core to the tank bottom into a subcritical geometry.

It is possible that a fuel element could be damaged so that the cladding would release some fission products to the primary water. How-

8.5 Reactivity Accident

The reactor fuel loading under the Technical Specifications can provide a maximum of 2.25% Δk ($3.00) excess reactivity above a cold, critical condition. In this section it is conservatively assumed that the core is loaded so that all of the innermost fuel positions contain fuel, starting from the central fuel (or B) ring, which consists of six elements. The maximum reactivity transient which could possibly occur would be that produced in such a densly packed core by the accidential insertion of the entire available amount of excess reactivity instaneously.

Prior to the development of the pulsing type fuel used on present TRIGA pulsing reactors, prototype TRIGA reactors (e.g. Torrey Pines, etc.) at Gulf Energy and Environmental Systems, Inc., using the same type of aluminum clad, low hydride fuel as is to be used in this reactor, were pulsed safely many times with up to 2.25% Δk insertions. The resulting power excursions attained a peak power of 1500 MW on a reactor period of 4.0 msec, with a total energy release during the burst of approximately 20 MW-sec. The maximum measured fuel temperature of all these pulses was less than 500°C. Curves of the typical transient power level and fuel temperature resulting from such power excursions are shown in Figure 8.5.1. These tests demonstrated that even a TRIGA reactor at zero power using non-pulsing type fuel can withstand a $3.00 reactivity insertion without hazard to either operating personnel or the public.

By deliberately violating the operating license and the several interlocks and scrams a sudden insertion of reactivity is possible while the reactor is at a power level up to 100 kW. From operating data reactivity loss due to temperature and xenon poisoning from operation at a steady

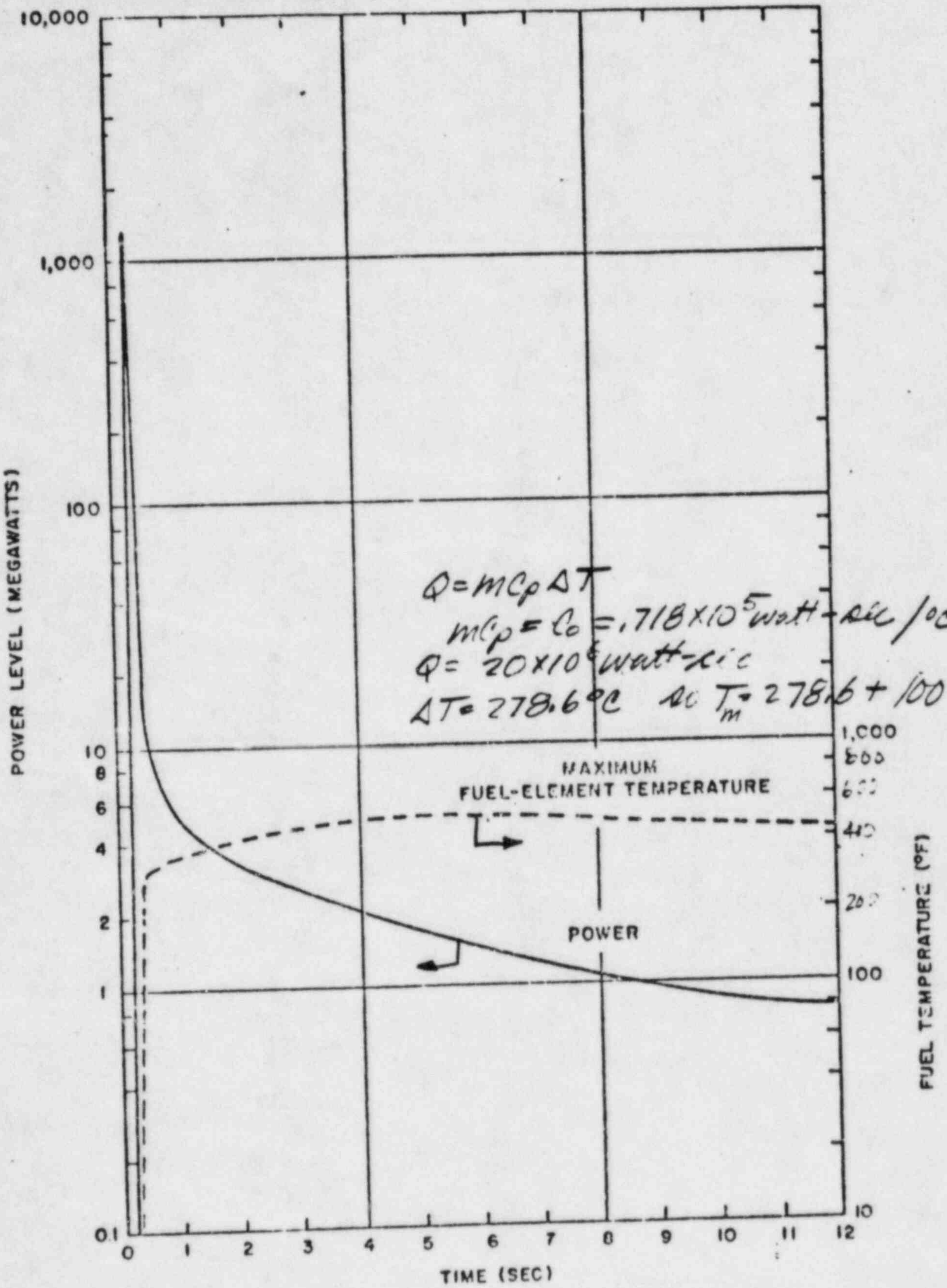


Fig. 8.5.1 Transient power and fuel temperature as functions of time after 2.25% $\delta k/k$ (\$3.00) reactivity insertion.

power of 100 kw is about 0.488% (or \$0.65) and the average fuel element temperature is 70°C. Now to estimate the fuel temperatures resulting from a prompt reactivity insertion at power the Fuchs - Nordheim model will be used. Parameters used in the analysis here are as follows:

a = negative temperature coefficient of reactivity

$= 1.34 \times 10^{-4}/°C$

C = heat capacity at equilibrium power kilowatts

$$= \Big[\underbrace{0.718 \times 10^5 + b\ (T(\text{fuel temp.}) - 25)}_{Co\ +\ T}\Big] \text{ watt-sec/°C}$$

Co = heat capacity at 25°C = 0.718×10^5 watt-sec/°C

b = heat capacity rate of change with temperature

$= 0.825 \times 10^2$ watt=sec/°C

Po = 100 kw

p = reactivity above prompt critical

= (\$3 - \$1) = \$2 = \$2 x 0.7×10^{-2}/\$ = 1.4×10^{-2}

This model which averages parameters, neglects heat transfer and delayed neutron effects yields the following set of coupled differential equations for reactor power P and fuel temperatures T:

$$l\mathring{P} = (p - aT)P$$

$$C\mathring{T} = P - Po$$

These equations may be combined to give

$$\frac{dP}{dT} = \frac{(p-aT)\ (Co + bT)P}{l\ \ \ (P-Po)}$$

and integration after separation of variables yields

$$l\left[(P-Po) - Po \ln \frac{P}{Po}\right] = T\left[bCo + (bp-aCo)\frac{T}{2} - \frac{abT^2}{3}\right]$$

The maximum average fuel temperatures occur when

$$\frac{dT}{dt} = 0 = P-Po$$

so P = Po and Tmax is found from the expression

$$Tmax = -3/8(A-1)\left[1 - \left\{1 + \frac{16}{3}\ \frac{A}{(A-1)}2\right\} 1/2\right]\frac{2p}{d} = 202.4°C$$

where $$A = \frac{ac}{bP} = \frac{1.34 \times 10^{-4} \times 0.776 \times 10^{5}}{0.825 \times 10_{2} \times 1.4 \times 10_{-2}} = 9.00$$

The result for the core-averaged fuel temperature at the conclusion of the pulse is

$$T = T_{fuel} + T_{clad} = 202.4 + 70 + 25 = 297.4°C$$

The maximum fuel temperature in the hot test fuel element for a peak-to-average power ratio of 1.45 for the UUTR is found to be

Tmax = 431°C (in the hottest element)

which is well within the 500°C range found from pulsing at zero power. Thus pulsing the reactor while at power up to 100 kw produces no significantly higher fuel temperatures than pulsing from zero power.

This anlysis conservatively neglects any transfer of thermal energy during the step impulse and assumes that the system parameters remain constant. Experimental data has shown that some transfer of thermal energy does occur during the pulse and system parameters to change so as to reduce the power output from the pulse. Furthermore, without a specially design pulsing rod it is difficult to achieve any significant reactivity insertion within a few milliseconds.

The first cladding failure in an aluminum clad, low hydride TRIGA fuel element was detected by general atomic in November 1959 after the element had been subjected to 709 pulses, 410 of which were at or about \$3. Subsequent investigation showed that failure was caused by ratcheting of cladding along the fuel body. None of the UUTR aluminum fuel has ever been exposed to pulse operation and the probability of failure of aluminum clad in a single, accidental pulse is extremely low.

8.6 Loss of Reactor Pool Water

8.6.1 Total Loss of Reactor Pool Water

Rapid and complete loss of the water in the reactor tank is considered to be highly improbable. Such an event could only reasonably arise from an earthquake shock so intense that the reactor hazard would only be one of many serious hazards to persons in the Merrill Engineering Building. Furthermore, the UUTR primary coolant system employes two water tight tank enclosures, viz an 8 foot diameter aluminum tank which is contained within an outside 12 foot diameter steel tank. Probability of simultaneous failure of both tanks is very low. However, provision has been made for warning of the loss of pool water as an isolated event and calculations have been made of the fuel temperatures and radiation levels to be expected.

A float switch alarm, operating on a continuous basis is provided in the tank to alert the operating staff and Campus Police Department it the water should reach about 6 inches below its normal level. In such event, the operating staff and/or the Campus Police Department will alert the Reactor Supervisor, and the Radiation Safety Officer immediately. If the water level were to fall while the reactor is or is not operating, the ceiling area radiation monitor over the reactor would also alert the operator of the high gamma radiation level and the reactor would scrammed automatically from either or both a low level water signal and a high radiation alarm and the area cleared. The operator would then attempt to maintain an adequate level of water over the core if possible by directing an emergency water line (tap water) into the tank. The reactor cooling system and return lines contain anti-syphon holes approximately one foot below the normal water level to prevent the loss of significant water.

The time it would take for the tank to drain may be conservatively estimated by considering a cataclysm that results in the complete opening of both containment tanks and the concrete pad to the ground soil at the bottom of the tanks. Under these conditions the flow of water through the soil is described by Darcy's law, where p is the pressure gradient

$$\frac{dx}{dt} = -.Kp$$

in meters of water head per meter and K is a constant depending on the nature of the soil, having the units m/year. The time to drain the tank is then t = 1/K, years/meter (H_2O). The premeability K has a maximum value for sand of about 3×10^3.[1] Thus to drain the tank to the bottom of the core (22 ft.) or 6.7 meters the time required is 0.0022 years or 19.3 hours. This maximum drain rate allows ample time for emergency procedures to be enacted. Again the primary emergency response procedure is the continual flooding of the core tank from the tap water line in the laboratory until radiation levels are acceptable, or the fuel can be removed to an acceptable storage facility.

8.6.2 Integrity of Fuel Element Cladding

The following calculation shows that rupture of fuel element cladding is not likely following loss of cooling water from the tank. The calculation assumes that the reactor has been operating continuously at full power for one year prior to loss of the water which is considered to be lost instantaneously following reactor shut-down. The maximum tempera-

1. Handbook of Applied Hydraulics, McGraw Hill, New York (1952) p. 165.
*Note, in Section 5, internal pressures of up to 15 atmospheres at the time of 10% burnup were conservatively assumed. Even such a high pressure gives cladding stresses well below the yield.

ture reached by the fuel an consequently the aluminum cladding temperature is less than 61°C (152°C @ 250 kw). This temperature is such that the pressure exerted by trapped air and fission product gases in the fuel is less than 30 psi. Such a pressure produces an internal stress of 264 psi (660 psi @ 250 kw). The yeild stress for the aluminum cladding is about 8,000 psi at 150°C.*

Therefore, the fission products will be retained in the fuel elements following operation at either 100 or 250 kW even with instaneous loss of coolant.

It should also be noted that the calculated temperatures are conservative (high) and assume that the only mechanism for heat removal from the fuel after the watger loss is through natural convection to ambient surrounding the core. Actually some heat will be removed by conduction to the grid plates, evaporation of water film on the fuel and some will be removed by radiation.

Method of Calculation

Use has been made of the two-dimensional, transient heat transport computer code, RAT, developed at Gulf Energy and Environmental Systems, Inc., for calculating the maximum temperature in the core after a water loss.

It is assumed that at the time at which the water was lost, the temperature distribution in the fuel element considered in this calculation was equal to the temperature distribution in the hottest (B ring) fuel element during steady-state operation of the reactor at 250 kW. The results of these calculations are obviously conservative for estimating the maximum core temperature at 100 kW operation.

It is also conservatively assumed that the reactor has been operating for a long time at 250 kW, and with a minimum fuel loading (only 62 elements) in the core. The rate of energy release in the B ring element was determined from consideration of the energy deposition of fission product gammas and betas only. The energy release from delayed neutrons is relatively small (about 1700 watt-sec total in the B ring element) and has an average decay constant of about 0.08 sec^{-1}.

The after-shutdown power density (in $Btu/hr\text{-}ft^3$) in the B ring fuel element is given by:

$$\frac{q}{v} = 0.1\ p\ \frac{P}{V_f} \cos\left[0.78\ \frac{\pi}{L}\left(x - \frac{L}{2}\right)\right]$$

$$\left\{\left[t + t_o + 10\right]^{-0.2} - 0.87\left[t + t_o + 2 \times 10^7\right]^{-0.2} - 0.05\right\} \qquad (1)$$

where p = Peak-to-average power density in the core = 2.0

P = Operating reactor power = 8.525×10^5 Btu/hr (250 kW)

V^f = Volume of the fuel in the core = 0.87 ft^3

L = Length of the fuel = 1.25 ft

x = Distance measured from the bottom of the fuel element, ft

t = Time after the core is exposed to the air, sec

t_o = Time from shutdown to the time the core is exposed, sec.

Equation (1) is a modification of the Untermeyer-Weill formula that matches the work of Stehn and Clancy[1] to about 5×10^4 sec after shutdown. It is also conservatively assumed that all the energy produced by fission product decay in the element is deposited in the element.

While the decay gammas and betas are raising the fuel element temperature, the flow of air between the fuel elements will be removing heat, tend-

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1. J. R. Stehn and E. F. Clancy, "Fission Product Radioactivity and Heat Generation," Paper No. 1071, Proc. Second United Nations Con. on the Peaceful Uses of Atomic Energy, Geneva, 1958.

ing to lower fuel temperature. The air velocity through the channel can be determined by setting the frictional pressure loss equal to the buoyancy. Entrance and exit losses which are about 2 to 5% of the friction losses and have been ignored.

$$\delta P \text{ (buoyancy)} = \delta P \text{ (friction)} \tag{2}$$

Term on the left is given by

$$\delta P \text{ (buoyancy)} = (\rho_0 - \rho_1) \frac{L}{2} \tag{3}$$

where L is the length of the channel and ρ_0 and ρ_1 are the entrance and exit air densities, respectively.

Since the frictional pressure drop calculations for laminar flow in non-circular channels are incorrect when expressed in terms of the hydraulic radius, the pressure drop for the TRIGA reactor must be predicted by other means. The method selected was to convert the free-flow area into an annulus around the fuel element. With an annular space of inner diameter D_1 and outer diameter D_2, the frictional pressure drop becomes[1]

$$\delta P \text{ (Friction)} = \frac{32\ \mu v L}{g\left[D_v^2 + D_1^2 - \dfrac{(D_2^2 - D_1^2)}{\text{Ln}\ (D_2/D_1)}\right]}$$

where μ = Viscosity of air, lb/hr-ft

v = Velocity of the air, ft/hr

L = Length of the fuel element, ft

D_1 = Fuel element diameter, ft

$D_2 = D_1 + 2b$, ft.

The term b in D_2 is the effective separation distance between the B

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1. W. H. Adams, "Heat Transmission", McGraw-Hill, New York, N. Y., 1954, p. 149.

ring fuel element and those in the C ring. The use of the minimum separation distance as b would yield too large a pressure drop and is considered too restrictive. The use of the average separation distance, based on the free flow between the B ring element and the C ring, would yield too low a pressure drop since the pressure drop is not a linear function of the separation distance. As an approximation, b is taken as the mean of the two values, that is

$$b = 1/2(0.01130 + .04930) = 0.0303 \text{ ft.} \tag{5}$$

For the channel between the B ring and the C ring, the equation for the pressure balance (2) becomes

$$(p_0 - p_1) = 6.88 \times 10^{-5} \mu v, \tag{6}$$

where $\overline{\mu}$ is the average viscosity of the air (lb/hr-ft) in the channel and is a function of the entrance and exit temperatures, v is the average air velocity (ft/hr) in the channel, and the entrance and exit densities are ρ_0 and ρ_1 (lb/ft^3).

The mass flow rate of air in the channel is

$$w = v\overline{\rho}A_c \tag{7}$$

where $\overline{\rho}$ is the average density of air in the channel and A_c is the flow area associated with the channel. Combining equations (6) and (7), one obtains the mass flow rate

$$w = 405.(p_0 - p_1)\overline{\rho}/\overline{\mu} \, . \tag{8}$$

Assuming that the average properties in equation (8) are the average of the properties at the entrance and exit, equation (8) becomes

$$w = 405. \frac{(p_0{}^2 - p_1{}^2)}{(\mu_0 + \mu_1)} \tag{9}$$

Over the range of temperatures of interest the properties of air have been approximated by linear equations. Thus,

$$\rho = \frac{1}{2.5 \times 10^{-2} T} \ \mathrm{lb/ft^3} \qquad (10)$$

and

$$\mu = (0.01135 + 0.6017 \times 10^{-4} T) \ \mathrm{lb/hr\text{-}ft}$$

where T is the temperature in °Rankine.

Using these expressions in equation (9) one obtains

$$w = \frac{3.24 \times 10^5 \ (T_1^2 + T_0^2)}{(T_1^2 T_0^2) \left[0.01135 + 0.30085 \times 10^{-4} (T_1 + T_0)\right]} \ \mathrm{lb/hr} \qquad (11)$$

The determination of the amount of heat removed by this air flowing past the element rests on the evaluation of a heat transfer coefficient. For a free-standing cylinder cooled by natural circulation, the heat transfer coefficient is given, conservatively, by

$$h = 0.531 \frac{k_f}{L} [GrPr]^{0.25} \qquad (12)$$

where k_f is the thermal conductivity of air film at temperature T_f (Btu/hr-ft-°F), Gr is the Grashof number, and Pr is the Prandtl number. Because the fuel element is surrounded by adjacent elements, the flow will probably not be laminar, even at low Reynolds numbers, and the heat transfer correlation should be larger than that assumed, perhaps by as much as a factor of 2.

Again, over the temperature range of interest, one can write for the thermal conductivity and specific heat of the air

$$k_a = (0.0009 + 0.26 \times 10^{-4} T) \ \mathrm{Btu/hr\text{-}ft\text{-}°F} \qquad (13)$$

and

$$C_{pa} = 0.240 \text{ Btu/lb-°F}$$

with temperature, T, in °R. Using these values and those in equation (10) in equation (12) one obtains

$$h = 50.16 \left[\left\{ 0.0009 + 0.13 \times 10^{-4} (T_w + T_a) \right\}^3 \left\{ [1.25 \times 10^{-2} (T_w + T_a)]^{-2} \right. \right. \tag{14}$$

$$\left. \left. [0.5(T_w + T_a)] [0.01135 + 0.30085 \times 10^{-4} (T_w + T_a)] \right\}^{-1} (T_w - T_a) \right]^{0.25}$$

where T_w is the wall temperature, T_a is the bulk air temperature, and T_f is the average of the two, all in °R.

For the purpose of this anlaysis the heat transfer along the graphite end reflectors was neglected, and the fuel element was considered to be a 1.25-ft-long, 0.1225-ft-diameter cylinder of U-ZrH_1 with specific heat and thermal conductivity given by:

$$C_{pf} = (26.3 + 0.0245 \text{ T}) \text{ Btu/ft}_3 \text{ - °F}$$

and (15)

$$k_f = (10.7 - 6.42 \times 10^{-4} \text{T}) \text{ Btu/hr-ft-°F}$$

where T is the local temperature in °R. The temperature drop in the clad was ignored because, at the time of peak temperature, it has been calculated to be insignificant (1 - 5°C). For the computer program, the fuel element was divided into five radial and five axial regions, and the temperature in each region was computed as a function of time after complete water loss.

The maximum fuel temperature found from the program as a function of time after the loss of cooling water was found to be 152°C. The loss of water in the core was assumed to take place within seconds after shutdown, and the maximum initial fuel temperature is 113°C.

To determine the pressure exerted on the cladding by released hydrogen, fission products, and air trapped in the fuel can, the conservative assumption will be made that the entire system is at peak fuel temperature, i.e., about 152°C.

The total number of fission product nuclei released to the gap between the fuel and clad was determined from Blomeke and Todd,[1] and the results of the experiment described in Section 8.7. The total quantity of Br, I, Kr, and Xe released to the gap in the B ring fuel element after continuous operation at 250 kW for about one year will be

$$N_i = 0.028 \times 6.87 \times 10^{20} = 1.93 \times 10^{19} \text{ atoms.} \tag{16}$$

The number of gram-atoms in the gap is

$$n_{fp} = \frac{1.933 \times 10^{19}}{6.02 \times 10^{23}} \approx 3.22 \times 10^{-5} \text{ gram-atoms} \tag{17}$$

The partial pressure exerted by the fission products gases is

$$P_{fp} = n_{fp} \frac{RT}{V} \tag{18}$$

where, initially, the volume, V, is taken as the 1/8-inch space between the end of the fuel and the reflector end piece. This is quite conservative because the graphite reflector pieces have a porosity of 20% and the fission product gases can expand into the graphite. The initial volume, then, is

$$V_0 = \pi r^2 h = \pi(1.80)^2 \ 0.317 \text{ cm}^3 = 3.23 \text{ cm}^3,$$

where the radius of the fuel material, r, is about 1.80 cm and the space width, h, is 1/8-inch (0.317 cm).

