THE UNIVERSITY OF TEXAS
COLLEGE OF ENGINEERING
AUSTIN, TEXAS 78712

*Department of Mechanical Engineering*
*Nuclear Engineering Program*
*512—471-5136*

February 27, 1985

Standardization and Special Projects Branch
Division of Licensing
Nuclear Regulatory Commission
Washington, D.C. 20555

Atten: Cecil Thomas
Re: Docket #50-602

Dear Sir:

Enclosed are the written replies to questions generated by the initial review and site visit of the proposed reactor facility at The University of Texas at Austin (application letter dated November 9, 1984).

Sincerely,

Thomas L. Bauer

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TLB:bb
Enclosure

cc: Dr. Gerhard J. Fonken
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THE UNIVERSITY OF TEXAS AT AUSTIN

Response to

Final Questions

1. An operational core of 86 fuel elements, 3 fuel followed control rods, and one air followed control rod is to be arranged in 5 rings with a central, water filled hole. Dimension of the active fueled core, a cylinder, is 15 inches in height with an average radius of 8.6 inches. The cylinder radius is calculated as the average radius of a hexagonal fuel array with 91 unit cells (6 filled rings including the center hole) and a unit cell area of 2.55 square inches.

2. Calculations indicate that no lead shielding is required to protect the reactor concrete shield from gamma heating at power levels of 1000 or 1500 kilowatts. Thermal capacity of the shield and effects of a 6.5 foot diameter cooling pool are sufficient for the proposed power levels of operation. Shadow shields of steel plate are installed in the concrete reactor shield around each beam port for the control of radiation fields at the shield beam access ports.

3. The typical TRIGA core unit cell has a water coolant volume to cell volume ratio of one-third. An operational reactor core will have basically the same reactor coolant to core volume ratio although differences of about one percent may be caused by the center hole or installed experiments in the reactor core.

4. A concrete foundation with a minimum thickness of 2 feet is planned for the area beneath the reactor tank. Actual thickness of the concrete foundation pad beneath the reactor tank may exceed the minimum thickness as required by the ground load conditions under the reactor tank or shield structure.

5. Storage facilities for fuel are provided inside and outside the reactor pool. Most routine fuel storage is intended to be inside the reactor pool with the storage outside the reactor pool utilized for isolation of damaged fuel elements, temporary storage of elements transferred to or from the facility, storage of new or expended fuel and emergency storage. Storage racks inside the pool for fuel storage will also provide temporary storage for some reactor experiment components. Storage wells outside the pool

for fuel storage may be used for routine storage of other radioisotope sources such as isotopic neutron sources.

Storage racks inside the pool are aluminum racks suspended from the pool edge by connecting rods. Elements are stored six per rack in a linear array. Each rack is 24" long by 12" wide by 3" deep and is generally located below more than 8 feet of water shielding. To facilitate extra storage, 2 racks may be attached to the same connecting rods by locating one rack at a different vertical level and offsetting the horizontal position slightly. For the current fuel inventory of 92 elements, 13 six element racks are available.

Storage wells outside the pool are pits in the reactor floor that are fabricated of 10" diameter stainless steel pipe. Six wells are designed each 15′ deep and located 3′ from the adjacent well. Nineteen elements may be stored in each well and water added for radiation shielding. An element spacing rack will provide an array storage for the fuel equivalent to the inner most 3 rings of the reactor core (includes one element in the center). Cover plates on the wells will provide access control and may include some shielding.

6. References 15 through 17 omitted from Chapter 3 of the Safety Analysis Report are:

(15.) Spano, A.H., "Quarterly Technical Report SPERT Project, April, May, June, 1964," ISO-17030.

(16.) Dee, J.B., et al., "Annular Core Pulse Reactor," General Dynamics, General Atomic Division Report GACD 6977 (Supplement 2), 1966.

(17.) Adler, J., et al., "Users and Programmers Manual for the GGC-3 Multigroup Cross Section Code," General Dynamics, General Atomic Division Report GA-7157, 1967.

7. The transient rod is a scrammable rod operated in both pulse and steady-state modes of reactor operation. During non pulse operation, the transient rod will function as an alternate safety rod with air continuously supplied to the rod. Rod position is thus controlled by operation of an electric motor that positions the air drive cylinder. The position of the

transient control rod and its associated reactivity worth will generally dictate removal of the rod as the initial step of a startup for steady-state operation.