Thus the initial pressure exerted by all the fission product gases is

$$P_{fp} = \frac{3.216 \times 10^{-5}}{3.23} RT = 0.997 \times 10^{-5} RT \tag{20}$$

1. J. O. Blomeke and Mary F. Todd, "U-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time", ORNL-2127.

The partial pressure of the air in the fuel element is

$$P_{air} = \frac{RT}{22.4 \times 10^3} = 4.46 \times 10^{-5} RT, \tag{21}$$

and the total pressure exerted by the air and fission products is

$$P_r = \left(1 + \frac{p^{fp}}{P_{air}} \right) P_{air} = \left(1 + \frac{0.997 \times 10^{-5}}{4.46 \times 10^{-5}} \right) P_{air} = 1.22\ P_{air} \tag{22}$$

Also $P_{air} = 14.7 \ \frac{T}{273} \ \frac{V^0}{V}$ (23)

where P_{air} is in lb/in.^{2} and T is in °K.

The equilibrium hydrogen pressure over the U-$ZrH_{1.0}$ fuel material is simply a function of the fuel temperature. At a temperature of 152°C, the hydrogen pressure is negligible (see Figure 5.2.2)

The total gas pressure at the maximum fuel temperature of 152°C is

$$P = P_H + P_r = 0 + 1.223\ (14.7)\ \left(\frac{152 + 273}{273}\right) \frac{3.23}{V} = \frac{90.0}{V} \text{ psi.} \tag{24}$$

Ignoring the negligible increase in volume caused by the expansion of the clad the total pressure then is

$$P = \frac{90.0}{V} = \frac{90.0}{V_0} = \frac{90.0}{3.23} = 27.8 \text{ psi.} \tag{25}$$

The tangential stress in the fuel element cladding when subjected to an internal pressure, P is

$$S = Pr/t, \tag{26}$$

where r is the radius of the fuel element can (1.80 cm) and t is the wall thickness (0.076 cm) or

$$S = 23.7P \tag{27}$$

The stress in the cladding is

$$S = 23.7(27.8) = 660 \text{ psi.} \tag{28}$$

From the Aluminum Company of America handbook,[1] the yield stress for type 6061 aluminum at 150°C is greater than 8,000 psi.

Consequently it is concluded that, subsequent to the complete loss of cooling water, the release of hydrogen from the fuel and the expansion of air and fission product gases in the space between fuel and graphite end pieces will not result in the rupture of the fuel element cladding.

8.6.3 Radiation Levels After Loss of Pool Water

Dose rates have been calculated for two locations; 1. at the top of the pool, 2. at the floor level of the laboratory over the reactor. Again, the reactor was assumed to have been operating for a long time at power levels of 100 kW and 250 kW prior to loss of water.

The dose rates calculated for instantaneous loss of water at shutdown are given in Table 8.6.1 and are clearly hazardous. However, as shown in section 8.6.1, the instantaneous loss of all the water surrounding the core is not conceivable and it is more realistic to estimate the dose rates with gradual water loss. These values are given in Table 8.6.2.

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1. Aluminum Company of America, "Alcoa Aluminum Handbook," (1957), p. 26.

TABLE 8.6.1

CALCULATED RADIATION DOSE RATES AFTER

TOTAL LOSS OF REACTOR POOL WATER

Time after shutdown	Direct Radiation Level (r/hr) (1) Top of Pool 100 kw	250 kw	(2) Floor of Laboratory 100 kw	250 kw
0.1 hr	440	1100	4.1	10.2
1.0 hr	236	590	2.3	5.75
10 hr	112	280	1.1	2.75
1 day	80	200	.69	1.73
1 week	36.4	91	.36	0.90
1 month	6.6	16.5	.065	0.164

TABLE 8.6.2

CALCULATED RADIATION DOSE RATES WITH

LOSS OF POOL WATER AT 0.4 METERS PER HOUR

Time after shutdown	Water Depth over core (m)	Direct Radiation at Floor of Laboratory (mr/hr) 100 kw	250 kw
0.1 hr	6.1	10^{-10}	10^{-10}
1.0 hr	5.6	10^{-10}	10^{-10}
5.0 hr	4.1	2.1×10^{-10}	5.2×10^{-10}
10.0 hr	2.1	0.051	0.128
15.2 hr	0	880	2200**
24.0 hr	0	692	1730

**This value could be reduced to less than 1 mr/hr by maintaining the water level at about 6 feet in the tank from the main water line in the laboratory.

More than 5 hours is provided by the alarm* before the dose rate approaches the "permissible" 0.24 mr/hr level in the area above the reactor. Such time is ample to clear the area, investigate the damage and initiate remedial action. In addition a rough estimate of the radiation dose at the side of the pool from scattered radiation indicates that an individual who did not expose himself to the direct radiation from the core could work there for at least 12 hours, 1 day after the accident, without receiving a dose in excess of that permitted by 10 CFR 20.101 for a calendar quarter. This time would be sufficient to view the interior of the pool with a mirror and to continue emergency repairs.

Method of Calculation

The following assumptions were made:

1. The core had been operating at power for a very long time (1000 hr) prior to shutdown.
2. The reactor is shutdown as soon as the leak develops.
3. The core can be approximated as a point source with effective strength determined by the photons escaping from an isotropic volume-equivalent spherical source.
4. The photon energy can be averaged to 1.0 Mev and only photons from fission product decay are important.
5. Attenuation by core components other than fuel elements is negligible.

Assumptions 1. and 5. are conservative, and 4. is optimistic. The net effect is conservative.

Effective Source Strength

The fission product decay photon source strength is given by Perkins and King[1] and the value of the effective source strength is assumed to be all at 1.0 Mev.

1. J. F. Perkins and R. W. King, Nucl. Sci. Engin., 3, 726 (1958).
*Activated by float switch

The source strengths obtained are given in Table 8.6.3.

Each fuel element has a volume of 400 m^3, thus a 67 element core has a volume of 2.68×10^4 cm^3 which is equivalent to a solid sphere of 18.4 cm radius. The fraction of photons which escape from the core is given[1] as a function of $\mu_c r_c$, and is 0.197, where μ_c is the attenuation coefficient of the core material for 1 Mev photons (= 0.207 cm^{-1}; GA-5400, p 8-53) and r_c is the radius of the sphere, 18.4 cm.

The product of the source strength and the fraction escaped is the effective source strength of the core. These values are listed in Table 8.6.3.

TABLE 8.6.3

EFFECTIVE SOURCE STRENGTH FOR SPHERICAL

REACTOR CORE OPERATED FOR 1000 HOURS AT 100 KW

Time after shutdown	Source Strength Mev/sec-100 kw	Effective Source Strength
0.1 hr	6.49×10^{15}	1.28×10^{15}
0.5 hr	4.52×10^{15}	8.88×10^{14}
1.0 hr	3.60×10^{15}	7.08×10^{14}
5.0 hr	2.20×10^{15}	4.32×10^{14}
10.0 hr	1.70×10^{15}	3.36×10^{14}
24.0 hr	1.20×10^{15}	2.36×10^{14}
1 week	5.48×10^{14}	1.08×10^{14}
1 month	1.00×10^{14}	1.96×10^{13}

1. K. K. Aglintsev, Applied Dosimetry, (Eng. Ed.), Iliffe, London (1965).

Dose Rates-Direct Radiation

The dose rate is computed on the basis that the core is a point source, 21 feet below the top of the pool:

$$D = \frac{S_{eff}}{4\pi R^2 K} \qquad B(R\mu)e^{\mu x}$$

where D = the dose rate in rads/hr (= rem/hr for 1 Mev γ)

S_{eff} = effective source strength (col. 3., Table 8.6.3)

R = distance between core and location considered.

K = conversion factor for Mev/cm^2 to rad/hr (= 5.77×10^5 for 1 Mev photons)

B = build-up factor for medium involved. B is a function of $R\mu$ for any material, so will vary with R. Values for water were taken from ref. 1, those for concrete from ref. 2.

u = linear attenuation coefficient, cm^{-1} ref. 1 gives for 1 Mev photons $\mu_{concrete}$ = 0.149 cm

μ_{water} = 0.067 cm

x = thickness of material, cm. (10 cm for concrete ceiling of reactor room).

Attenuation by air was neglected.

8.7 Fuel Cladding Failure

The effect of a fuel element rupture resulting in a release of fission product s from a B ring fuel element has been evaluated using a recently determined experiment value for the fission product release fraction. Also considered is the incredible situation in which cladding rupture occurs following the loss of pool water. This accident is considered incredible as it entails two essentially independent accidents occurring simultaneously.

1. H. O. Wyckoff, "Radiation Attenuation Data", in Radiation Hygine Handbook, H. Blatz, ed. McGraw-Hill, New York 1959. Chapter 8.
2. Reactor Handbook, Vol. III, Part B, "Shielding", (1962).

8.7.1 Fission Product Inventory

The inventory of radioactive fission product gases, halogens and noble gases, in the B ring fuel element in which the average power density is 1.6 times greater than the core average, was determined by assuming long term operation at 100 kw yield with 70 elements in the core. Using the data developed by Blomeke and Todd[1] results in the compilation of the volatile fission products given in Table 8.7.1.

For the purpose of these analysis, the gaseous fission products have been divided into two groups. The first comprises those isotopes that will remain in the tank water should the rupture occur when the pool is filled with water. In this group are the bromines and iodines. In the incredible event that no water is present, these isotopes would be added to the radioactive cloud and add to the hazard.

The second group comprises the insoluble volatiles, the krypton and xenon isotopes. They are the major source of radioactivity in the room (and outside) if the pool water is present.

8.7.2 Fission Product Release Fraction

The steady state release of short-lived fission gases from specimens of TRIGA-type fuels has been studied at temperatures ranging from ambient to 400°C. The specimens (0.33 inches in diameter, 0.33 inches long) were cylindrical in shape and of the TRIGA fuel composition, U-$ZrH_{1.7}$.

1. J. O. Blomeke and Mary F. Todd, "U-235 Fission Product

TABLE 8.7.1

GASEOUS FISSION PRODUCTS IN B-RING FUEL ELEMENT 100 kw SATURATED ACTIVITY--70 ELEMENT CORE

Nuclide	Decay Constant (hr^{-1})	Inventory (curies/elem.)
Group I		
Br 83	3.02×10^{-1}	9.84
Br 84	1.31×10^{0}	22.6
Br 84m	6.95×10^{0}	.465
Br 85	1.39×10^{1}	30.6
Br 87	4.49×10^{1}	55.0
I 129	4.60×10^{-12}	20.5
I 131	3.58×10^{-3}	59.4
I 132	3.07×10^{-1}	90.0
I 133	3.34×10^{-2}	133.0
I 134	7.93×10^{-1}	155.7
I 135	1.04×10^{-1}	121.0
I 136	2.90×10^{1}	63.6
	Total Iodines	644.
	Total Group I	762
Group II		
Kr 83m	3.66×10^{-1}	9.84
Kr 85m	1.59×10^{-1}	30.6
Kr 85	7.67×10^{-6}	6.15
Kr 87	5.35×10^{-1}	55.2
Kr 88	2.50×10^{-1}	76.1
Kr 89	1.31×10^{1}	94.5
Kr 90	7.55×10^{1}	106.0
Kr 91	2.54×10^{2}	63.3
Xe 131m	2.41×10^{-3}	0.594
Xe 133m	1.26×10^{-2}	3.20
Xe 133	5.50×10^{-3}	133.0
Xe 135m	2.67×10^{0}	36.4
Xe 135	7.60×10^{-2}	98.1
Xe 137	1.07×10^{1}	12.1
Xe 138	2.45×10^{0}	112.4
Xe 139	6.08×10^{1}	116.3
Xe 140	1.56×10^{2}	122.0
	Total Group II	1077

The TRIGA King Furnace used for these studies is essentially a minature in-core-loop and is described in detail in several reports. (See the HTRDA Semi-annual report for July to December, 1964, pp 9-11, and Gulf General Atomic report GA-8597 by F. C. Foushee). The essential feature of the furnance is that it can be inserted into a fuel element position of the TRIGA reactor, a sample heated to any desired temperature from 100 to 1700°C (present license upper limit) and the gases released during a subsequent irradiation purged to a cold trap. Gases are then analyzed by γ-spectrometry and their steady state release fractions determined.

Aside from experimental errors and statistical scatter there is no temperature or large half-life effect from room temperature to 300°C. The mechanism of release through this temperature range is principally due to recoil of fission fragments. The runs were all performed on the same sample, and on the same day. As long as release was of a recoil nature the build-up of the longer lived fission gases from run to run did not affect the results. From these measurements it can be concluded that below 300°C the fractional release of fission products is about 1.5×10^{-5}.

Applying this release fraction to the inventory of gaseous fission products produced in the fuel, as given in Table 8.7.1 gives the total activity that would be released should the integrity of a fuel element cladding be compromised. Thus, the toital rate gas activity released would be 1.62×10^{-2} curies, and the total iodines 9.65×10^{-3} curies, and the total halogens 1.14×10^{-2} curies.

8.7.3 Water Activity

In the fission product release accident in which the pool water remains in situ, the fraction of the total soluble fission products which are released from the element is distributed in the water. Thus 1.5×10^{-5} times the Group I fission products or 1.14×10^{-2} curies remains in the pool. Since the volume of water in the reactor pool is 3.57×10^{7} cm^3, the activity concentration is 3.20×10^{-4} μc/cm^3. The activity remains moderately high because of the small decay constants for I-129, I-131, and I-133. However, the demineralizer can be used to remove the soluble volatile fission products.

8.7.4 Exposure Inside the Reactor Room

The maximum exposure to a person in the reactor room will occur if the fission products from a ruptured element are distributed instantaneously within the room and no air change occurs.

The total release of noble gases is 1.62×10^{-2} curies and the volume of the reactor room is 5.7×10^{8} cm^3 so that the concentration is 0.284×10^{-4} microcuries/cm^3, or 1.04 dis/sec-cm^3. If the room is assumed to be equivalent to a hemisphere of radius R = 643 cm, the dose rate at the center is given by

$$D = \frac{S}{2K\mu}(1 - e^{-\mu R})$$

S = source strength, dis/sec-cm^3

K = conversion factor for flux--dose rate

= $4.2 \times 10^{4}\gamma$/cm^2-sec per mr/min for photons of 0.7 Mev (average energy of f.p. decay)

μ = attenuation coefficient for air

= 3.5×10^{-5} cm^{-1} for 0.7 Mev γ

Thus the maximum dose rate if 0.8 mr/min. If the halogens were to be released as well, this would be increased to about 1.36 mr/min. Thus an individual could remain in the room for about 75 min before exceeding the dose of 100 mr. The controlling factor, however will be the dose to the thyroid from halogen activity in the second case. In either event the dose rates are sufficient to set off the radiation alarms in the area so that the room will be cleared rapidly.

In addition it is probable that the emergency purge exhause (see 4.6.2) would be operating in such an accident, venting air at the rate of 150 cfm. This will decrease the dose rate to a person in the reactor room.

The dose to the thyroid can be calculated at any time (t) from[1]

$$D = \frac{(5.92 \times 10^2)\ A\ f\ E\ (\ 1 - e^{-\lambda\ t})}{m_\lambda} (\text{rads})$$

where

A = inhaled iodine, in curies

f = fraction which is deposited in critical organ

E = effective energy absorbed by thyroid per disintegration (Mev)

λ = effective decay constant, comprising both radioactive decay and biological elimination

m = mass of the thyroid in grams

in a long time, t = ∞, and using T (=0.693/λ), the equation becomes:

$$D_\infty = \frac{(8.54 \times 10^2\ A\ f\ E\ T)}{m}\ (\text{rads})$$

$$= \text{constant} \times \sum_i A_i\ E_i\ T_i$$

where the parameters are for the ith isotope.

1. J. J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, 1962.

The inhaled iodine is given by

$$A_i = R \left(\frac{A_i}{V}\right)$$

where $\frac{A_i}{v}$ is the activity concentration, and R the breathing rate for a time τ. If τ is made equal to 1 sec. then

$$D_\infty = \frac{(8.54 \times 10^2)\, f\, R}{M} \sum_i \frac{A_1\, E_i\, T_i}{v} \quad \text{(rads/sec)}$$

For the standard man[1], $f = 0.23$

$m = 20$ grams

$R = 10\ m^3/8$ hr

$= 3.47 \times 10^{-4}\ m^3\ sec^{-1}$

and the value of the constant is thus 3.41×10^{-3}.

The necessary activity concentrations are found from Table 8.7.1 using 1.5×10^{-5} for the release fraction and summation for I-131 through I-136 yields a value for $D_\infty = 0.45$ rads/sec.

Assuming that 300 rads to the thyroid is a limiting exposure,[1] a person will have approximately 37 minutes to evacuate from the reactor room. In fact the time will be longer than this since the iodine isotopes will plate out in the reactor room and the room exhaust will be in operation and will reduce the dose.

1. "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)," Health Physics 3, June 1960.

8.7.5 Exposure Outside the Building

On release of radioactive fission products into the reactor room, the continuous air monitor will close the normal exhaust and start the emergency purge. The room air will then be exhausted through a filter at 150 cfm (0.071 m^3/sec) which will not remove the noble gases and may not remove all of the iodines released in the event of pool water loss.

At the west penthouse roof level, on the Merrill Engineering Building where the emergency purge exhausts, there are also many other exhaust outlets, excluding those from the reactor area. Those in the immediate vicinity of the reactor exhaust have an exhaust rate totalling 18,500 cfm are designed to operate 24 hrs a day. Assuming this condition prevails, it is reasonable to assume a dilution of the exhaust effluent by a factor of 70 at the very least. Using this, and assuming no further dilution or decay occurs outside the building the maximum dose to the thyroid may be calculated.

The calculation assumes:

(1) Complete mixing in the reactor room at all times

(2) The person is immersed in the effluent from the building in which the concentration of radioactivity X at any time t is equal to $C_t/70$ where C_t is the concentration in the exhaust and in the room.

With (1) C_t is given by a simple first order rate law:

$$C_t = C_0 \exp \left[- \frac{a}{V} t \right]$$

where a = exhaust rate in m^3/sec

V = room volume in m^3

C_0 = initial concentration, at t = o

1. J. J. DiNunno et al., loc. cit.

the differential inhaled activity is $dA_t = R\,X\,dt$ and the total amount inhaled up to time t is

$$A_t = \frac{RC_o}{70} \int_o^t \exp\left[- \frac{a}{V} t \right] dt$$

$$= \frac{RC_oV}{70a} \left[1 - \exp\left(- \frac{a}{V} t \right) \right]$$

If the exposure is for an infinite time,

$$A_\infty = RC_oV/a(70)$$

and, comparing withthe previous calculation, the total dose

$$\text{int } D_\infty = \frac{D_\infty(\text{rads/sec.}) \cdot V(m^3)}{70\ a\ (m^3/\text{sec})}$$

$$= 26 \text{ rads}$$

If the person were to remain in the effluent cloud for only one hour, the corresponding figure is about 15 rads. Thus we may conclude that for any location other than on the roof of the building, the maximum possible dose to the thyroid from a fission product release will be considerably less than 1/20th of a maximum permissible dose, even when no credit is taken for radioactive decay.

Culf Energy and Environmental Systems, Inc. has developed computer programs which take allowance both for decay and for further meterological dilution.[1] Calculations using these programs for a similar situation to that presented here,[2] indicate that these factors will reduce the maximum

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1. GA-6511, "GADØSE and GADØSET--Programs to Calculate Environmental Consequences of Radioactivity Release". GAMD-5939, "PHICHK and PHIBAR programs to Calculate Atmospheric Dispersion".
2. Addendum to GA-6499, "Hazards Report for the Oregon State University 250 kw TRIGA Mark II reactor--dated June 23, 1965", April 20, 1966.

dose to a person a short distance away from the building by several orders of magnitude. Thus for a slow leak rate from the room ($a = 1.69 \times 10^{-2}$ m^3/sec) and very stable atmospheric conditions (see 8.4.4) they find:

	Whole Body γ	Thyroid
Outside building Max dose (12 hours)	2.9×10^{-3} mr	2.6 mr
Inside reactor area Max dose (1 hour)	6.0×10^{-1} mr	7.0 r

These exposure values are negligible except for an individual in the reactor area. Even there the exposure is not unduely high for such a highly improbable accident.

ever, the release of fission products to the tank water from the complete cladding failure of one B ring fuel element in water at 100kN saturated activity (note the reactor core should be shut down during crane operation. (See Table 8.7.1 and SAR, Section 8.7.3) is given by the release fraction of 1.5×10^{-5} times the concentrations of the respective group of fission products. We find that the rate gas release is

1.62×10^{-2} curies, 9.65×10^{-3} curies for the iodines and 1.14×10^{-2} curies for halogens. The volume of primary water is 3.6×10^7 cm^3 so activities are respectively: 4.5×10^{-4} $\mu Ci/cm^3$ for rate gases, 2.7×10^{-4} $\mu Ci/cm^3$ for iodines and 3.2×10^{-4} $\mu Ci/cm^3$ for halogens. Although these values exceed 10 CFR 20 limits they pose no significant radiation exposure risk to laboratory workers and can be readily removed with the water purification system. The iodines and halogens would remain fixed in the primary water but it is possible that the rare gases could move into the reactor room air. Assuming an instaneous, complete loss of all rare gas fission products to the room air (5.7×10^8 cm^3) results in an activity of 2.8×10^{-5} $\mu Ci/cm$. The dose due to this release is found to be (see SAR, pg. 8-44) 0.88 mr/min neglecting the reduction that would occur from the ventillation system. Furthermore this level would trip the radiation alarm and the worker would have 75 minutes to exit the area before receiving a dose of 100 mr.

25. Confirm that you have no graphite loaded reflector elements.

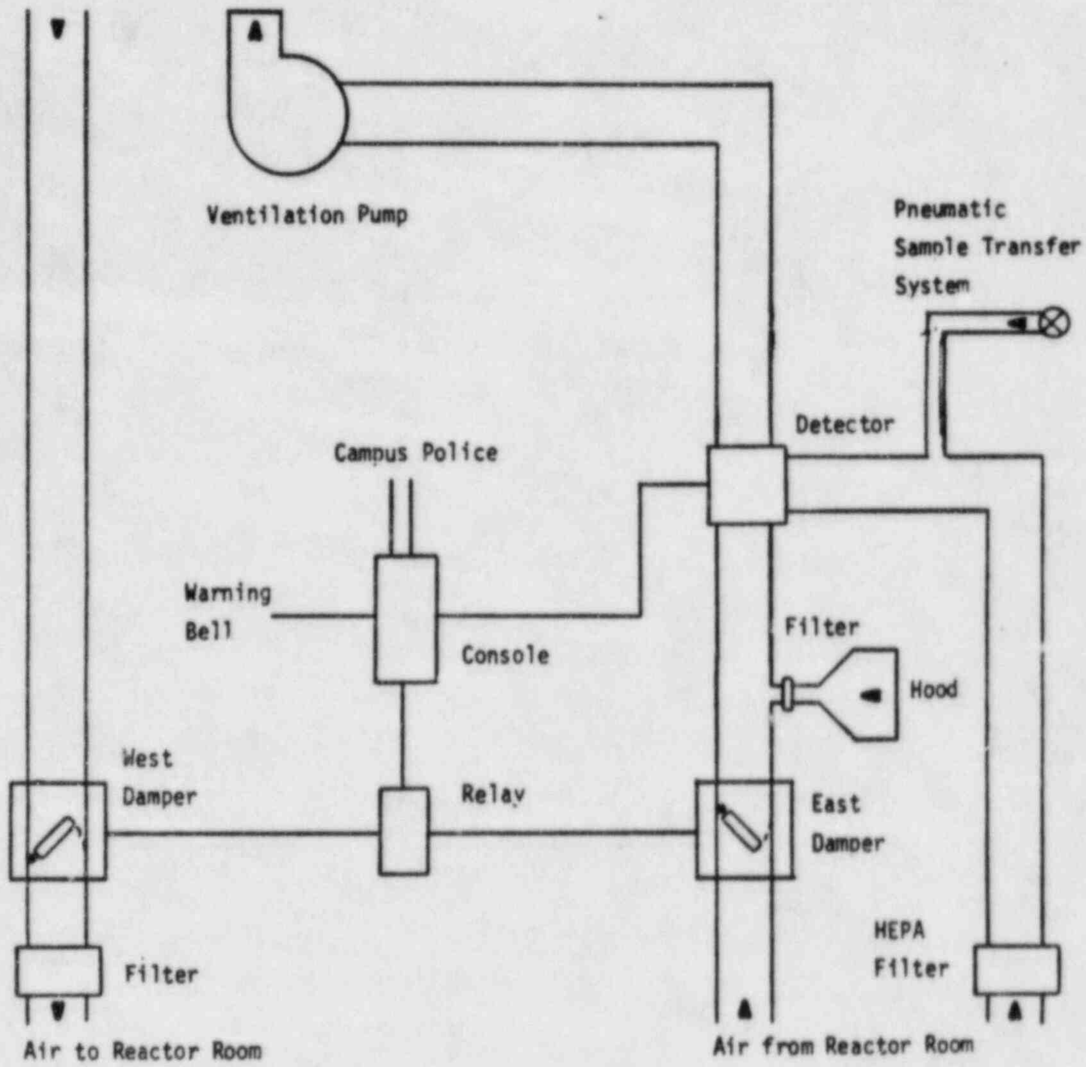
There are no graphite loaded reflector elements in our present inventory.

26. Provide a schematic drawing of the primary coolant circulation loop and the primary coolant purification system. What are the coolant flow rates through the circulation loop and the purification loop?

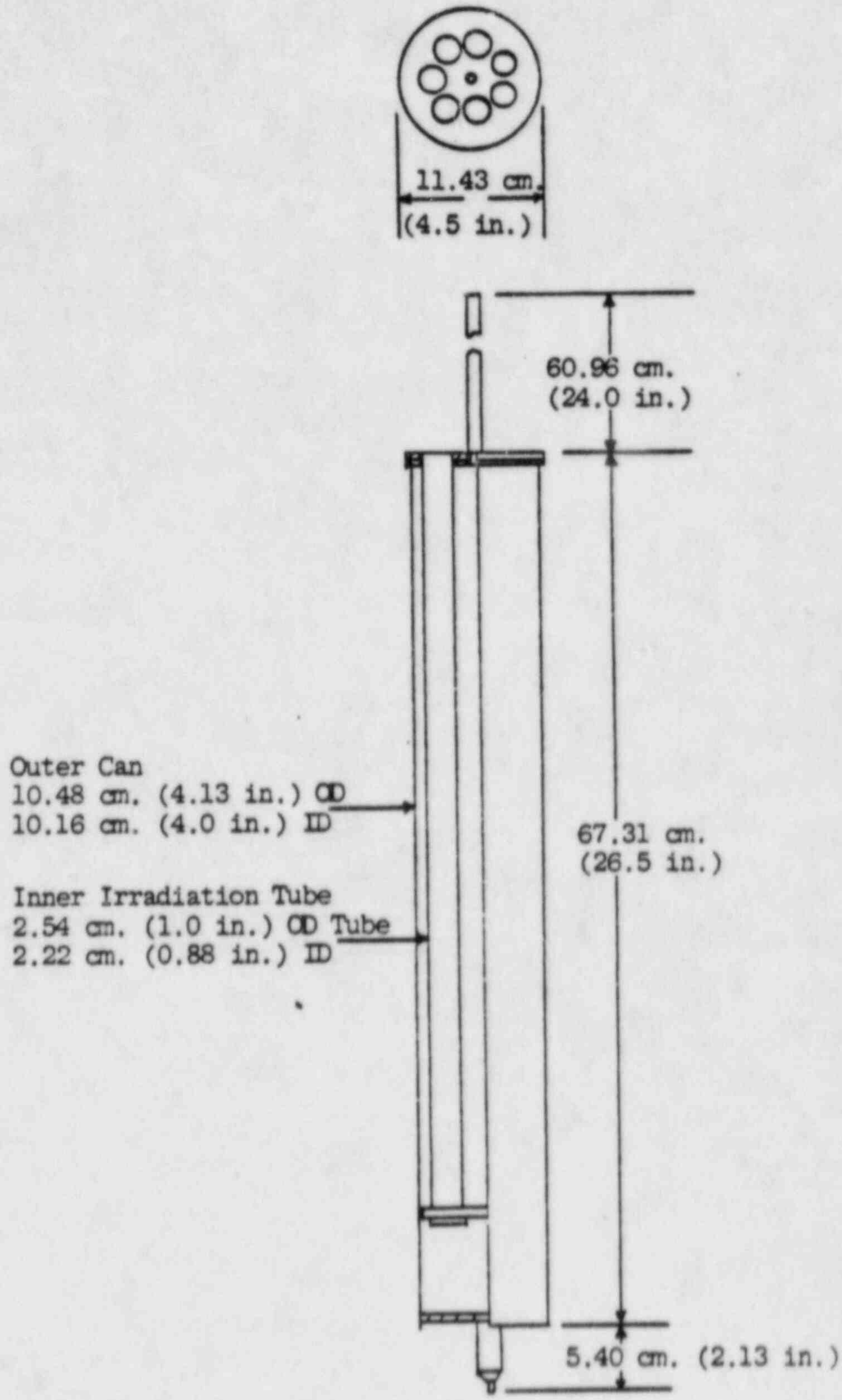
Average primary coolant flow through the main circulation loop is 50 gal per minute and 4 gal per minute for the purification loop. See figure for schematic.

27. What is the concentration of tritium in the primary reactor coolant and the D_2O in the reflector tanks and elements? What is the total volume of D_2O now in the reflectors used in the reactor?

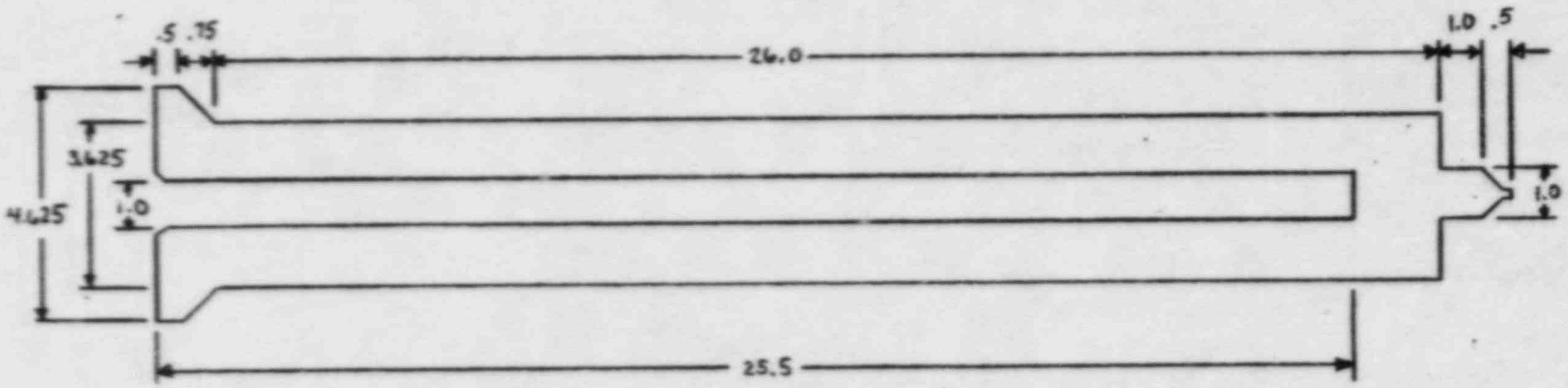
The tritium concentration in the D_2O reflector tanks in the reactor is ±0.04 μCi/ml over the volume of about 60 gallons of D_2O in the core. This represents a total inventory of 30 mCi which if all released would result in an activity in the tank water of about 0.8 uCi/ml which is about 25% of the release limit (3.0uCi/ml) in 10 CFR 20, appendix 3, Table 2, Column 2.



Ventilation Systems



D_2O Filled Central Irradiation Facility



SEPTEMBER 28, 1984 ALL MEASUREMENTS IN INCHES	HEAVY WATER AND AIR IRRADIATORS UNIVERSITY OF UTAH TRIGA REACTOR NUCLEAR ENGINEERING LABORATORY	KEVAN CRAWFORD SENIOR REACTOR ENGINEER

28. Describe the current use of a cooling fan to assist evaporative cooling of the reactor pool water. What is the status of the proposed heat exchanger mentioned in your SAR?

A Cooling fan is employed to provide increased forced air connective cooling for the top of the reactor water tank in addition to the reactor room ventilation system. A heat exchanger system is under development to remove heat from core as optionally provided in the SAR. The freon compressor unit is being installed and the heat exchanger has been order. All activities on the heat exchanger system are under constant review by the Reactor Safety Committee.

29. What are the normal evaporation losses from the pool? Describe the makeup water system at the UUTR.

Normal evaporation losses (depending upon the mean tank water and ambient air temperature) are about 3 gal per day without the cooling fan and about 26 gal per day with the fan.

Makeup water is added by opening makeup water lines and valving off any back-flow paths to force all makeup water through the demineralizer (see figure), pre-filter and post-filter systems.

30. What is the volume of the ion exchangers used in the water purification system? How is the ion exchange material of this system regenerated? What are the procedures for disposing the liquid and solid wastes from the regeneration and/or replacement of the ion exchange resins?

The volume of the mixed resin used to purify the primary coolant water is 2.5 cubic feet. The ion resin is regenerated by a local firm under contract using a standard acid-base reverse wash. The resin is not released to the firm until activity levels when properly diluted are with 10 CFR20 restrictions for release. This surveilence is performed by the Radiation Safety Officer before release.

31. How often are the pH and conductivity of the pool water monitored? What are the procedures used to maintain them within the Technical Specifications limits?

The pH and conductivity of the primary water are monitored at the position shown in the enclosed figure. There are no limits on pH or conductivity under the present Technical Specifications. However, the proposed, new tech specs will require a primary coolant pH between $< 5 \times 10^{-6}$ mhs. This will be accomplished by resin bed regeneration and acid (HNO_3) buffering as required.

32. Describe the heating and cooling devices integrated with the ventilation system in the reactor area.

Reactor area heating and air conditioning is accomplished by the building's general heating and cooling system. However, exhaust air from the reactor area is vented through a independent exhaust system which is continuously monitored by a radiation detector.

33. Provide a schematic drawing of the ventilation system for the reactor area, including the emergency filter arrangement and radiation monitors.

See attached figure.

34. Describe the system that supplies compressed air to the reactor area.

Compressed air is delivered to the reactor area from the building's air compressor and holding tank system. The air is filtered and dried before delivering to users.

35. Describe the fire protection system at the reactor facility. Which fire department serves the fire-related emergencies at the reactor facility? Who is reponsible for maintaining fire protection equipment at the reactor facility.

The building is fully outfitted with overhead sprinklers and near the reactor area is a fire-water line, chemical extinguishers and also installed are smoke alarms in the reactor room. The University of Utah has a fully equipped fire station on campus which would respond to fire-related emergencies in the reactor area. The University fire marshall is responsible for surveying and maintaining all fire equipment on campus

36. Describe the central irradiation facility of the reactor and the different types of irradiators now in use at this facility. Provide the dimensions and schematic drawings of these facilities.

There are three different irradiators used in the TRIGA core central irradiator location. Attached drawings provide dimensioned drawings of these irradiators. Two are filled with D_2O (one has a 7 hole concentric row for neutron sample irradiation and the other has a single central hole surrounded by D_2O) and a third irradiator is air filled to provide a fast neutron spectrum for the sample. The reactivity difference between the D_2O filled irradiator and the air filled irradiator has been measured at \$0.62. For H_2O in the irratiator this reactivity difference is less than \$0.15.

37. Describe the wet and dry irradiation facilities at the trapezoidal D_2O reflector tanks. Provide their dimensions.

There are 6 trapezoidal, D_2O filled reflector tanks each 6 inches wide surrounding the TRIGA core. Four of these tanks are completely D_2O filled with no irradiation fixtures installed. The other two tanks do permit insertion of samples for neutron irradiation. One tank has a large cylindrical cavity which will accomodate any one of the irradiators described in question. The other tank has three, 1 inch diameter openings which can be voided of water and prepared to provide a dry irradiation space. Existing procedures and monitoring activities control induced airborne activity that arises for such irradiations.

38. What are the volumes and Curie-contents of the solid and liquid wastes generated annually at the reactor?

The average annual volume and Ci content of solid and liquid waste is

solids: 1 cubic foot at 10 μCi
liquids: 100 ml at 1 μCi

typically these materials are held for decay, diluted and discharged or packaged and sent out for approved disposal.

39. What is the dose rate at the pool surface attributable to ^{16}N and ^{14}Ar when the reactor is operated continuously at the highest power level?

The radiation field at the top of the tank attributal to N-16 and Ar-41 is 0.1D mrad/hr. This value is in excess of the natural background value in the area of 0.012 mrad/hr.

40. What is the estimated annual release of ^{41}Ar from the reactor facility to the environment.

The air monitoring system can detect down to a limit of 1/3 of Ar-41 limits found in 10 CFR20 APP. B TAB. II. Since no Ar-41 has been detected, the maximum Ar-41 release cannot exceed a value calculated from 1/3 of the table value.

$$\left(\frac{4\times10^{-8}\ \mu Ci}{3\quad ml}\right)\left(\frac{1800\ ft^3}{min}\right)\left(\frac{5.26\times10^5\ min}{yr}\right)\left(\frac{2.83\times10^4\ ml}{ft^3}\right)\left(\frac{3.7\times10^4\ dis.}{\mu Ci\text{-}sec}\right)$$

$$\left(\frac{6.59\times10^3\ sec}{\ln 2}\right) \quad X \quad \left(\frac{41 grams\ Ar\text{-}41/mole}{6.023\times10^{23}\ atoms/mole}\right) = \left(\frac{8.6n\ grams\ Ar\text{-}41}{year}\right)$$

The maximum yearly release would then be 8.6×10^{-9} grams of Ar-41. However, since the facility only produces Ar-41 while the reactor is running, a more realistic calculation follows.

$$\left(\frac{8.6\times10^{-9} g\ Ar\text{-}41}{year}\right)\left(\frac{6\ hrs\ operation}{168\ hrs}\right) = \frac{3.1\ \times10^{-10} g\text{-}Ar\text{-}41}{year}$$

Annual Release of Ar-41 would be less than 3.1×10^{-10}g.

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

The radiation protection program at the University of Utah is administered by the Radiological Health Office with a Radiation Safety Officer. The attached figure provides the administrative organization structure. The Institutional Council of the University of Utah is the licensee for the TRIGA reactor to the NRC. However, the Reactor Administrator is the primary point of contact with the NRC in behalf of the Institutional Council. The reactor Safety Committee and Radiological Health Committee serve as review and approval bodies for all activities which are under the primary control of the Reactor Supervisor. The Reactor Supervisor is responsible for all reactor operations and operators in the Nuclear Engineering Laboratory.

42. Describe the responsibilities of the Radiation Safety Office staff at the reactor facility. Identify the radiation safety related tasks that are performed routinely by the reactor staff.

The Radiation Safety Office under direction and review of the Radiological Health Committee is responsible for all radiation safety activities associated with the TRIGA reactor. The reactor staff provides routine monitoring functions and personnel dosimetry coverage for personnel and activities at the reactor facility. However, all radiation safety activities are subject to review by the Radiation Safety Officer.

43. Describe any radiation protection training for the non-Health Physics staff. If possible, provide a topic outline of the courses and indicate the normal duration of each course or lecture.

The Radiation Safety Office provides periodic training in radiation technology for all campus users. Reactor staff and students who need training and review of radiation related matters are requested by the Reactor Supervisor to attend lectures and laboratory work. These courses include instruction on radiation physics, measurements, regulations, procedures and practice; and generally require about 10 contact hours of instruction.

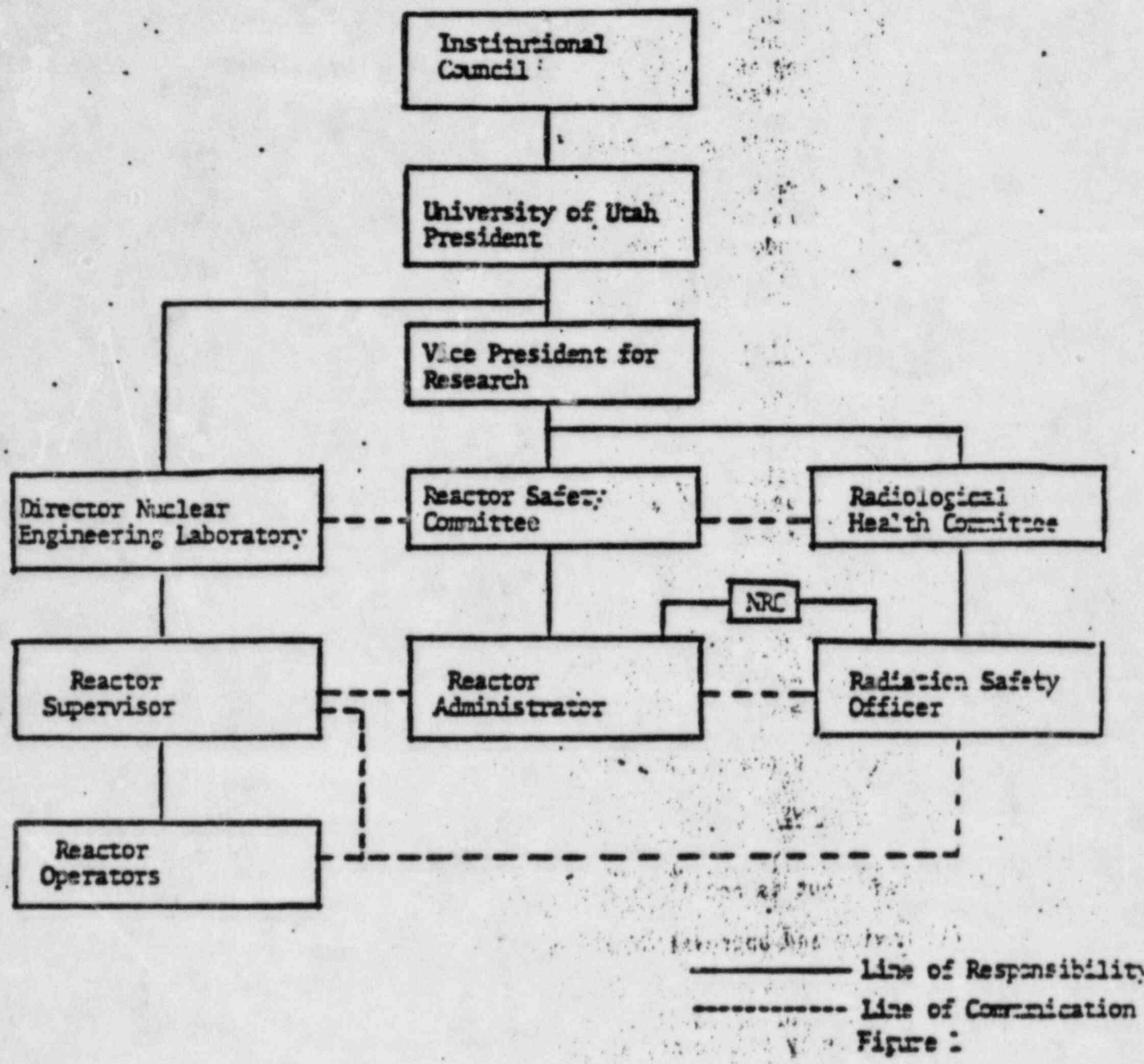


Figure 6.1
University of Utah Administrative Organization for Nuclear Reactor Operations

44. 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.

The Radiological Health Committee exercises review authority and seeks to insure. Compliance for all federal and state regulations. This committee reviews and identify needs and actions appropriate to ALARA activities. See the Radiation Safety Manual supplied to the NRC review team for specifics.

45. Describe the gaseous effluent monitoring equipment with respect to location, stack flow rate, and probe geometry. For the fixed-position radiation and effluent monitors, specify the generic types of detectors nad their efficiencies and operable ranges, and describe the methods and frequency of instrument calibrations and routine operational checks.

The gaseous effluent form the reactor is exhausted through a fixed volume, radiation monitoring chamber which is equipped with a calibrated, G-M tube. A continuous strip chart recorder provides a permenant record of activity released through the dedicated exhaust to the roof of the building. The area monitoring ranges between 10^{-2} to 10^{-3} mr/hr. The continuous air monitor ranges from 100 to 100,000 counts per minute and monitors a control volume of 250 liters and operates at 0.1% efficiency (i.e., the detector measures 0.1% of the total gamma activity arising from a uniformly distributed gamma source, Ar-41, in the control volume). Routine operational checks are perfor checkout procedures. Calibration is performed once a month during the Monthly Checks Procedure.

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

The area radiation monitors consist of 1) the above core tank monitor, 2) the primary water monitor and 3) the exhaust air monitor. These monitors have low level set points of approximately 0.1 mrem/hr and a high level set point of 1.0 mrem/hr. A low level alarm implies an elevated radiation field in the vicinity of the monitor and corrective action is required. A high level alarm results is a scram of the reactor and further reactor operation is prohibited until the high radiation source is identified and corrected.

47. 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.

Portable health physics equipment inventory is

- Gamma Scintillation Detector

  range: 0 to 3 mR/hr
  sensativity: 1 micro R/hr

- Geiger Counter

  range: 0 - 2 R/hr
  sensativity: 0.01 mR/hr

- Neutron Bonner Sphere
  Texas Instruments

  range: 0 - 1 rem/hr
  sensativity: 0.1 mem/hr

- Self reading pocket dosimeters
  various manufac

  range: 0 - 200 mrem
  sensativity: 5 mrem

all health physics instruments are calibrated semiannually.