8. Maximum licensed pulse reactivity insertion requested is 2.2% $\Delta k/k$ (3.14\$). A limit on the total worth of the transient rod is set at 2.8% $\Delta k/k$ (4.00\$) although the potential worth is expected to be about 2.6% $\Delta k/k$ (3.71\$) in a typical core configuration.

9. The reactor console system is currently being developed by the TRIGA manufacturer, GA Technologies. System checkout of the instrumentation, control, and safety system at the manufacturers site is planned as normal part of the system production. Detailed plans for the checkout procedures and final installation evaluation are to be developed as the system details and specifications become substantially complete. The enclosed checkout and evaluation plan is intended to be applied by both the system manufacturer (GA Technologies) and the system user (The University of Texas at Austin). Changes to the plan are expected to occur as more complete details of the system become available.

However, these changes will not decrease the scope of the present plan but may increase specified components of the plan. A report of results of the checkout and evaluation performed at GA Technologies will be sent to NRC before initiation of the startup program at The University of Texas at Austin. Results of the check out and evaluation at The University of Texas at Austin will be sent to NRC with the results of the startup program for the TRIGA reactor.

10. A startup plan is to be implemented at the TRIGA facility after initial tests of facility systems are complete. These initial tests will include function of room isolation and HVAC system, operation of instrumentation and control system, and operation of radiation monitoring system.

The first phase of the startup plan will be an approach to critical experiment. Configuration of the minimum critical core and inventory of the fuel loading will be documented for future reference. An estimate of integral control rod worths will be made from the criticality data.

The second phase of the startup plan will consist of determining major core parameters. Control rod calibration curves will be determined and adjustments made in the core excess reactivity. A thermal power calibration will be made for steady-state operation and a step reactivity insertion made for peak power and energy release of pulse operation. Final core parameters, configuration, loading, rods worths, excess reactivity, power calibration, and pulse data will be documented for future reference.

A third phase of the startup plan will consist of operational testing. Operational measurements will include characteristics (power, energy, temperature) of pulse insertions as a function of reactivity, instrumentation linearity and radiation doses around the reactor shield structure. Plots or tabulation of data will be documented for future reference.

Data obtained from the startup program will represent a nominal TRIGA reactor core for The University of Texas at Austin. The burnup characteristics of the core are expected to be that of a mid-life core as opposed to a freshly fueled facility.

11. References in section 4.1.3 of the Safety Analysis Report to forced convection cooling modes should be deleted. The design features of the control

system include adaptability to convective and forced flow TRIGA pool reactor types. Specific conditions, such as cooling mode, power level, number of control rods and so forth, are designated and fixed at the time of each control console installation. Forced flow cooling modes are not to be configured for The University of Texas at Austin installation.

12. Routine pulse operation occurs by insertion of reactivity into the TRIGA reactor with the power level initially established at some low level (< 1 kilowatt). However, circumstances may occur in which for experimental reasons a reproducible pulse condition is desired such that the transient rod is totally removed from the reactor core. Execution of such a pulse requires the insertion of an amount of reactivity to compensate for the total worth of the transient rod compared to the amount of desired pulse reactivity to be inserted. The procedure is to drive the reactor subcritical so that the total pulse rod reactivity can be inserted. The pulse characteristics will be determined by the net reactivity insertion that occurs above the amount necessary to compensate for the reactor subcritical state.

13. Ventilation dampers designed for isolation of air in the reactor room will fail to the closed position for a "fail-safe" type of operation. The major locations of isolation dampers in the reactor room are in the supply air HVAC system, reactor room unfiltered (large volume) exhaust air system, and filtered (small volume) air exhaust system.

14. An isolation signal to the reactor room air control dampers will shutdown the unfiltered (large volume) room air exhaust and initiate operation of the filtered (low volume) room air exhaust. Each exhaust is to be ejected vertically upward with a significant velocity and shall be exhausted from separate stacks. Filtration of the large volume reactor room exhaust will be fixed by normal HVAC operating requirements. Filtration of air in the small volume room exhaust initiated on shutdown of the main system will include a High Efficiency Particulate Air filter. A provision may exist for future installation of a charcoal filter in the purge air system. Manual operation of start/stop controls of both main and purge air systems will be available in the reactor control room.