48. Describe your personnel monitoring program.

The personnel monitoring program at the nuclear reactor facility includes the following respects:

a) Familiarization with the facility, the emergency procedures and personnel monitoring program.
b) Issurance of a TLA dosimeter which is changed monthly, creation of a personal radiation history and record.
c) Issurance of self reading dosimeters to personnel during reactor operations.
d) Periodic urinanalysis of personnel involved with handling of isotopes that pose internal hazards.
e) Monthly issuance of individual radiation exposure records and review of above minimum exposures for ALARA compliance.

49. Describe your environmental monitoring program and summarize the results for the past 5 years.

See enclosed environmental report provided as part of the applicants relicensing application.
The applicant also has a complete environmental radiation monitoring station installed outside the reactor laboratory. This station includes:

a) Moble gas monitor (eg. Kr, Xe, etc)
b) Tritium sampler
c) Air borne particulate monitor
d) Activated charcoal sampler (eg. I-13)
e) Pressurized Ion Chamber for gross gamma survey
f) TLD's for gamma field dose integration

50. Comment on the ability of 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.

The reactor components and systems have proven reliable in general over eight years of operation. Those systems and components that have experience undue wear or faculty performance have been modified ore replaced as necessary to provide safe, reliable operation. The radiation monitoring system has been upgraded and modernized as necessary. The present reactor console is being upgraded with modern digital electronic subsystems as necessary and after review and approval of the Reactor Safety Committee. The incore neutron detectors are eight years of age or less and have operated reliably and safely without any malfunctions. Rack and pinion control rod drives have been installed and are being tested for acceptance and operation. With the installation of an IBM series 1 Digital Computer, more accurate, frequent and extended monitoring of operational and safety related parameters and variables will provide enhanced training and improved control of the reactor. The University of Utah TRIGA serves as a major training resource for the nuclear program and it is the intention that students be subjected and trained in the most current and safety concious technology we can provide. It is anticipated that over the renewal license period the reactor system will continue to be expanded and improved to meet both technical and education demands and NRC requirements. Fuel resources and their integrity are deemed adequate to satisfy core fueling requirements for the foreseeable future.

51. What is the maximum rate capacity of the ventilation system airflow and what is the flowrate when the HEPA filter is in line?

See the attached figure for the ventilation system under question 33 in the reactor facility. The maximum airflow through the system is 1740 cfm with the HEPA filter off line and 150 cfm with the filter on line.

52. Provide a copy of the proposed Technical Specifications with a maximum operation of 100 kW. Provide a more applicable basis than you have for the maximum safe operating power of Al-clad, low-hydride UZrH fuel.

A copy of the proposed Technical Specifications is enclosed. The TRIGA Reactor at the University of Utah operates in a steady state mode only and pulsing operation is not permitted under the present license or proposed license renewal. The proposed Technical Specifications (3.2(4) and (5)) restrict excess reactivity worth under any assumed operating condition to \$3.00 or less so this is assumed to be largest prompt reactivity insertion that could conceiveably occur. Furthermore, the maximum fuel temperature that has been monitored with the instrumented fuel element at 100 kW power is 112°C. The conservative assumption made for low hydride, aluminum clad fuel so as to prevent fuel distortion and possible cladding failure is that the fuel should not exceed the phase change temperature in zirconium hydride from $\alpha + \delta + \gamma$ to $\alpha + \beta$ Phase. This phase change is known to not occur at a temperature lower than 530°C (reference GA-7882, Kinetic Behavior of TRIGA Reactors, March 1967; K.E. Moore and W.A. Young, "Phase Relationships at High Hydrogen Content in SNAP Fuel System," USAEC Report NAA-SR-12587, Atomics International, 1968; K.E. Moore and M.M. Nakata, "Phase Relationships in Alpha-plus Delta Region of the Zr-H System", USAEC Report AI-AEC-12703, AI, Sept. 1968.) Futhermore at achieving this temperature in the fuel the fraction of release of fission products is given by

$$f = 1.5\times10^{-5} + 3.6\times10^{3} \exp\left[\frac{-1.34\times10^{4}}{T}\right]$$

where T is mean fuel temp in °K above 530°C or 803°K

Reference is Baldwin, N.C., Foushee and Greenwood, "Fission Product Release from TRIGA- Low Enriched Uranium Fuels" GA TOC-12, 7th Biennial U.S. TRIGA User's Conference, March, 1980.

The University of Utah's SAR details an analysis of the fuel temperature reached in a LEU TRIGA fuel element and shows the core averaged fuel temperature at the end of a \$3.00 pulse to be 297°C. From experimental data obtained in the University of Utah TRIGA reactor, the hottest possible fuel element in the B-ring of the minimum fuel core (which leads to the highest peaking of power in the core) is 1.45 so

$$T(\text{max fuel temp}) = 1.45 \times 297°C = 431°C$$

This anlaysis neglects any transfer of heat from the fuel to the primary coolant during the pulse and assumes an instaneous step change in reactivity of $3.00. Futhermore, the heat capacity of the TRIGA fuel is not constant, but increases with temperature and has been shown to be at 400°C about 1.8 times it value at 0°C. The assumption of a constant specific heat overestimates the core fuel temperatures by as much as 10%. (Reference GA-7882, Kinetic Behavior of TRIGA Reactors, March 1976.) Furthermore, experimental measurements from a thermocouple mounted near the center of the fuel for $3.00 step insertion yielded a fuel temperature of 405°C (reference GA-7882). Furthermore, operational experience gained over many years with aluminum clad, low hydride TRIGA fuel elements demonstrates the high degree of safety and integrity which such fuel exhibits particularly in a non pulsing facility such as the UUTR.

In the course of experimental program carried out on the prototype TRIGA Mark I reactor at General Atomic, three cladding failures have occurred. The first took place in August, 1959, in a special thermocouple-equipped fuel element. It is believed that vaporization of volatile material in the insulation of the thermocouple wire created a local region of high pressure between the fuel rod and the cladding material and caused the failure.

In November, 1959, a cladding failure was detected in an aluminum clad fuel element. When the failure occurred, this element had been subjected to a total of 709 pulses, 410 of which were at or above 3 dollars. A subsequent investigation indicated that the failure was caused by ratcheting of the aluminum cladding on the fuel body. Further research indicated that this ratcheting, as evidence by progressive lengthening of the element during a series of pulses.

The second and third failure (May, 1960) were both caused by ratcheting of the cladding material. In all three cases the release of radioactivity to the surrounding atmosphere was confined to essentially insignificant quantities of short-lived noble-gasses. It seems very conservative to assume that for the small inventory of low hydride-aluminum clad fuel (17 elements in the present core all of which are in the F or G ring only) compared to the 70 stainless cladhigh hydride fuel (in the C-D-E and F rings) that a highly improbable single pulse of reactivity even up to $3.00 would lead to any major damage of the aluminum cladding that would result in a significant relase of radioactivity. Futhermore, the complete cladding failure of a single fuel element has been previously analyzed in question (24) and the consequences we deemed not to pose undue hazards to the staff or public.

53. What is the effect of radiation damage and aging on the electrical lines that lead to the control rod magnets?

The electrical power lines that provide power to the electromagnets for control rod drive are at a minimum distance of 4 feet above the top of the core. At this position the neutron flux and gamma dose are estimated to be 6×10^3 nts)cm^2 sec and 250 rad/sec respectively. The integrated neutron flux and gamma dose over 20 years of future operation at anticipated operating history of 10 hrs per week in 1.8 x 10^8 rads. This dose is far below the damage threshold for metals such as copper and aluminum and less than 1% of the threshold damage to synthetic materials used as insulation for electrical cable. Futhermore, failure in any of the electrical power cables, would simply result in a reduction or loss of magnet force at the rod inteface and the rod would drop into the core, shutting down the reactor as a consequence. Thus, power line failure results in a fail safe, shut down of the reactor. Furthermore, a rack and pinion drive for the control rods is being installed and only the control rods and aluminum extensions will be exposed to radiation.

54. What is the capacity of the overhead crane? What are the administrative and physical limitations on its use over the reactor?

The overhead crane is rated at 4000 pounds and is employed in the facility for moving the fuel storage. Cask and other heavy objects. Because of the potential risk posed by the risk of dropping a heavy weight into the core with attendent fuel damage, the switch box which powers the crane is locked at all times except when authorized by the reactor supervisor for use. The crane is not to be operated during reactor operations. Futhermore, the heavy, structural beam members which support the control rod drives abovethe reactor core shield the core and are assessed to be capable of withstanding a vertical load dropped from the highest point of the overhead crane carraige. Finally, the control rods are securely attached by bolts to the core assembly and any object dropped on the core would result in a subcritical core array as a result of water floding and fuel relocation.

The restriction of no crane operations during reactor operation, and crane operations only under direct supervision of the reactor supervisor and a normally locked crane power supply box provide assurance of acceptable low risk potential for the core.

55. Where applicable, provide updated information and discussion on demography local industrilization, hydrology, meteorology, etc. Include student population information.

56. In your SAR and Technical Specifications, you have referenced relatively old reports. Please use more up to date references where applicable, and especially where significant differences exist.

The SAR and the proposed Technical Specifications reflect recent reference and testing data.

1. Introduction

This report supports an application to the United States Atomic Energy Commission (AEC) by the University of Utah (U/U) at Salt Lake City, Utah, for the construction and utilization of a TRIGA* Mark I nuclear reactor. The reactor utilizes the fuel elements[1], the control console, the control rods and drives, and other minor components of the TRIGA Mark I 100 kw reactor previously located at the University of Arizona in Tucson, Arizona. These components were given by the University of Arizona to the University of Utah, who with additional fuel supplied by the AEC proposes to construct and utilize a modified TRIGA Mark-I 100 kw (thermal) reactor which employs a water reflector, similar to that of the TRIGA Mark III, rather than the standard graphite reflector.[2] Because the reactor fuel is typical of that in most TRIGA's and the proposed system has a well established operating experience, no new significant research and development activity has been required in designing this system. It has been possible to use accepted, standard safety analysis techniques with regard to evaluating the characteristics of this facility. The reactor will be administered and operated by the University of Utah for education, research, and irradiation services. However, it is deemed that the proposed facility will benefit all interested parties within the regional area.

The proposed reactor will be located in the Nuclear Engineering Laboratory (ground floor) of the existing Merrill Engineering Building where an AGN-201M 5 watt reactor is presently located and has been operated safely for 13 years.

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* Trademark registered in U.S. Patent Office

1. Additional fuel elements required to realize a critical system will be obtained from Gulf Energy and Environmental Systems Inc.
2. The U/U proposes to apply initially for operation at 100 kw and after acquiring operating experience and evaluating the reactor's performance at 100 kw to apply for an ammendment if warranted for operation at 250 kw.

and it is deemed that there is no danger from gamma radiation to persons in this second floor area.

4.3 Third Floor (Figure 4.1.3)

The third floor above the reactor area is comprised of office and classroom space. Maximum dose rate levels on this floor are determined to be 0.004 mr/hr (0.010 mr/hr at 250 kw) which is well below natural background levels.

4.4 Reactor Area (Figure 4.1.1 - Rooms 1001 A - G)

The reactor area comprises seven rooms (Figure 4.4.1) a supervisor's office, control room, laboratory classroom, counting laboratory, neutron generator office, radiochemistry laboratory, and the reactor room. Entry to the reactor room from inside the building is restricted to one door in the control room. The reactor room has direct access through a 12 feet wide overhead door to the outside loading area. This overhead door and the doors between reactor and control room form the confinement enclosure and will be tape sealed for this reason. The reactor room and the radiochemistry room will be treated as a single unit for ventilation and safety-confinement purposes.

The walls in the area are of one hour fire resistance standard, plaster and metal stud construction except the west wall which is an exterior reinforced concrete wall.

Visibility of the reactor room, laboratory classroom, the supervisor's office, and the control room will be complete because of the large windows to these rooms. Nonporous enamel paint finishes are on all walls and the ceiling of the reactor area for ease of clean-up.

4.4.1 Reactor Room (42 ft x 23 ft x 20 ft high - Room 1001 E)*

The reactor core will be situated near the bottom of a water-filled tank located as shown in Figure 5.3.1. The cylindrical tank has aluminum walls, 5/16 inch thick on the sides and a 1/2 inch thick bottom. The tank and liner will be constructed similar to the design and specifications of Gulf Energy and Environmental Systems, Inc. for their Mark I TRIGA installations. The design is considered to be more than adequate to endure earthquake shocks. A detailed examination of the aluminum tank will be made and hydrostatic pressure leak tests will be performed with the tank in place on its foundation pad.

Three existing fuel storage pits (Figure 4.4.1), each 10 feet deep, with aluminum liners and surrounded by concrete have been approved and licensed by AEC and are serviced by an overhead crane. The crane travels on an overhead boom and is accessible to any location in the reactor room. The crane has a 4000 pound capacity and is locally controlled from a pendant box.

4.4.2 Neutron Generator and Irradiator Facility (11 ft x 17 ft - Room 1001 B)

South of the laboratory classroom is the neutron generator and irradiator room. The walls of this room are 12 inch thick concrete. A 3 feet wide by 6 feet long by 10 feet deep pit is provided for a TMC-211 neutron generator $T(d,n)He^4$ which produces 14 Mev neutrons (Figure 4.4.1). The tritium target assembly is at the bottom of the pit and a pneumatic sample transfer system enables samples to be irradiated. A large movable concrete and hydrocarbon shield covers the pit and shields the room from neutrons. The high voltage transformers and cooling units for the generator are

* An AGN-201M 5 watt Nuclear Reactor (License Number R-25) used for teaching and training purposes is also located in the reactor room. However no neutronic or hazard coupling is considered credible between the TRIGA and AGN. All credible hazards associated with the AGN have been previously reviewed by the AEC.

located in the room. The control console for the generator is located in the laboratory classroom.

There is little danger from tritium leakage from the generator. In the event of an accident, the exhaust system is sufficient to prevent any overdose to operating staff.

In addition to the fast neutron generator, the facility will include a Cf-252 neutron irradiator. This Cf facility is fully licensed* by the AEC and its potential hazards have been previously considered in a Safety Analysis Report submitted to the AEC. No nuclear coupling or interaction of this irradiator facility with the reactors is possible because of the separation distance (40 ft.) involved.

4.4.3 Supervisor's Office and Control Room (16 ft x 14 ft - Room 1001 D)

The control room will contain the reactor control console and will have good visual access to the reactor. Area monitor instrumentation will be located adjacent to the reactor control console. Portable monitoring equipment will be retained here. The only nonemergency access to the reactor room is through the control room except under special circumstances when the doors to the loading dock will be opened; these doors will not be opened while the reactor is operating. The door between the reactor room and the control room will have seals and will be provided with a closer so that it will normally seal the confinement area.

4.4.4 Radiochemistry (10 ft x 20 ft - Room 1001 F & G)

Two stainless steel fume hoods are located in this laboratory. The venting for these two hoods has a rated capacity of 1800 cfm and is separate from all other ventilation from the building and exhausts above the penthouse level on the roof of the building. This independent venting system will serve as the exhaust system for both the radiochemistry hoods and the reactor ventilation system.

* By-Product Material License No. 43-01884-15

It is not intended that moderate or high levels of radioactivity shall be processed without special provision for shielding and waste disposal. The reactor administration will require that due provision be made before authorization of production of such quantities of activity.

4.4.5 Counting Laboratory (22 ft x 16 ft - Room 1001C)

The pneumatic sample transfer system and counting facilities will be located in this room. The pneumatic terminus will be provided with an exhaust system connected to the reactor room exhaust.

4.5 Air Confinement

As mentioned previously, the reactor room and radiochemistry laboratory are designed to act as confinement area for possible contaminated air. All penetrations of walls, ceilings and floors by pipes and ducts have been specially sealed. The large doors to the loading dock and the door to the control room have weather seals. The ventilation system is designed to maintain a negative air pressure* in this area both with respect to the outside and to the adjoining areas during reactor operation.

4.6 Ventilation System

4.6.1 Normal Operation

The ventilation system proposed for this area is shown in Figure 4.6.1. A schematic of the control system is given in Figure 4.6.2. In normal operation fresh air is drawn into the building at the roof level and after appropriate processing (filtering, heating or cooling) is distributed around the building. A local thermostat controls dampers at the entrance to the area. Temperatures and static pressures in the main supply ducts are

*Minimum negative air pressure requirements are as follows:
1. 0.15 in. water with inlet vents closed and exhaust vent opened.
2. 0.10 in. water with inlet vents closed and exhaust vent dampered for HEPA filter flow.
3. 0.01 in water with inlet vents and exhaust vent opened.

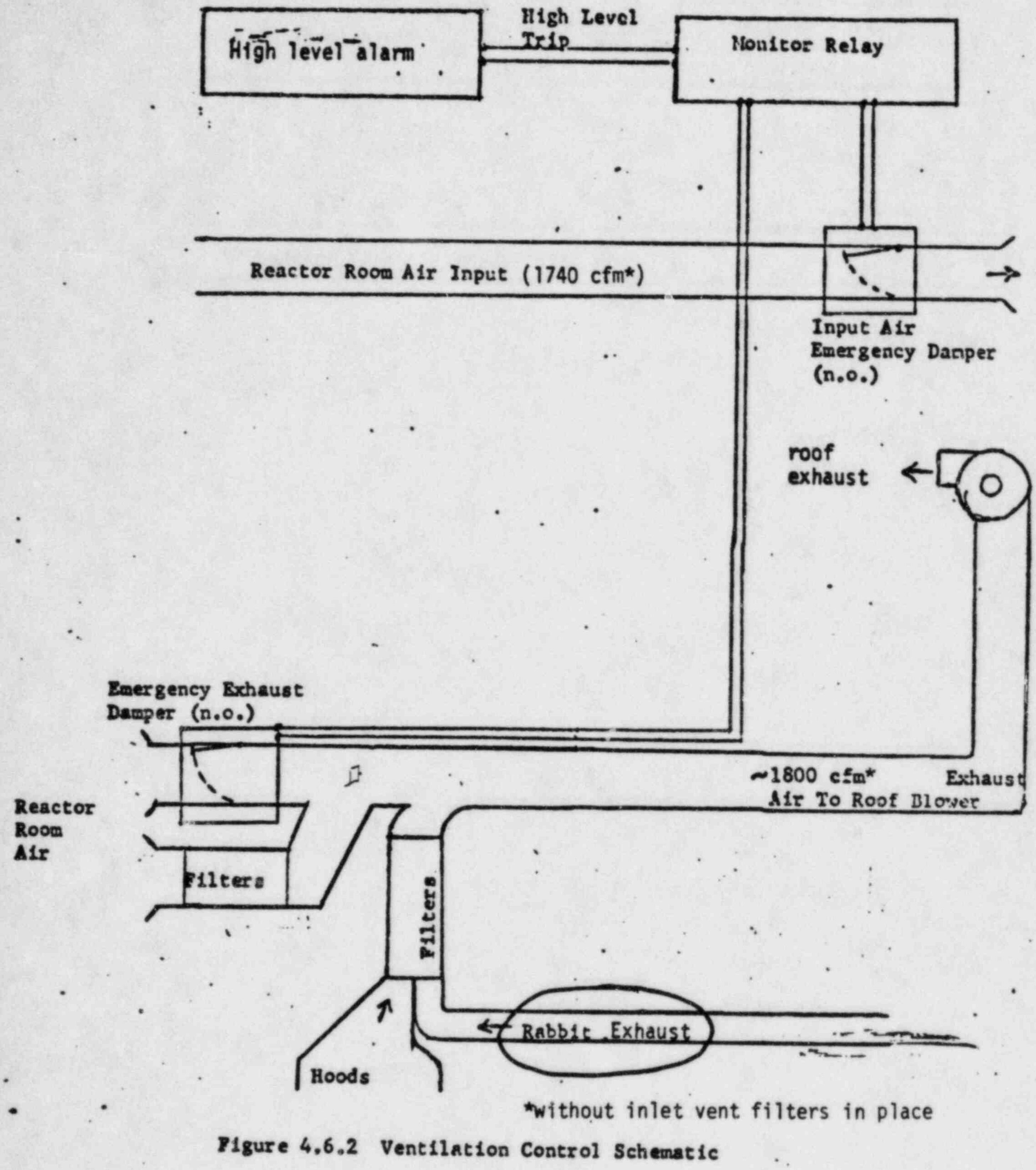


Figure 4.6.2 Ventilation Control Schematic

maintained by automatic controls. The complete system is under the control of the Physical Plant Center which is manned 24 hours a day.

In the reactor area system, in addition to the control dampers, additional static pressure control and emergy isolation dampers are provided. The first is designed to regulate the supply air to the reactor room so that the negative pressure requirement is met; the second operates in emergency status as described below.

Air discharge from the reactor room is accomplished through the laboratory exhaust vents in rooms 1001 F & G. These exhaust vents will operate at all times during reactor operation and move sufficient air to insure at least four changes of air per hour throughout the reactor room (MEB 1001E). These vents are powered by a fan located on the roof of the building and the discharge is located 10 feet above the penthouse roof.

4.6.2 Emergency Operation

In the unlikely event of an accidental release of particulate or gaseous radioactivity into the reactor room air, it is desirable that this be exhaused slowly to the atmosphere to allow dilution, decay, and filtration to reduce activity. Thus an emergency purge exhause, actuated by the air monitoring system, will be employed.

Isolation dampers will close off the air supply to the reactor room. The outlet vent will be equipped with a by-pass HEPA filtration system which will be dampered into operation. This will provide a continuous filtered exhaust and maintain the reactor area (Rooms 1001 E, F & G) under negative pressure. The flow under filtering conditions will be such that a negative pressure of at least 0.1 inches of water will be maintained.

4.7 Radiation Monitoring System

4.7.1 Continuous Air Monitor-Operation of Emergency System

Experience with the University of Illinois TRIGA reactor[1] indicates that in the case of the most probable accident, namely, leakage of fission products from a fuel element, the release is more rapidly detected by a continuous air monitor operating alongside the reactor than by a monitor in the exhaust tank. Accordingly, for the U/U TRIGA, the emergency purge system will be actuated by the high level radiation alarm with detectors in the air exhaust system and at the water tank. The unit provides audio-visual alert and alarm levels; the latter will be set to actuate the emergency purge system described previously. This unit must be operative at all times and a periodic testing procedure will be a part of the written operating instructions for the reactor and the periodic system inspection schedule.

4.7.2 Area Monitor

An area monitoring system will be installed with control and readout adjacent to the reactor control console. Stations will be provided in four locations.

1. Ceiling immediately over reactor core (also "sees" fuel storage pits)
2. Reactor tank
3. Radiochemistry Laboratory
4. Neutron Generator & Irradiator Facility

The object of these locations is to maintain a check at the most likely positions of release of radiation to uncontrolled areas. Local warning

1. Nuclear Reactor Laboratory, University of Illinois, Report INRL-21, October 24, 1966.

and readout will be provided in addition to central control. As with the continuous air monitor, a program of testing and calibration will be initiated. The instrument itself provides for detector failure alarm by the use of "live zero".

The area monitoring system will be powered by an independent battery powered supply.

5.0 Reactor

5.1 Reactor Design

The TRIGA reactor to be constructed and installed at the University of Utah is similar to that which operated for 13 years at the University of Arizona from 1958 to 1971. The core fuel elements are largely identical, having been transferred from Arizona to Utah. Therefore, much of the reactor description given in the Safety Analysis Report for the original Arizona reactor (Hazard Report for a Training, Research, and Isotope-Production Nuclear Reactor at the University of Arizona, GA 413, June 1958) is applicable to this reactor. There are, however, a few significant differences, as follows:

1. Grid plate and lattice arrangement.

   The Arizona reactor had a grid plate providing a modified hexagonal-circular lattice characteristic of most of the early TRIGA's. However, the University of Utah has decided to adopt the uniform hexagonal lattice with 1.72 inch pitch of the TRIGA at the University of California at Santa Barbara (Figure 5.1). The upper and lower grid plates have been fabricated in the machine shop at the University of Utah using Gulf Energy and Environmental Systems, Inc. drawings and specifications. The grid plates are virtually identical to those manufactured by Gulf Energy and Environmental Systems, Inc. and installed in the TRIGA at Santa Barbara, California.

2. Reflector.

   The solid graphite reflector of the Arizona reactor will be replaced by either an all water or a combination heavy water, graphite, water reflector. Up to 47 dummy fuel elements containing graphite or heavy water in aluminum cans may be used in the outer F-ring of lattice, which has fuel element positions 62 to 91 or for use in the G-ring, position numbers 92 to 127, or in other positions in the core. These dummy elements will be aluminum tubes, 1.47 inch O. D., packed with powdered graphite or heavy water over the core length centered at the active core midplane. The use of the dummy elements will slightly reduce the required fuel loading of the reactor and will thus enable the reactor core to accomodate various irradiation capsule experiments. Furthermore, several dummy sample irradiation elements fabricated from radiation resistant plastic (to minimized induced activity) will contain inserts that can be used for the insertion of irradiation capsules. The dummy elements so equipped can then be inserted in place of fuel elements in different regions of the reactor.

The characteristics and operating parameters of this reactor have been calculated and extrapolated using experience and data obtained from existing TRIGA reactors[1] as bench marks in evaluating the calculated data. There are several TRIGA systems with essentially the same core and reflector relationship as the proposed U/U TRIGA so the values presented here are felt to be accurate to within 10%. The hexagonal fuel element array is identical to that being used by the University of California at Santa Barbara, and hence the characteristics of that reactor should be quite similar. Though the Santa Barbara reactor uses stainless steel clad elements vs. the mixed stainless and aluminum clad elements of this reactor, the thermal utilizaiton of the two is nominally equivalent due to a slightly different loading of fissile material. Table 5.1.1 gives nuclear parameters for the proposed U/U TRIGA core.

1. E.g. University of California at Santa Barbara and Irvine, Michigan State University, University of Arizona, and Reed College.

TABLE 5.1.1

TRIGA CORE NUCLEAR PARAMETERS

Cold, clean critical loading	59[1] elements 2.1 kg U-235
Operational loading with used (partly burned) fuel elements both aluminum and stainless steel clad	80[1] elements 2.8 kg U-235
ℓ Prompt neutron lifetime	39 μsec
β Effective delayed neutron fraction	0.0070
Prompt temperature coefficient	$-1.0 \times 10^{-4} \Delta k$ per °C
T_f Average centerline fuel temperature at 100 kw	100°C
T_w Average water temperature at 100 kw	27°C (with 16°C pool temperature)

[1] Includes 3 control rods

The following are reactor physics parameters as described by the 2-group equation. These are presented here merely for reference since calculations of reactivity effects are currently performed with multigroup computer codes.

n Neutrons released per absorption in U-235	2.0730
f Thermal utilization	0.7241
p Resonance escape probability	0.877
ϵ Fast fission factor	1.052
k_∞ Infinite multiplication factor	1.3845
τ Fission neutrons to thermal energy	22.23 cm^2

TABLE 5.1.3

ESTIMATED CONTROL ROD NET WORTHS

	Worth (%Δk)
C ring--regulating, 0.25 in. diam.	0.5
D ring--shim, 0.375 in. diam.	2.0
D ring--safety, 0.375 in. diam.	2.0
Total	4.5

The maximum reactivity insertion rate is that associated with a safety rod which can be fully removed from the core in 2 1/2 minutes producing a reactivity insertion rate of 0.019% Δk/sec. at mid-travel.

The total reactivity worth of the control system is about 4.5% Δk. With an estimated core excess reactivity of 2.25%, the shutdown margin with all rods down is approximately 2.2% Δk and with the most reactive rod stuck out is 0.2% Δk. All of these values are quoted from the cold, clean (no short-lived fission products) condition.

The reactivity worth of the fuel elements is dependent on their position within the core. Table 5.1.4 indicates the average values that are expected in this reactor. (Data from University of California, Santa Barbara, Safety Analysis Report) Though these data are for stainless steel fuel elements, the worth of an aluminum clad element differs only slightly because of the lower fuel loading of the latter.

TABLE 5.1.4

ESTIMATED FUEL ELEMENT REACTIVITY WORTH COMPARED WITH WATER AS A FUNCTION OF POSITION IN CORE

Core Position	Worth (%Δk)
B ring	0.98
C ring	0.87

reactors with much higher operating limits than will apply to this reactor.

This TRIGA system will not be operated in the pulsed mode, hence, there are only three major operating considerations and are major non-operating conditions used to define the reactor design bases:

1. Fuel temperature
2. Clad temperature
3. Reactor power
4. Retention of water in the reactor tank

The following summarizes the conclusions obtained from the reactor design bases subsequently described:

a) Fuel Temperature

The fuel temperature is a limit to steady state power operation because of the need to avoid changes in the U-ZrH_x fuel which could lead to sufficient pressure to rupture the aluminum cladding. Since most of the fuel to be used in this TRIGA is of the low hydride type (ZrH_x where $x < 1.5$), the phase transition that occurs at approximately 530°C should avoided. Thus, a limit of 530°C has been set for the fuel in this reactor, which will contain mostly U-$ZrH_{1.0}$, the older type fuel. A few of the elements may be of the newer type, with $x > 1.5$ in ZrH_x. These have much higher operating capability, approximately 1150°C for the stainless steel clad type.

b) Clad Temperature

Both the aluminum and stainless steel clad must maintain their cladding. Hence, they must not melt or reach a temperature sufficient to reduce their yield strength to the level that will allow the internal pressure inside the fuel rods to cause swelling and rupture. A conservative temperature limit of 400°C has been set for the more restrictive aluminum clad, at which the creep rate from internal pressure is negligible.

c) Reactor Power

Fuel and clad temperature limit the operating power of the reactor. However, it is also convenient to place an operating limit on core power which is easily measured. The design bases analysis indicates that operation at 2000 kw with natural convective flow will just reach DNB conditions (Departure from Nucleate Boiling) and that below DNB, neither fuel nor clad temperatures will exceed their limits. The allowable K-excess limit is also an inherent limit on core power, preventing long-time steady state operating powers above 600 kw, and initial steady state power (before xenon buildup) above approximately 1000 kw. Thus, the requested licensed operating power limit of 100 kw with 2.25% cold K-excess is most conservative.

d) Retention of Water by Reactor Tank

Though not directly affected by operating power levels, the reactor tank must maintain its integrity so that the water level above the core will not drop significantly enough to create high radiation levels above the top of the tank during times that the reactor is shutdown. The tank has been designed with double containment, and emergency tap water lines are available for adding makeup water should leaking occur. Use of the double tank with sand between the liners makes the system more resistant to earthquake damage than would be a single rigid tank.

5.2.1 Reactor Fuel and Clad Temperature

The reactor fuel temperature limit is dictated by the characteristics of the ZrH_x ceramic fuel matrix. These characteristics are shown in Figure 5.2.1, where it can be seen that the "low hydride" (x in ZrH_x less than 1.5) undergoes a solid-solid phase transformation at approximately 530°C. The hydride content in most of the ZrH_x fuel elements to be used in this TRIGA is x=1.0. Therefore, 530°C has been established as the upper limit for the fuel matrix. At this temperature, the equilibrium hydrogen pressure is less than 0.1 psia, as shown in Figure 5.2.2, and distortion of the matrix by phase transformations will not occur. Thus the cladding integrity is assured since the stress at 0.1 psig on the 0.030 inch thick aluminum cladding is only 2.4 psi and 3.6 psi for the 0.020 in. stainless steel cladding; $S = \frac{Pr}{t}$ i.e, tangential stress in a thin-walled vessel is where $r = \frac{1.47}{2}$ inch radius, t = vessel (clad) thickness, and P = internal pressure

The other possible causes of high internal pressure are:

a) fission gas buildup

b) chemical reactions, such as metal-water reaction from water seeping through small breaks in the cladding and reaching the fuel matrix.

In regard to fission gas release, the U-ZrH fuel has been shown to retain a large fraction of even the gaseous fission products. However, assuming conservatively that all of the noble gases escape from the fuel matrix, 10% burnup of the fuel loading in an element will create approximately 0.003 moles of noble gas atoms. If all of these atoms leave the fuel matrix and accumulate in 10 cm^3 of effective plenum area at a temperature of 400°C, the total internal pressure in the fuel element due to these noble gases will be 15 atmospheres, or 220 psig. The use* of 10 cm^3 as the characteristic volume is realistic since a large fraction of the noble gases will not even escape from the matrix. Using the above equation, the stress in the cladding is 5200 psi under these conditions.

Typical stress vs. temperature data (for instance, see Nuclear Engineering Handbook, H. E. Thorington, Editor, 1958) for aluminum and aluminum alloys show that the creep rate under these conditions is quite

* As shown in Section 8, actually about 3.3 cm^3 of void is built into the element as the clad-fuel gap. Other voids and interstices in the fuel probably account for much more than an effective 10 cm^3 of void.

tolerable, less than $10^{-3\%}$ per hour. Thus, 400°C is an acceptable limit for clad temperature with a fuel element burnup of as high as 10%. At the 10% burnup level, the reactivity penalty will be intolerably high, and will generally preclude use of more than a very few fuel elements to this level of burnup. Fuel element logs will be kept in which approximate burnup will be periodically estimated.*

Among the chemical properties of U-ZrH, the reaction rate of the material with water that might leak in through the cladding is of particular interest. Since the hydriding of zirconium is an exothermic reaction, water will react more readily with zirconium than with zirconium hydride. Hence the water reaction is unlikily to occur with ZrH. The reaction is more likely to occur with the uranium. Experiments carried out of Gulf General Atomic Incorporated have studied the quenching with water of both powder and solid specimens of U-ZrH after heating to as high as 850°C. A relatively low chemical reactivity with both water and air was observed. Thus, a water leak in the cladding will not result in a rapid chemical reaction, and gases produced in the slow chemical reaction would probably

In summary, limits of 530°C for the fuel element matrix and 400°C for the aluminum fuel element cladding (conservative for the stainless clad fuel) will assure sufficiently low internal pressures and sufficient strength for the cladding that the fuel element integrity will be maintained and the cladding will not be breached. The design basis limit (safety limit for reactor operation) is that the temperature of an aluminum clad low hydride

* The estimated average burnup on the University of Arizona fuel is 0.6% maximum, or less than 1% on the 'hottest' (highest burnup) element. This is following 13 years of operation. Similar operating experience is anticipated at the University of Utah.

full element shall not exceed 530°C under any conditions of operation. Futhermore, if the cladding should break for other reasons, these same temperature limits will assure that whatever chemical reaction occurs with the fuel will be quite non-violent.

5.2.2 Reactor Power

The permissible fuel and clad temperature are the fundamental design bases limits and a temperature instrumented fuel element will be employed to monitor full temperature in the core, as specified in the technical specifications for the reactor.

The anticipated peak full and cladding temperatures have been obtained by others using a numerical methods, and the derivation and results are reproduced in Appendix II, as extracted directly from the Safety Analysis Report for the TRIGA reactor at the University of California's Santa Barbara Campus. It is sufficient to state here that the results of that calculation show that, for convective cooling, departure from nucleate boiling (DNB) does not occur until above a total reactor power level of 2000 kw. Furthermore, the maximum allowable K-excess of 2.25% will not allow long-time steady state operating powers above 600 kw. Prior to reaching xenon equilibrium, this K-excess would allow operation to close 1000 kw. Thus, the 100 kw maximum allowable operating power is most conservative. Furthermore, the K-excess limit is a backup inherent power limit, which will limit operating powers such that fuel surface temperatures would remain far below melting conditions. The following table shows the approximate temperatures for the hotspot region of the core at a total operating power equal to 100 kw licensed power.

Core inlet-water temperature	$70^{o}F$
Core average outlet temperature	$110^{o}F$
Core outlet-water temperature-hot channel	$122^{o}F$
Peak, cladding temperature ($70^{o}F$ coolant inlet)	$297^{o}F$
Peak drop across 1-mil clad-fuel air gap	$140^{o}F\Delta T$
Temperature difference between center and edge of fuel (peak)	$67^{o}F\Delta T$

It is apparent from the above table that the reactor at 100 kw, at which the initial operating license request is being made, is operating at conditions that leave the hotspot highly subcooled and the fuel element meat and cladding temperatures well below the maximum allowable from the design basis. (Appendix III contains details of the calculations of these temperatures.)

The 100 kw power level will also serve as a controlling parameter. The power will be continuously indicated to the reactor operator by the readings of the nuclear ionization chambers. Calibration of these chambers will be undertaken at low power with the use of foil or wire type flux detectors. Both bare and cadmium covered data will be taken, when feasible to determine the thermal flux activation component, from which the fission rate in the U-235 will be determined. Multigroup diffusion theory calculations could then be used to obtain details of the radial and longitudinal flux and power distributions and of the neutron energy spectrum. If possible, a U-235 fission rate using catcher foils will be measured at one location in the reactor. From these variety of data, it will be possible to calibrate the flux chambers to within an uncertainty of less than $\pm$ 10%.

5.2.3 Retention of Water by Reactor Tank

The reactor tank consists of an inner liner, 7 foot. 8 inch outer diameter, of 5/16 inch thick aluminum and an outer liner of 3/16 inch thick steel with a diameter of 12 ft. All soil surfaces of the latter tank are painted with epoxy to inhibit corrosion. The two foot space between the liners is filled with tamped sand to the concrete pad base. The bottom of the aluminum tank is a welded sheet of aluminum material the same as the side walls. (6061-T6 aluminum) Both tanks are water tight, and welds on the inner aluminum tank have been made to meet ASME unfired pressure vessel standards. (See Figure 4.4.1.1)

Under a hydrostatic head of 24 feet of water, the hoop stress at the bottom of the inner tank is 3500 psig, while the hoop stress on the outer steel tank, if subject to the same head, would be 10,000 psig, both well below the yield stresses for these materials. Since the bottom of the aluminum tank rests on a reinforced concrete pad which forms the bottom seal for the outer steel liner, the maximum stress that the bottom of the aluminum tank is subject to is the hoop stress at the bottom weld. Therefore, the only major consideration is the ability of the tanks to resist the unexpected, such as damage from earthquake, penetrating objects impinging on the inner tank, or from corrosion.

Corrosion problems appear to be of little concern. Normal ground temperatures are approximately 60°F, and the reactor water will be maintained within a few degrees of that temperature when the reactor is shutdown for long periods (i.e. over night). Through 13 years of operation at Arizona with tank water maintained in the 60°F range, no buildup of corrosion has been observed on any exposed aluminum parts beyond that initially formed as a protective layer. The outer wall of the aluminum tank and the inner wall of the steel tank will not be exposed to any moisture. The outer wall of the outer tank should be adequately protected by the epoxy paint, asphalt seal and polyethelene sheeting, particularly since the ground water level is well below the bottom of this tank.

Rupture of the inner tank will be detectable by a drop in water level. Even a large breech in the tank wall, however, should not result in a fast loss of water, which must percolate through the sand and concrete pad below the aluminum tank. If the integrity of the outer tank has not been breeched, the inner tank water level cannot drop more than 6 feet, based on a sand porosity (packing density) of 60%.

The possibility of an earthquake of sufficient magnitude to damage and breech both tanks, though quite remote, still exists. It is believed that under expected types of ground motion, the sand packing between the annular portion of the two tanks adds ductility to the system such than it is better able to resist ground shear forces that might otherwise fracture a rigid fully concrete-filled structure.

None of the above considerations relate to operating power levels of the reactor. In fact, operating power has no effect on the tank materials, except possibly the concrete pad. Aluminum is one of the most resistant materials known to fast neutron damage. Concrete will suffer some embrittlement, but is unlikely to be significantly affected over the projected 30 year life of the system, for which peak integrated fast doses will be less than 10^{19} nvt. The steel outer liner will be in such low flux that radiation damage will be negligible.

In summary, the double containment system is designed to safely retain the water in the reactor tank, except perhaps under the most catastrophic earthquake. Should the system loose some water, an emergency makeup hose is readily available, connected to a tap water line. Furthermore, a low water level indicator on the tank immediately sounds alarms both in the reactor lab and at the Campus Security Office. Should uncontrollable

5.3 Mechanical Design and Evaluation

5.3.1 General Description

The reactor is located below ground level inside a shielded reactor tank. The general design of the tank and shield are shown in Figure 5.3.1. This design provides for shielding of the neutron flux from the earth by at least 2 feet of water* in the tank and 2 feet of sand surrounding it. Approximately 20 feet of water and/or concrete and/or sand and earth above and to the side of the core provides the necessary personal shielding.

The reactor system will contain provision for 3 diagonally directed beam tubes between the reactor core and the reactor room floor. Each tube will be composed of two sections aligned along a common axis. The top tube section will be a 1 foot diameter tube between the reactor floor and the wall of aluminum reactor tank. This tube will not penetrate the aluminum tank but will be sealed at the end where it butts against the tank. Except when in use to extract a neutron beam for experimental purposes the upper beam tube will be adequately shielded by being filled with sand and capped at the reactor floor level with a 1/16 inch thick locked steel cap for security. When the beam tube is to be used, the sand filler will be removed and a sealed aluminum beam tube (carefully weighted so as to have a net density greater than water) will be installed between the inner tank wall and the reactor core shroud along the common axis of the upper tube.

* Operating procedures will require that at least 2 feet of water reflector-shield be between the core edge and the inner tank of aluminum.

U/U TRIGA

NUCLEAR REACTOR TANK CROSS SECTION

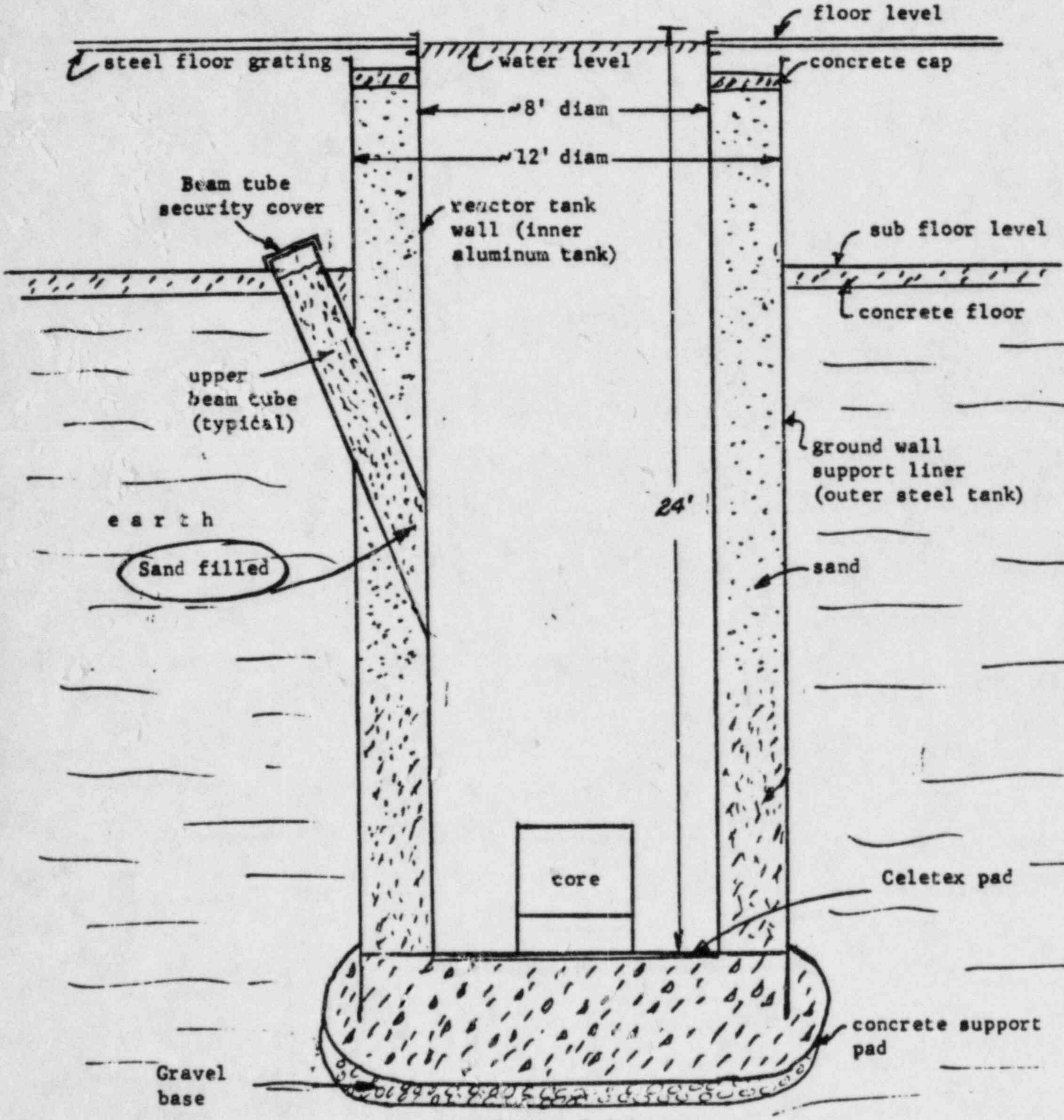


Figure 5.3.1

Special attaching fixtures will be welded to the inner tank wall to permit correct alignment. During beam tube use since both the lower and upper beam tube sections are sealed against air flow so that no radiation problems arising from argon-41 or nitrogen-16 buildup will occur. Furthermore, no penetration of the reactor tank is made for the beam tubes. Both inner tubes only butt against the reactor tank so no increased risk of inner tank water leakage is incurred. When the beam tube is being used appropriate shield at the reactor floor level will be made with dense concrete block and other shield materials available.

The core components are mounted between two grid plates, which are part of a fixed structure setting on a six-legged pedestal support. This entire pedestal structure can be moved as a unit by lifting it with a crane, thus permitting occasional changing of locations of the core within the tank in order to accomodate large experiments between the core and the tank wall (such as a thermal column or beam tube, if added in the future). Note, procedures will not allow shifting of the core without first partially unloading the core to insure subcriticality.

Various irradiation facilities may be provided within the core of the TRIGA Mark I reactor. This particular grid plate structure will allow for a central experimental flux trap the size of seven fuel elements and for possibly two off-center triangular shaped, three-fuel-element sized irradiation facilities. These special facilities may be replaced with fuel elements, within the technical operating limit constraints, by inserting special inserts into the large holes in the upper grid plate. In addition to each of these special irradiation facilities, and irradiation space can be provided in any one of the fuel element spaces. One such special irradiation device is a high speed pneumatic "rabbit" transfer system, operating with nitrogen gas as its propellant.

5.3.2 Grid Plates and Core Structure

The upper and lower grid plates were constructed from drawings and specifications supplied by Gulf Environmental Systems, Inc., except that an effective lattice pitch of 1.49 inches between centers has been adopted, identical to that of the TRIGA at the University of California at Santa Barbara.

Upper Grid Plate - This plate is 3/4 inch thick, type 6061-T6 aluminum. It laterally positions the fuel elements, control rods, irradiation facilities, and neutron source. The plate carries no vertical load other than its own weight. It is mounted to six side plates, each 3/16 inch thick 6061-aluminum which are secured to the bottom grid plate.

The upper grid plate has 127 locations for fuel elements (or moderator rods, control rods, etc.). These 127 holes are in six hexagonal rings around the center hole and are 1.505 (+0.005, -0.000) inch in diameter. Cooling water passes through the top plate by means of the clearance provided by the differential area between the triangular spacer blocks on the top of each fuel element and the round holes in the grid plate. There are also several small diameter flux-wire insertion holes in the top grid plate in the interstices between the fuel element holes. A 7 central fuel element cutout is provided for A & B ring positions which can accommodate a 4.215 inch diameter experiment with a cross section area of 13.96 square inches.

Two triangular holes encompassing three fuel elements and with their apexes rounded to the shape of fuel element holes maybe provided in the upper grid plate at D and E ring locations. These cutouts may be removed to accomodate special experiments, or may be inserted into the upper grid, supported on an undercut step.

Lower Grid Plate - The lower grid plate provides lateral positioning and supports the entire weight of the core. It is also type 6061-T6 aluminum, and is 3/4 inches thick. It rests on six legs (pads), six inches from the base of the aluminum reactor tank. Attached to the lower grid plate is the 3/16 inch thick hexagonal aluminum shroud plates for the core.

The lower grid plate also contains 127 holes for fuel elements, but each hole is 0.25 inches in diameter with a 5/8 inch diameter counter sink. The tapered fuel element ends fit snugly into these holes. Cooling water cannot pass through the fuel element holes in the lower grid plates. Cooling water does, however, flow through 5/8 inch diameter holes in the interstices between each fuel element position in the lower grid plates. The fuel insertion holes in the lower grid plate of course line up with the similar holes in the top grid plate. Since the control rods contain no followers, any fuel element position may also accomodate a control rod.

5.3.3 Moderated Fuel Elements

The aluminum and stainless steel clad fuel elements and the optional graphite or heavy water loaded reflector elements all have similar outside dimensions. These are nominally 1.47 inches O. D., except for the upper and lower end fixtures, and 28.44 inches long tip to tip. Figure 5.3.3 shows the typical fuel element. The fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing about 8 1/2% by weight of uranium enriched to 20% in U-235. The hydrogen to zirconium ratio is approximately 1.0. Each aluminum clad element is clad with 0.030 inch thick aluminum and the stainless steel clad elements with a clad 0.020 inches thick. The active fuel portion is 1.40 inch nominal diameter by 14 inches long. On each end are 4 inch long sections of graphite. Aluminum end fixtures are on both ends of the fuel element, as shown in the Figure. The upper fixture contains a knob for grasping by the fuel handling tool. Table 5.3.3.1 is a summary of the fuel-element characteristics.

TABLE 5.3.3.1

Summary of Fuel Element Specifications

Item	Nominal (Value)	
	Aluminum	Stainless Steel
Fuel-Moderator Material		
U-ZrH_n		
H/Zr atom ratio	1.0	1.0
Uranium content	8.5 wt%	8.5 wt%
Enrichment (U-235)	20%	20%
U-235 mass (per element)	37 gm	37 gm
Diameter of fuel meat	1.41 inches	1.41 inches
Length of fuel meat	14 inches	15 inches
Graphite End Reflectors		
Porosity	20%	20%
Diameter	1.4 inches	1.4 inches
Length (top and bottom) each	4.0 inches	4.0 inches
Cladding		
Material	Aluminum	Stainless steel 304
Wall thickness	0.030 inches	0.020 inches
Length	23 inches	23 inches
End Fixtures	Aluminum	Stainless steel 304
Overall Element		
Outside Diameter	1.475 inches	1.475 inches
Length	28.44 inches	28.37 inches
Weight	7 pounds	8 pounds

5.3.3.2 Evaluation of Fuel Element Design

Gulf Energy and Environmental Systems, Inc. has acquired extensive experience in the fabrication and operation of U-ZrH fuel elements. With these elements operating well below the design bases limits, no stress-associated failure of the elements is expected. The moderator elements if employed will be manufactured at the University of Utah shops. Since these elements experience no nuclearly induced stresses, failure of the elements would not be expected. If water should leak into the graphite or heavy water, the effect would be a loss of reactivity.

5.3.4 Core

The active core will have the approximate shape of a right hexagon. The fuel element spacing is such that 33% of the cross sectional core area in the lattice is water. The extra fuel element spaces on the outside of the active core allow for the insertion of reflector elements. The minimum critical loading using the elements that have been in the University of Arizona's reactor is anticipated to be 62 elements, with control properly placed within the critical array and with the core surrounded by reflector elements. The operational loading with a cold clean k-excess of 2.25Δk is expected to be approximately 75 elements, if these are reflected with one row of reflector elements, or about 80 elements with no reflector elements.

5.3.5 Reflector

The reflector will generally consist of a single row of graphite or heavy water reflector elements. These are such that the equivalent volume fraction of the first 1.6 inches of reflector is approximately 33% water, 62% graphite or heavy water and 5% aluminum. The remaining part of the reflector is water, plus the 1/4 inch thick aluminum core shroud. The effectiveness of this combination reflector is expected to be approximately 1.5%ΔK less than that of the mostly graphite reflector used on the University of Arizona reactor. This means that the required number of fuel elements to achieve a certain operational loading will be between 5 and 15 more than the number used in the Arizona configuration.

5.3.5 Irradiation Facilities

The irradiation facilities that will be installed are constrained by the size of the various available locations. The center seven elements will accomodate a facility up to 4.2 inches in diameter. However, since the central facility is intended to utilize a flux trapping technique, much of this diameter will contain water or other moderator. There may also be provided two triangular openings in the top grid plate to accomodate a round facility as large as 2.4 inches in diameter. All other facilities will need to fit through the 1.5 inch diameter fuel element positioning holes in the upper grid plate, unless they are placed outside of the core shroud.

A pneumatic rabbit may be operated within one such 1.5 inch O. D. tube. It will be driven by dry nitrogen gas* so as to minimize Ar-41 production. As noted in Section 5.1, flooding of the central irradiation space from a sealed gas filled condition is the worst flooding accident. Its worth is estimated to be 0.6%Δk, less than one dollar. Thus the worst flooding accident cannot make the reactor prompt critical if initiated from a steady power condition..

*Or equivalent gas with a low radioactivation potential.

The pneumatic rabbit will have a slight curve in its tube such that direct uncollided streaming of neutrons from the core to the surface of the pool cannot occur. Most dry irradiation tubes inserted into the reactor will not be dry to the top of the tank unless specifically designed for use as beam experiments. Under such conditions, special precautions will be taken to adequately shield the floor above the reactor from irradiation by these beams.

5.3.7 Control

The reactor contains three control rods. These rods operate in aluminum guidetubes in fuel-element positions in the core. The tubes are held in place by the top and bottom grid plates. The 3 rods are located(approx) symmetrically near the center of the core and include a safety rod, with a worth of approximately 2.0%; a shim-safety rod, with a worth of about 2.0%; and a regulating rod, with a small worth of about 0.5% Δk. Figure 5.3.7.1 shows a control rod, magnet, and guide tube.

All rod drives will be essentially identical. These drives, of a winchtype design shown in Figure 5.3.7.2 were designed at Gulf Energy and Environmental Systems, Inc. and have been in use on the Arizona reactor for 13 years. The components for the control-rod drive mechanisms are arranged in line on a base plate. The components consist of a motor, a brake, a cable drum, a 10-turn Helipot, and a limit-switch mechanism, with suitable hangers and flexible couplings for mounting them. The drives are mounted below floor level on a concrete shelf alongside the reactor pit, underneath the reactor pit cover. A housing will be provided for each drive.

The motor is a nonsynchronous, single-phase, instantly reversible type. Contained in the motor housing is a four-stage spur-gear reducer unit having a total speed reduction of 900 to 1. This motor-gear reducer unit has an output speed of 1.9 rpm. When the motor is not in motion, the shaft is locked by means of an electrical brake. In the case of the regulating rod, a special servomotor replaces the conventional motor.

A stainless-steel aircraft cable extends horizontally from the drum to an enclosed cylindrical sheave mounted directly over the control-rod guide tube. Attached to the lower end of the cable is a holding magnet. This magnet travels up and down in the guide tube and engages the armature on the absorber rod. In the event of a power failure or a scram signal, all magnets de-energize, thus allowing the absorber rods to fall into the core.

The pitch diameter of the cable drum is such as to give a linear rod speed approximately equivalent to 3 cents/second for the rod near the center of its travel. Overtravel is prevented by means of both electrical and mechanical stops on the limit-switch mechanism. The limit positions are adjustable and can be set for any desired travel.

Instrumentation and Control System

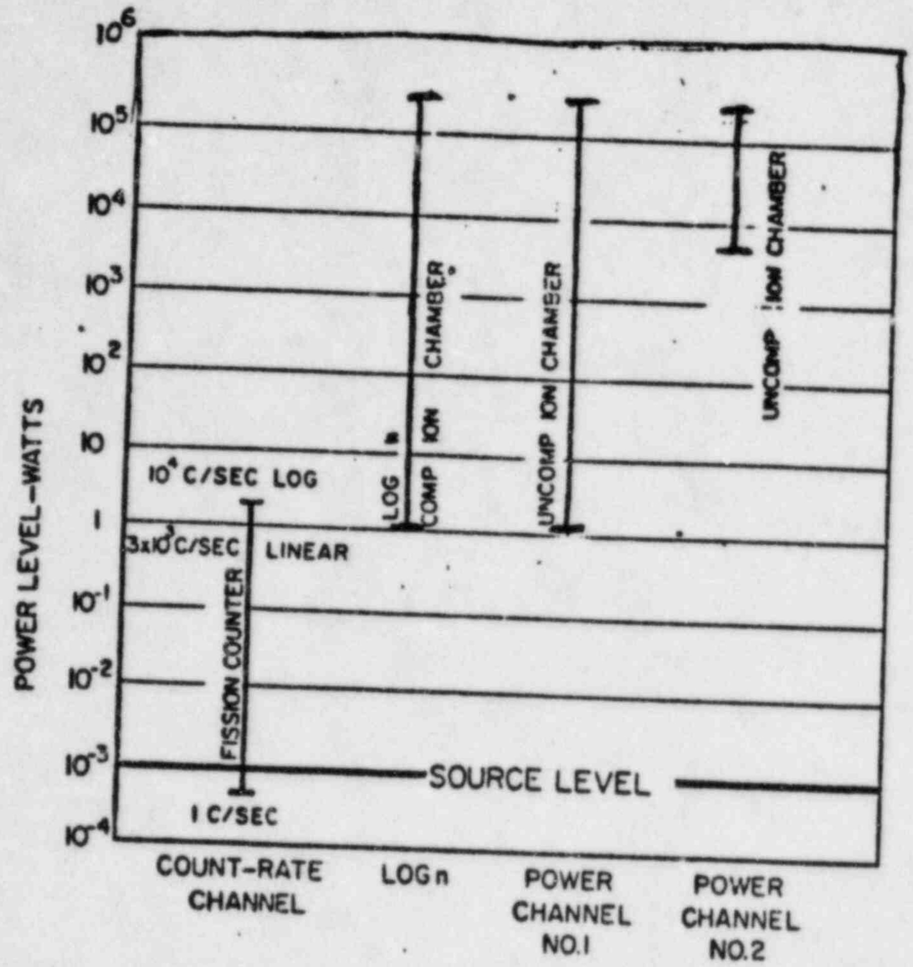
The instrumentation and controls for normal reactor operation are contained principally in the main control console, shown in Figures 5.3.7.3 and 5.3.7.4. For the initial approach to criticality, additional instrumentation is provided in an auxiliary rack. Figure 5.3.7.5 shows a block diagram of the principal features of the instrumentation and scram system. Detailed specifications of the various neutron-measuring channels described below are given in Table 5.3.7.

A ~2 curie Pu-Be start-up source, the initial source strength of which is ~10^7 neutrons/sec, is located in a special reflector element source holder suspended in one of the top grid plate fuel holes (see Fig. 5.3.7.6). Once the reactor is critical the source can be moved from its in core position to storage position by means of a cord attached to the top of the secured source holder cap. An indicator light (and source interlock) (See Figure 5.3.7.4) visible to the operator will indicate if the source is in or out during startup.

The ion chambers and counters are located in the water approximately on the core axial midplane on the outer perimeter of the core-shroud assembly.

A fission counter is used with a transistorized linear amplifier and preamplifier to feed both linear and log count-rate circuits. The log count rate is read on a meter. Both linear and log count rates may be read on the linear and log recorders when the reactor is in the countrate range of from about 10^{-3} w to about 2 w (source level). These source levels are estimated to give count rates of about 5 counts/second and 10,000 counts/second, respectively. An interlock circuit is used to prevent rod withdrawal unless the source count level is above the required minimum value of at least 2 counts/second. The amplified pulses from the fission counter scaler are also audible from a small speaker mounted within the console.

The log n channel uses a compensated ion chamber and, with the transistorized log amplifier, will give usable readings from about 1.0 w to 300 kw. The log n output is read on a meter and on the log recorder. The output of the log n amplifier circuit also feeds a transistorized period circuit. This furnishes a period indication on a meter from -40 to ∞ to +7 seconds.



LINEAR COUNT-RATE — FULL-SCALE (RECORDER SCRAM) 30, 100, 300, 1000, 3000 C/SEC
DISPLAY RECORDER ONLY

LOG COUNT-RATE — FULL-SCALE (RECORDER SCRAM) $1-10^4$ C/SEC
DISPLAY METER AND RECORDER

LOG n — FULL-SCALE (RECORDER SCRAM) 3.0 to 300 kw
DISPLAY METER AND RECORDER

POWER CHANNEL NO.1 — FULL-SCALE (RECORDER AND ELECTRONIC SCRAM) 3, 10, 30, 100, 300, 1000, 3000, 10 000, 30000, 100000, 300000 w DISPLAY METER AND RECORDER

POWER CHANNEL NO.2 — FULL-SCALE (RECORDER AND ELECTRONIC SCRAM) 0–150 % FULL POWER
DISPLAY METER ONLY

|--Estimated range of reactor instrument channels

Figure 5.3.7.7

TABLE 5.3.7

SPECIFICATIONS FOR NEUTRON-MEASURING CHANNELS

1. Count-rate Channel (one required):

   a. Counter--Westinghouse WL-6971 or equivalent fission counter having a sensitivity of 0.14 count/neutron/cm^2.

   b. Preamplifier--transistorized model.

   c. Linear amplifier--General Atomic transistorized model.

   d. Log count-rate meter--General Atomic transistorized model.

   e. Linear count-rate meter--General Atomic Model CR-100.

2. Log n and Period Channel (one required):

   a. Chamber--Westinghouse WL-6377 or equivalent compensated ion chamber having a sensitivity of 4×10^{-14} amp/nv, electrical compensation, and compensated γ sensitivity of about 3×10^{-13} amp/r/hr.

   b. Log n amplifier--Burr-Brown Model 3061/25 integrated circuit operational amplifier following a multiple-silicon-diode log input circuit. Range: 5 decades from 4×10^{-4} to 4×10^{-9} amp.

   c. Period amplifier-General Atomic Model AP-130 transistorized period amplifier having a range of -40 to ∞ to +7.

   d. Log n recorder--Varian G-11 (also used with log count-rate circuit).

3. Power-level Channels (two reguired):

   a. Ion chambers--Westinghouse WL-6937 or equivalent uncompensated ionization chamber using enriched U_3O_8 and having a sensitivity of 4.4×10^{-14} amp/nv and a γ sensitivity of 4.5×10^{-11} amp/r/hr. (Full-power γ flux at chamber is estimated to be 2×10^5 r/hr.)

   b. Power-level amplifiers--(1) Burr-Brown Model 3061/25 integrated circuit operational amplifier, accuracy $\pm 0.02\%$, drift less than 0.03% of full scale for 8 hrs within temperature limits of 20° to $50^\circ C$. (2) Chamber feeds meter and scram circuit directly through an attenuator. Range 0%-150% full power.

The layout of these controls and the various monitoring lights is shown in Figure 5.3.7.4. The reactor cooling-water temperature is measured at the tank. The fission-product activity of the water is monitored continuously, and a red pilot lamp lights, and an alarm bell sounds if the activity rises above a preset level.

A brief check-out procedure for the reactor console prior to start-up is as follows: The power switch is used to energize all instrument and control circuits except the magnet-holding circuit. Filament circuits of the few tubes involved in the console are normally left on at all times (they may be turned off when necessary at the rear of the console). The key-operated switch will turn on the control rod magnet power supply. This arrangement allows technicians to check out most of the console without inadvertently pulling a control rod in the absence of a licensed operator. Only licensed operators have access to the console key. A minimum source count from the count-rate circuit is also used to interlock the rod-drive circuit to prevent raising of the rods unless the source count is adequate. The power-level amplifiers are first adjusted for zero and then a test signal is used which not only tests the amplifier but also calibrates it. A test signal may also be used to raise the signal level to check the scram circuit. Artificial signal sources for low and high calibration points are also available for checking the log n. A signal is available for testing the period circuit and for setting the period scram. The operation of the count-rate channel can be checked by the source count and the channel can be calibrated at both high and low counting rates by means of test signals.

5.3.8 Cooling System

Cooling of the reactor is accomplished by natural convection of the pool water. A five ton rated capacity evaporative cooling tower may also be utilized as the heat sink. Heat will be rejected from the reactor tank to the cooling tower through a heat exchanger. Initially the reactor will be operated without the cooling tower heat sink by permitting heating of the reactor tank water up to a temperature of 95°F.

The heat exchanger will receive warm water from the pool and returns cooler water to the pool. The return line carries the water to a location several feet above the core where it is ejected into the pool water as a horizontal jet in order to diffuse the nitrogen-16 activity and prevent it from reaching the top of the pool as fast as it would otherwise. The discharge temperature will be about 50°F. Both the inlet and return lines to the pool have small ~1/4 inch holes approximately one foot below the normal pool water level. These holes will stop siphoning action and prevent draining of the pool below that level should a siphoning drain be accidentally initiated. The cooling tower water to the heat exchanger will be circulated at the rate of approximately 20 gallons per minute by a centrifugal pump. Since the cooling tower is subject to freezing, it will be provided with weather covers, and the water will either be drained from it or be continued to circulate when temperatures are below freezing.

Normally the reactor pool temperature will be maintained at about 60°F, which is approximately the ground temperature. For any operation above 25 kw, the heat exchanger-cooler has insufficient capacity to maintain the 60°F. The pool water heat capacity is approximately

22 kw-hrs/^{o}F. Thus, at 100 kw and with the heat exchanger operating the pool temperature will rise approximately 5°F per hour. The water temperature will not be allowed to exceed 95^{o}F. However, after the reactor is shutdown, the cooler can lower the pool temperature at approximately 1^{o}F per hour.

5.3.9 Water Purification System

A small ion-exchange water purification system will be installed inside the reactor building. Water from the pool will be circulated through this bed at a slow rate (1 to 10 gallons per minute) by a centrifugal pump. The water will be both taken from and discharged to the top of the pool. The ion exchange system is of standard design, using readily available mixed resins from commercial manufacturers.

The water in the pool will be regularly sampled for ph and conductivity measurements. In order to minimize aluminum corrosion, it is desirable to maintain the water ph in the range of 5.5 to 6. Significantly lower ph than this value (more basic) will be buffered out by the addition of some nitric acid. Also, the conductivity will be maintained within acceptable limits. Failure to be able to do so will then indicate filter bed replacement is necessary.

5.3.10 Radioactive Monitoring and Disposal

Two types of automatic indicating alarms for high radiation will be continuously operating. These alarms will automatically indicate both in the reactor room and at the Campus Security headquarters, and thus will receive around-the-clock surveillance. The alarms when triggered will provide a sight and sound alarm in the reactor area and operate a red light alarm at the Campus Security Headquarters.

One radiation alarm will actually be a low water level indicator, set to alarm if the pool water level drops 12 inches below the normal level. This alarm will be actuated by a standard microswitch levered to a plastic float. A 12 inch drop in water level activates the switch.

The other alarm will be from a direct reading geiger tube mounted directly above the pool of the reactor. This will provide a high level radiation signal to a relay and will be set for levels higher than expected in all normal operation.

Few radioactive wastes will be generated, except those consisting of fission products in the fuel elements. The other main sources of wastes are from induced activity in irradiation samples and from ion collection in the filter beds. The latter will routinely be monitored for radioactive buildup. Disposal of any of the wastes will follow standard university policy, as specified in the Radiological Safety Manual for the University of Utah.

6.0 Initial Reactor Startup and Calibration Tests

Personnel

All reactor operations during reactor startup and calibration tests will be under the direct supervision of the Reactor Administrator. A Senior Reactor Operator has been trained on this TRIGA facility at the University of Arizona prior to the time that the reactor was shipped to Utah.

Chapter 9 gives a detailed account of the proposed operations and administrative structure for the Reactor Facility. Furthermore the University of Utah Radiation Safety Manual (enclosed with application) gives detailed emergency procedures for the reactor facility.

The following is an outline of the tests to be performed during the initial startup and calibration of the TRIGA reactor:

6.1 Pre-critical Test

6.1.1 Functional Tests of Mechanical Equipment

These tests include verification of proper installation, and performing operational tests where applicable on the isotope production facility, pneumatic system, water system, fuel element handling tool, and other related equipment.

6.1.2 Checkout of Instrumentation and Control System

A complete checkout of the control console is performed to insure that all circuits are calibrated and operating properly. All interlocks and scram initiations are shown to operate satisfactorily. In addition, the rod drives are checked for proper operation during rod withdrawal and scram situations. The fission and ion chambers are shown to be sensitive and responsive to neutrons.

A Keithley micromicroammeter and a scaler may be used as a temporary addition to the console instrumentation to provide more information during initial startup.

concentration assuming air rather than nitrogen filled the specimen rack produces a discharge concentration about 1% of the RCG for Ar-41.

To minimize Ar-41 activity the U/U pneumatic tube system will be operated with bottled nitrogen gas* (less than 10 ppm argon) and will discharge into the reactor room exhaust system. Even under the most severe operational conditions and assuming that air at atmospheric pressure filled the in-core terminal during the irradiation and is completely removed and vented with each sample change, the concentration in the discharge would reach about 2×10^{-7} $\mu c/cm^3$. Even under adverse dispersion conditions, it is very unlikely that this discharge will not be diluted by a factor of 10 (to the RCG for uncontrolled areas) before the exhaust cloud reaches a populated area. Estimates of probable dispersion are given in Section 8.4.4

Calculation

The concentration of Ar-41 produced in air irradiation of 100 kw for time t is

$$X = \frac{N(40)\phi\ \sigma\ (40)}{3.7 \times 10^4}\left[1 - e^{-\lambda(41)t}\right] \mu c/cm^3$$

where ϕ = av. neutron flux = $\begin{cases} 1.76 \times 10^{12} \text{ at } 100 \text{ kw} \\ 4.4 \times 10^{12} \text{ at } 250 \text{ kw} \end{cases}$

$N(40)$ = Number of Ar-40 atoms/cm^3 = 2.5×10^{17}

$\sigma(40)$ = Cross section for Ar-41 production

= $0.53 \times 10^{-24} cm^2$

$\lambda(41)$ = Decay constant for Ar-41 = 1.06×10^{-4} sec^{-1}

* Or equivalent gas with a low radioactivation potential

9.0 Operations and Administration

9.1 General Considerations

The reactor will be operated by the Mechanical Engineering Department (Nuclear Engineering Program) of the College of Engineering of the University of Utah, Salt Lake City, Utah. The proposed administrative organization is shown in Figure 9.1. This organization is designed to promote efficient utilization of the reactor facility and yet to establish adequate safeguards for the health and safety of the community. Each unit in this network is discussed further below. Also discussed are provisional criteria for the examination of proposed experiments and for ensuring safe operation of equipment. The suggestions are not meant to be limiting and will be subject to reconsideration by the appropriate administration in the light of operational experience. Biographical resume's of the Reactor Administrator, Reactor Supervisor, Reactor Safety /Committee, and Reactor Operators are attached as Appendix IV.

9.2 Administration

9.2.1 Reactor Administrator

The Reactor Administrator will be in complete charge of the facility. He and other members of the Reactor Safety Committee are appointed by the President of the University. The Administrator is Chairman of the Reactor Safety Committee. The Administrator is responsible for liason with the NRC regarding all matters concerning the facility and for enforcement of all regulations. In addition he is responsible for recommending the appointment of competent personnel as Reactor Supervisor, Operators and other staff of the facility.

9.2.2 Reactor Supervisor

The Reactor Supervisor will be the operational administrator of the facility--an academic staff member of the Mechanical Engineering Department, holding a Senior Operator's License.* He will be directly responsible to the Reactor Administrator for the operation of the facility; and thus will enforce administrative rules and operating procedures, be responsible for all planning and scheduling, and will review and classify all proposed experiments and modifications to the reactor and submit them to the Reactor Safety Committee for approval. He will consult directly with members of both the Reactor Safety Committee (RSC) and the Radiological Health Committee (RHC) concerning all safety aspects of reactor operation. Within the reactor facility he will enforce compliance with the requirements of the RHC and be responsible for monitoring operations. Other reactor operators and trainees will be under his control.

The Reactor Supervisor will be responsible for the operation, maintenance and scheduling of not only the TRIGA reactor but also of the AGN reactor, neutron-generator, californium irradiator, and all other irradiation facilities in the Nuclear Laboratory.

9.2.3 Reactor Safety Committee (RSC)

This Committee, which is chaired by the Reactor Administrator, must approve new and unreviewed reactor operations and will periodically review all operations of the entire nuclear facility. Members are persons having experience and interest in the use of reactor facilities. The Reactor Supervisor and the Radiation Safety Officer are members.

The Reactor Supervisor will report to this Committee on any problems concerned with the reactor. In particular, the approval of this Committee will be required for start-up of the reactor following major modification or repair. He must be able to satisfy the Committee at all

* Initially for the first critical startup of the reactor a single senior reactor operator will be licensed. The Reactor Supervisor will obtain a senior operator's license within one year of the initial startup.

times that the reactor and facility are operating properly. The Committee may forbid any operation it does not believe is safe.

This Committee must approve all documentation relating to reactor operation. All experiments, which may significantly alter safety considerations previously reviewed by the Committee, must be formally submitted to the Committee as test programs for the Committee's review and approval before such experiments may be conducted. Experiments which do not alter safety considerations but nevertheless are deviations from previously approved experiments must be documented in the reactor files and approved by signature of the reactor supervisor.

9.2.4 Radiological Health Committee (RHC)

This Committee, appointed by the President of the University advises the President concerning radiological safety on Campus. This Committee controls the movement and use of all radioisotopes and radiation producing machines on Campus. All isotopes produced in the reactor for use outside the Department will be controlled and regulated by this committee. The regulations of this committee will be enforced within the reactor area by the Reactor Supervisor who is a member of the RHC. The Committee and the Radiation Safety Officer will be consulted and kept fully informed concerning radiological problems in reactor operation.

At present, this Committee comprises a Professor of Radiology (Chairman), a Professor of Internal Medicine, a Professor of Radiobiology, a Professor of Physics, a Professor of Medical Technology Science, a Professor of Mechanical Engineering (Reactor Supervisor), a Professor of Biology (Radiation Safety Officer), and a Professor of Chemistry.

9.2.5 The Radiation Safety Officer

An experienced Health Physicist serves on the Staff and functions as the Radiation Safety Officer for the University. He is responsible for the radiological health and safety of the University. Working through the Reactor Supervisor he ensures the safety of operations in the reactor area and is responsible for the transfer of isotopes outside the Department. He is available for consultation with the Reactor Supervisor, and is available to any university official in the event of any emergency relating to radiological safety. He is a member of both the RSC and the RHC.

9.2.6 Reactor Operators

All reactor operators will be licensed by the U. S. Nuclear Regulatory Commission to operate this reactor. Initially a Senior Reactor Operator has been trained to operate the TRIGA reactor at the University of Arizona where the reactor is presently located. Subsequent operators will be trained under his supervision. Graduate students may take examinations for an operating license and be employed as reactor operators. Initially, the reactor supervisor will be trained on the console by the previously trained senior operator. Due to prior operating experience, the senior operator can be licensed before the initial loading of the reactor. The Reactor Supervisor will be licensed (senior operator's license) within a year after the critical startup of the reactor.

9.3 Administrative Controls and Procedures

9.3.1 Introduction

This section describes the proposed administrative procedures pertinent to the operation of the facility and the performance of experiments. The Reactor Supervisor will be responsible for ensuring compliance with the established controls, with the Reactor Administrator bearing ultimate responsibility for assuring enforcement of these regulations.

9.3.2 Access to the Reactor Area

Normal access to the reactor will be through the control room. Persons entering the reactor room will be required to wear personal dosimeters.* The overhead door to the loading area will normally be closed and locked. Emergency exit may be made through the radiochemistry laboratory or through the control room. When the reactor is not operating, unsupervised access will be permitted only to persons holding authorization from the Reactor Supervisor. Such authorization will require the holder to have a knowledge of safety and emergency procedures within the reactor area. Emergency instructions will generally require that a senior member of the reactor staff or the Radiation Safety Officer shall be contacted prior to entry by Police or Firemen.

9.3.3 Reactor Operation

For each reactor operation, an Operator-in-Charge shall be designated. He shall be responsible for ensuring that:

(a) he is in a position to operate the control of the reactor at all times the reactor is critical.

(b) the correct start-up check and shut-down check procedures are followed and that log sheets are filled out for all operations.

(c) any experiment is correctly authorized and that any requirements noted by the RSC have been complied with.

(d) the experimenter's proposed procedure conforms to RHC recommended practice and that the experimenter has RHC approval if the handling of radioisotopes is involved.

* Exceptions apply to tours of visitors at times when the reactor is shut down. The authorized tour leader need be the only one wearing a personal dosimeter, but the names of all visitors and the tour leader will be recorded on the laboratory log.

(e) all samples removed from the reactor are monitored, their activity levels recorded, and any necessary temporary access barriers or shielding are used.

(f) the experimenter and the Reactor Supervisor are informed in case of any unusual or unexpected incident, apparent equipment or instrument failure or malfunction.

(g) the Radiation Safety Officer has been notified if any experiment predicted to involve high radiation levels is to be performed.

The Operator will normally satisfy (c), (d), and (g) by ensuring that the Reactor Supervisor has correctly approved the proposed experiment and schedule. All members of the Reactor Staff will expected to be cognizant with basic radiation safety procedures, so that adequate safety is ensured even in the absence of the Radiation Safety Officer from the university campus.

9.3.4 Routine Monitoring and Test Procedures

9.3.4.1 Components

The Reactor Supervisor will set up a program for regular testing of all safety equipment, procedures and certain reactor components.

In particular, radiation monitors will be checked frequently with calibrated sources, and the alarm set points on the continuous air and area monitors shall be checked. The operation of the emergency purge exhaust system and ventilation shut-down shall be tested semi-annually.

Similar checks shall be made of reactor controls such as interlocks, control rod drop times, power level safety circuits and the water level alarm.

Any malfunction of the above, or related systems shall be sufficient cause for suspension of reactor operation until the fault is corrected.

The results of all tests will be recorded in a log book kept in the control room.

9.3.4.2 Radiation Levels

In consultation with the Radiation Safety Officer, the Reactor Supervisor will draw up a scheme for routine monitoring with portable equipment to check radiation levels within and beyond the reactor area especially during reactor operation. Additional measurements will be made during installation start-up procedures so that radiation leakage levels estimated in this report can be ascertained.

The detection of any appreciable radiation above background at any point outside the general reactor area shall be cause for immediate investigation followed by corrective action in the form of procedural change or addition of shielding. The results of such monitoring will be recorded and maintained in the reactor control room.

Personnel monitoring will be under the direction of the Radiological Health Office which will provide a radiation monitoring service. These will be supplemented in the reactor area by pocket dosimeters which will also be used for occasional visitors. The Operator-in-Charge will ensure that persons entering the reactor room from the control room have a dosimeter.

9.3.5 Authorization of Experiments

All experiments proposed for the reactor will be designated Class I (Routine or Modified routine) or Class II (Special or New) by the Reactor Supervisor.

Class II comprises all experiments which must be submitted for review and approval by the Reactor Safety Committee. In addition, procedures involving radioisotope production for delivery outside the Nuclear

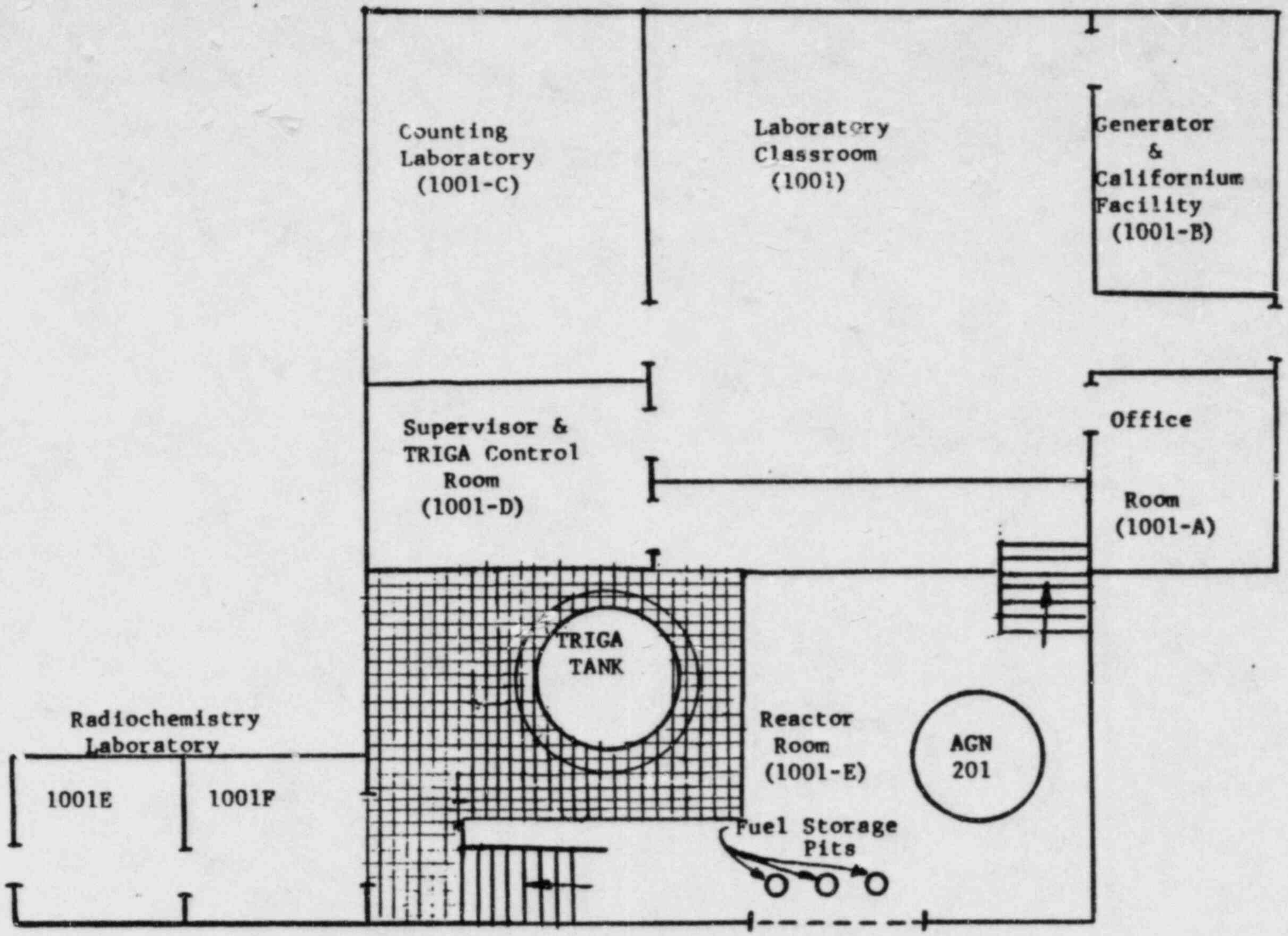
Engineering Laboratory and/or high radiation levels must be approved by the RHC. Experiment requests will normally be submitted to the RSC and RHC (if applicable) by the REactor Supervisor after evaluation by him.

Class I experiments include all repeats of successful experiments and those that are but minor modifications to a previous experiment. The Reactor Supervisor has the authority to approve these experiments.

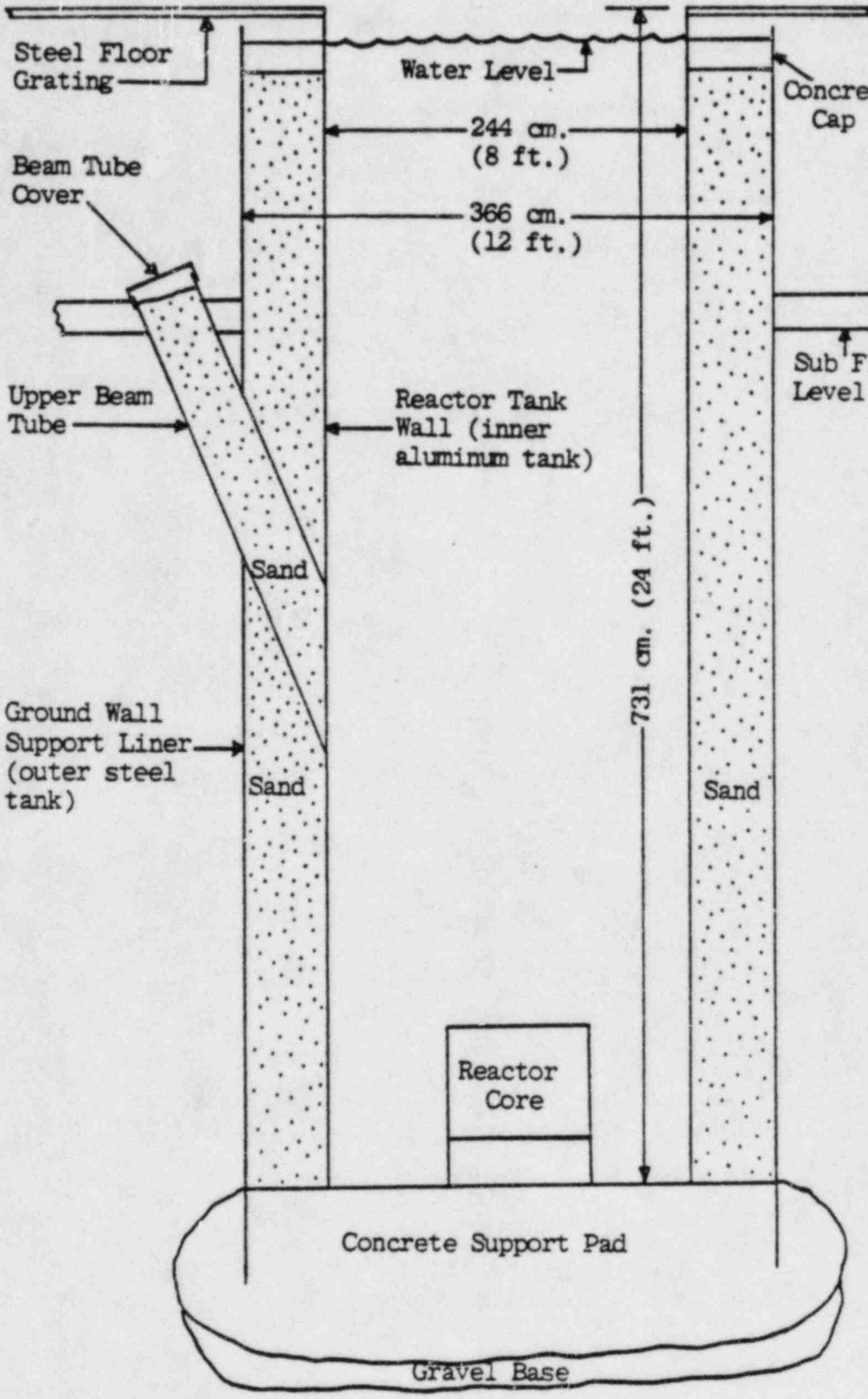
List of Reproducible Figures Needed for SER

1. Facility Floor Layout
2. Reactor Tank Cutaway View
3. Core Configuration (See Question #2)
4. Stainless-Steel-Clad Fuel Element Assembly
5. Aluminum-Clad Fuel Element Assembly
6. Primary Coolant Circulation and Purification Loops
7. Ventilation System
8. Control Rod Assembly
9. Control Rod Drive Mechanism
10. Block Diagram of Nuclear Instrumentation (Update)
11. Operating Ranges of Incore Nuclear Detector
12. Central Irradiation Facility, with multiple sample loading capability and rotation while being irradiated.

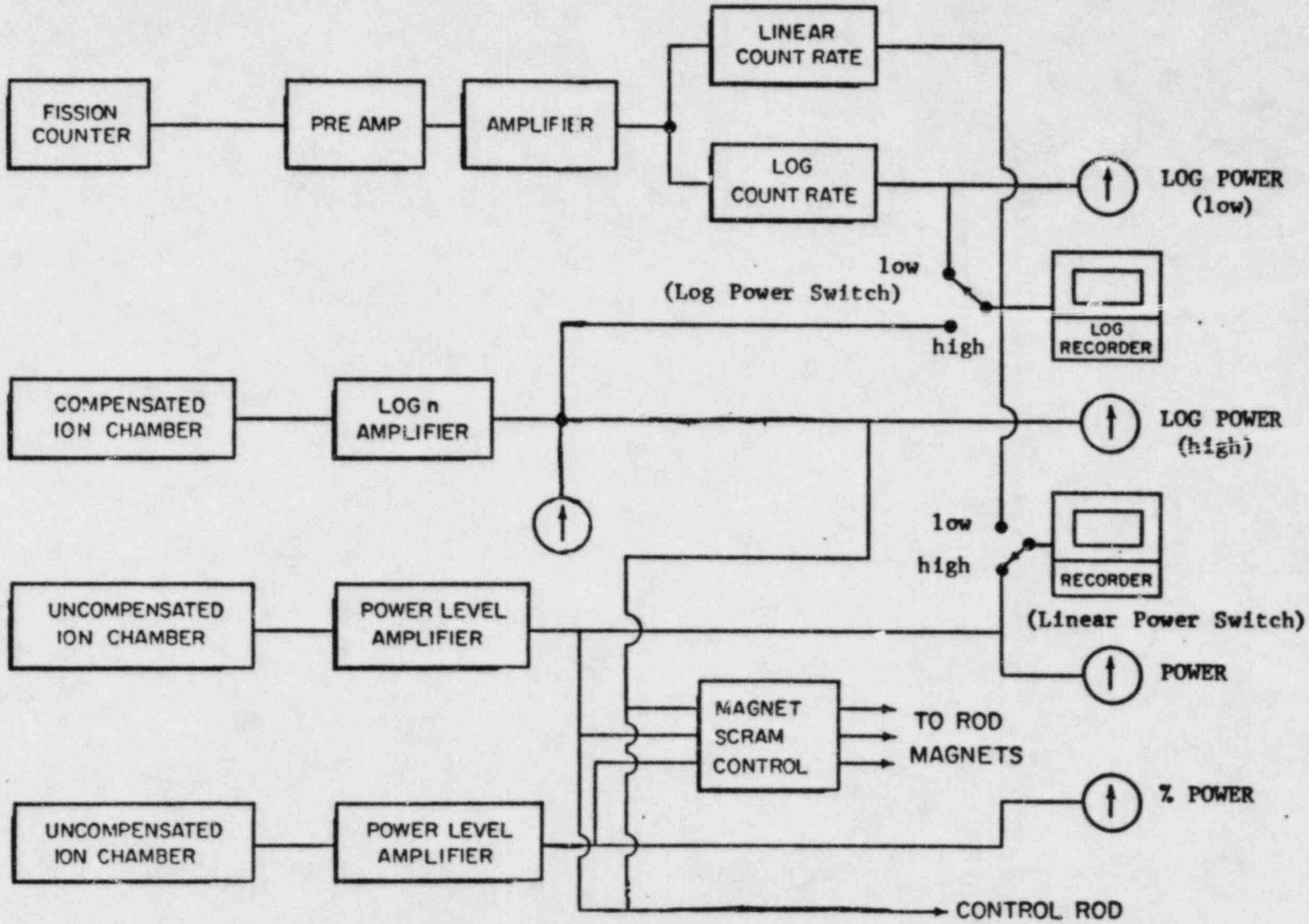
Some of these figures are provided in the proceeding question material



Facility Floor Layout



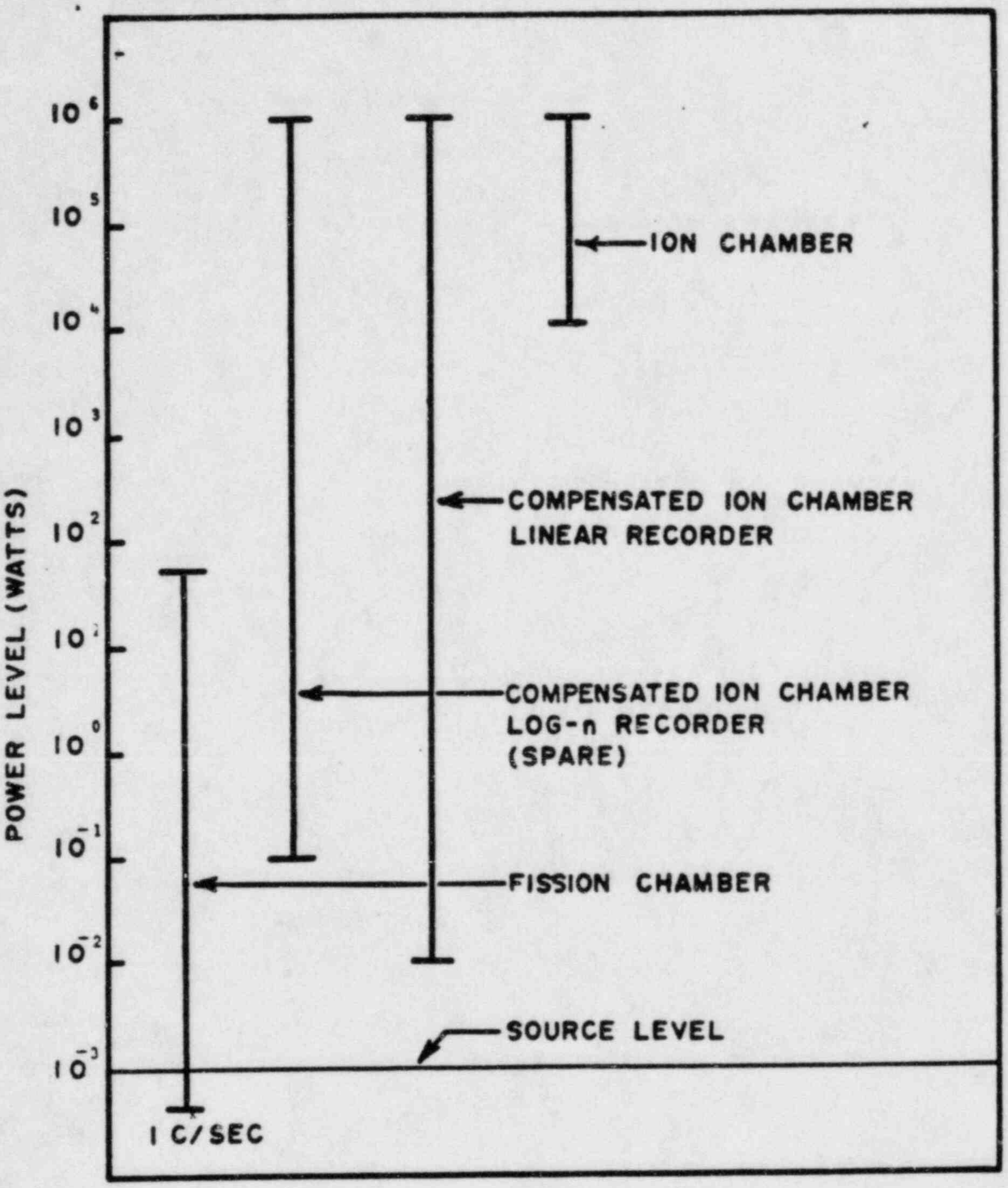
Reactor Tank Cutaway View



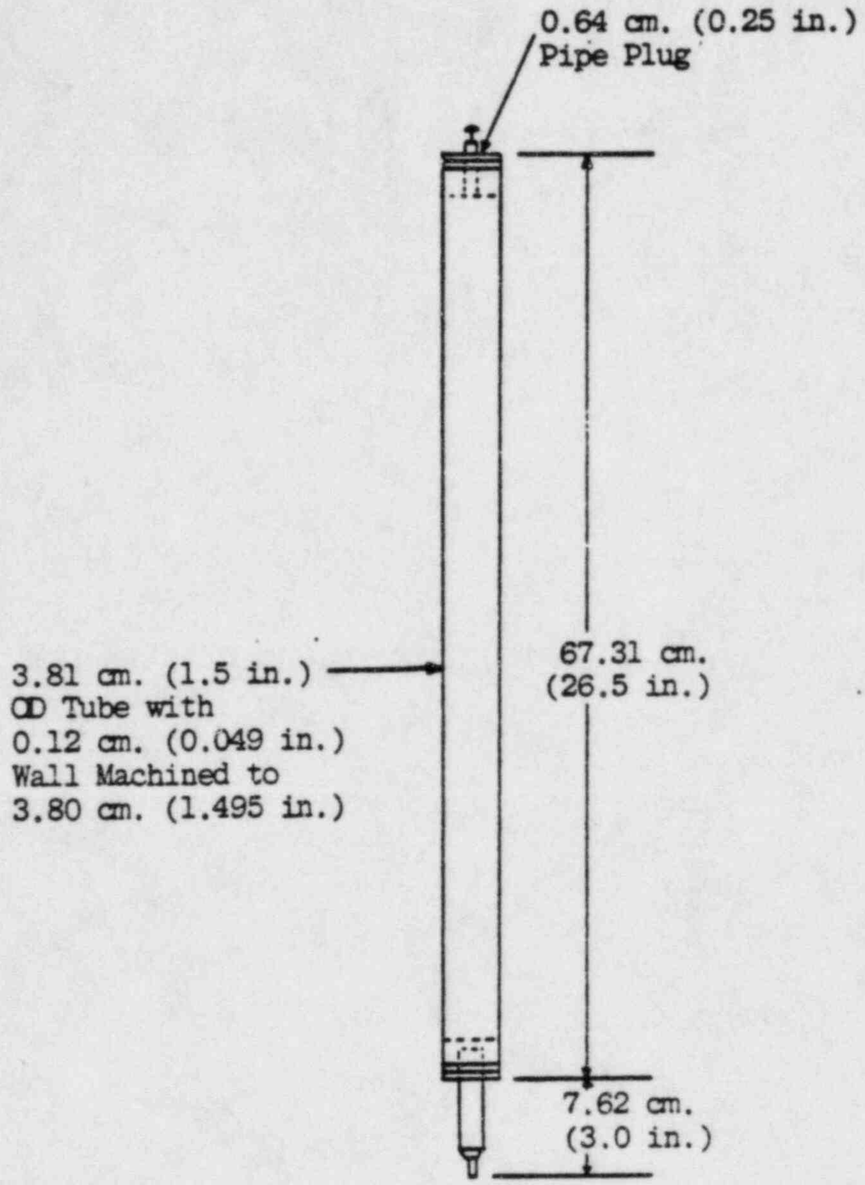
Block Diagram of Nuclear Instrumentation

Summary of Fuel Element Specifications

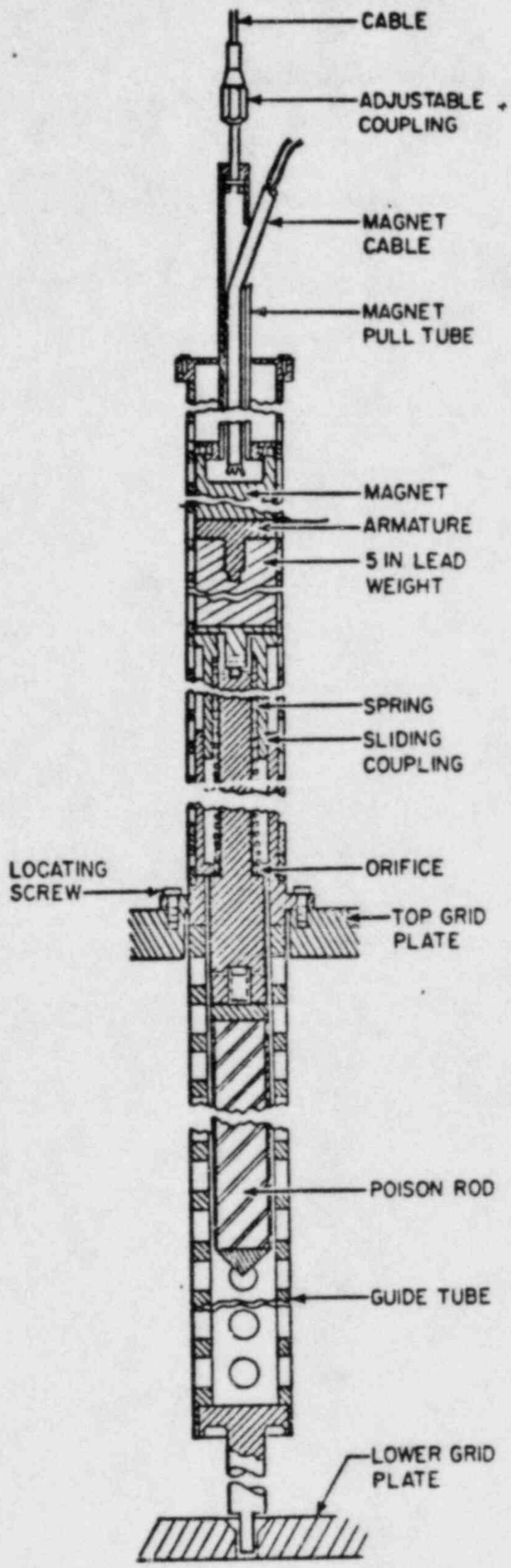
Item	Specification	
1. Element Type	Aluminum	Stainless Steel
2. Fuel-Moderator		
Material	$U\text{-}ZrH_n$	$U\text{-}ZrH_n$
H/Zr atom ratio (n)	1.0	1.7
Uranium content	8.0 wt-%	8.5 wt-%
Enrichment (U-235)	20 at-%	20 at-%
U-235 mass (per element)	37 gm	39 gm
Diameter of fuel meat	3.58 cm	3.63 cm
Length of fuel meat	35.56 cm	38.10 cm
3. Graphite End Reflectors		
Porosity	20%	20%
Length (each)	10.16 cm	8.74 cm
4. Cladding		
Material	Aluminum	S.S. 304
Wall thickness	0.762 mm	0.508 mm
Length	58.42 cm	58.42 cm
5. Overall Element		
Outside Diameter	3.73 cm	3.73 cm
Length	72.24 cm	72.06 cm
Mass	3.17 kg	3.63 kg



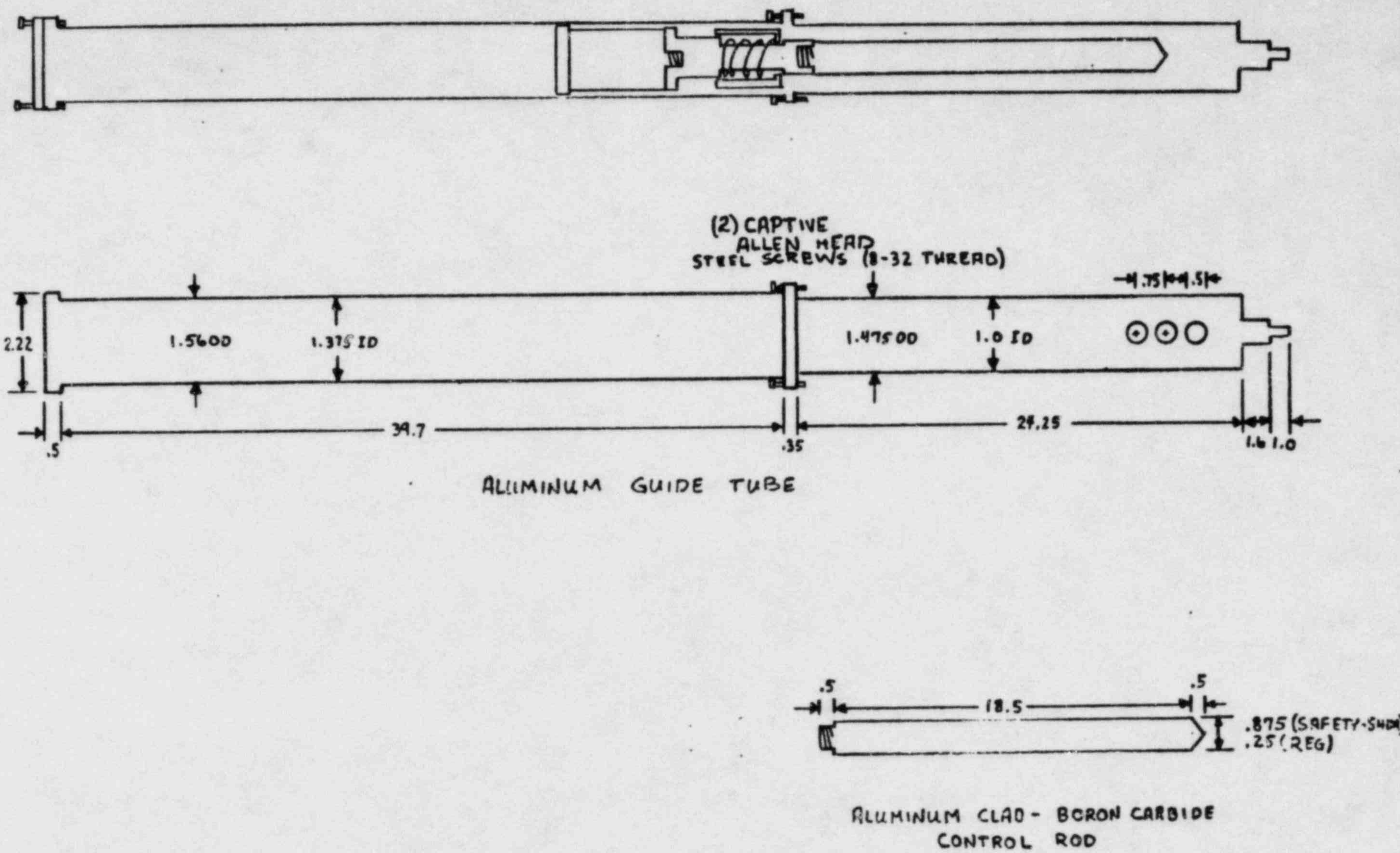
Operating Range of Incore Detectors



D_2O Moderator Elements



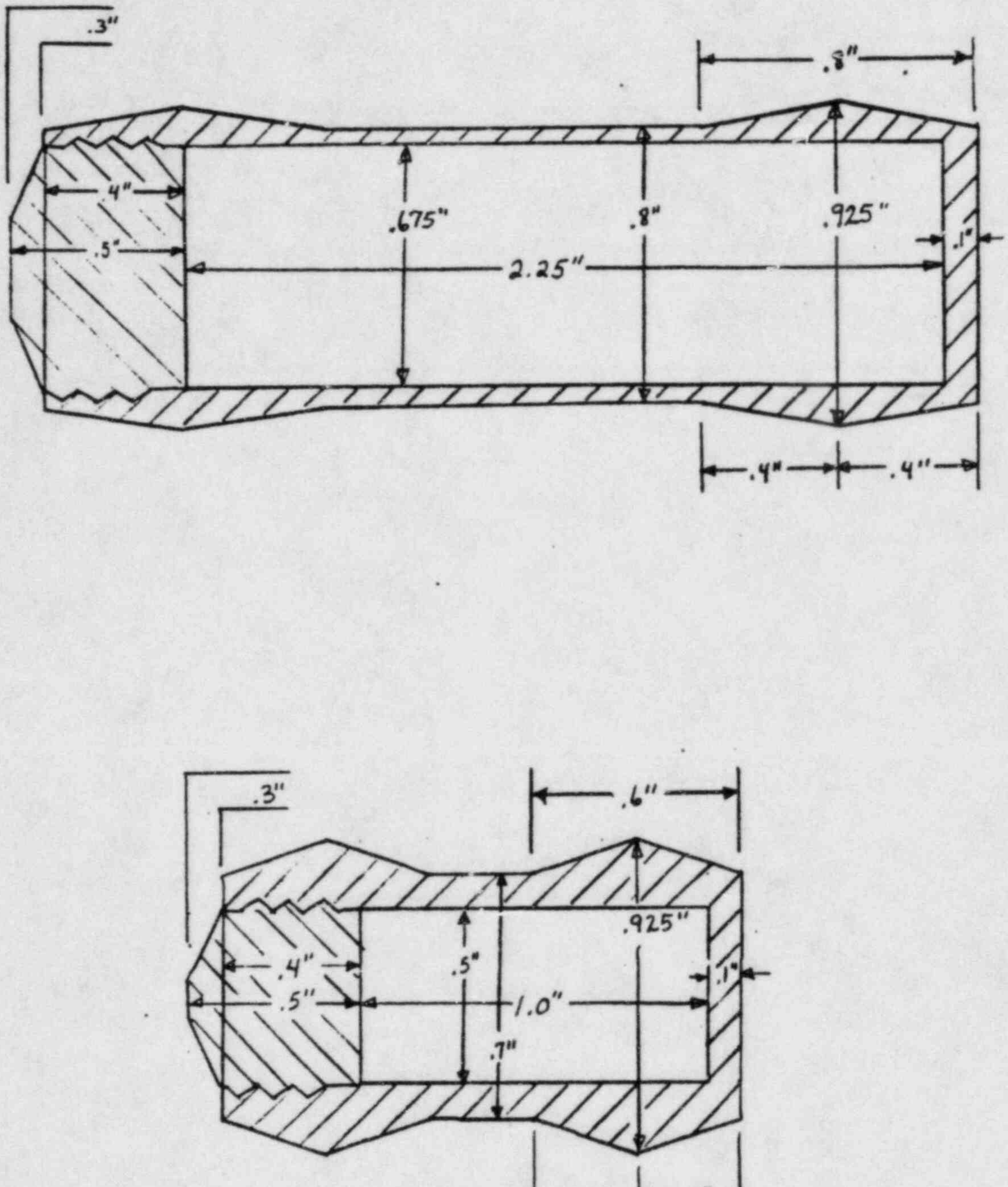
Control Rod Assembly



DECEMBER 28, 1983 ALL MEASUREMENTS IN INCHES.	CONTROL ROD MECHANISM UNIVERSITY OF UTAH TRIGA REACTOR NUCLEAR ENGINEERING LABORATORY	KEVAN CRAWFORD SENIOR REACTOR ENGINEER

Control Rod

$\pm .01''$



Polyethelene Irradiation Capsule

FUEL STORAGE ALLOCATION

	R1	R2	R3	R4	R5	R6	R7
1	272	419	355	438			293
2	356		344	~~293~~ 257			
3	346	436	305	360			
4	351	359	105	364			
5	527	898	109	2881			
6	340	427	190				
7	443	476	396				
8	433	428	371				
9	372	421	524				
10	370	528	298				
11	44[illegible]		379				
12	530		341				
13	416		343				
14							

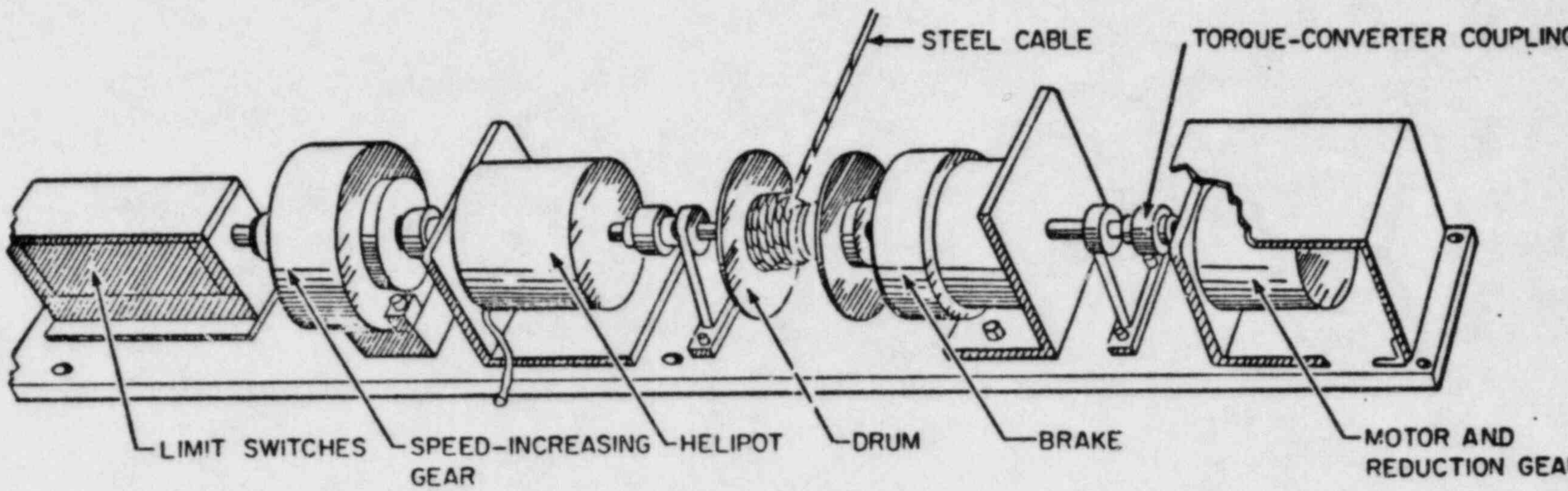
Fuel Storage Parts

South

193
252
295
361
3152 INST.
3153 INST.

Middle

North



Control-rod drive mechanism