15. No significant effects of the presence of a $Co^{60}$ irradiation facility in the reactor pool are anticipated that would impact the operation or safety of the reactor. The major anticipated effects are physical displacement of components caused by activities in the pool or the improper introduction of hazardous materials into the irradiator facility. Administrative controls are applied to prevent displacement of components or unsafe conditions caused by potentially hazardous materials introduced into the $Co^{60}$ irradiator. Periodic pool water samples or activity accumulations in the pool water purification system would provide detection of failure of the doubly encapsulated irradiator sources.

Radiation from the $Co^{60}$ source with no water shielding is estimated by assuming a 9000 curie point source located 10 feet from the top of the reactor tank. Th direct radiation dose calculated with $D = 6CE_n$ is $1.35 \times 10^3$ R/hr at the surface of the tank. Referring to the calculated doses for core radiation in a pool loss-of-coolant condition, the $Co^{60}$ dose would be roughly equal to the core dose after one hour but would not display the characteristic decay associated with the core fission products.

The scattered dose by comparison to the projected one hour core dose data would be 1 R/hr in the immediate vicinity of the tank opening. Thus, for a loss of coolant condition in the reactor pool, the radiation exposure at times greater than one hour after operation will be dominated by the $Co^{60}$ irradiator. At a source strength of 1000 curies (the approximate strength of the source to be initially installed), the dose is dominated by the irradiator at times greater than 1 day decay. In either case, dose exposures indicate that temporary shielding of the irradiator at the tank top or repositioning of the irradiator to the tank bottom will be initial considerations for loss of coolant conditions if the water shield cannot be re-established.

16. A temporary substitute for the air particulate monitor will be a portable $\beta$-$\lambda$ sensitive radiation monitor (preferably an ionization chamber).
The monitor will provide either an audible alarm or visual readout to the reactor operator. Manual operation of the reactor room ventilation start/stop control for air exhaust and purge will provide for room isolation. Temporary substitution of a portable

monitor for installed air particulate measurement equipment will not extend beyond a 24 hour period.

17. The reactor room exhaust stack for both main exhaust and purge exhaust of the reactor room will extend above the maximum elevation of the building. Maximum structure elevation is at least 44 feet above the mean grade level at the building site.

18. Insertion of 2.8% $\Delta k/k$ (4\$) as a step insertion for a reactivity accident analysis was determined by postulating the maximim reactivity value associated with a single component. The projected worth of the transient rod is ~2.6% $\Delta k/k$ and no other single reactor component or related experiment will have a reactivity worth in excess of the transient rod.

Simultaneous failures of multiple components such that a prompt critical condition in excess of that caused by the postulated 4\$ pulse would require nearly simultaneous event occurrences that are not considered credible. A substantial effort is made in reactors with very large reactivity insertions to obtain prompt critical conditions with the "simultaneous" insertion of more than one transient rod.

Typical pulse transients have half maximum values for the pulse width on the order of tens of milliseconds. For a multiple event transient accident the simultaneous events would have to occur within this time scale of fractions of a second. Any time separation of the events on the order of the pulse width would cause conditions less severe than the sum of $'s for the separate events.

The 4$ postulated insertion is approximately 10% greater than the expected worth of the single most reactive component in the reactor core. Two pulse analysis conditions, one a 4$ pulse from low power and one a 4$ pulse from high power, predict accident conditions for single event transient accidents. Although the insertion analysis applies only to single events, the results do provide data indicative of some multiple event situations. The reasons for this are that while no scenario for such an accident is presented the insertion of 4$ at high power levels vs. low power levels assumes that some reactivity (temperature compensated) has been inserted prior to the pulse. This reactivity insertion (3$ to reach steady-state) could be attributed to various insertion amounts of a previous transient with a corresponding temperature relationship. The exact relationship would depend on the delay time assumed between events.

19. The operational core configuration will contain two instrumented fuel elements with at least one located in the inner most reactor ring.

20. Maximum pulse power for a specified reactivity insertion is a function of several variables related to reactor core conditions. A first order approximation from the Fuchs-Nordheim model derives the peak pulse power as $P = C(\Delta k)^2/)2\alpha \ell)$ where C is the heat capacity available in the core, $\ell$ is the neutron lifetime, $\alpha$ is the prompt negative temperature coefficient and $\Delta k$ is the reactivity insertion above prompt critical. The core heat capacity is a function of the number of fuel elements (90 assumed). The value of $\ell$ varies by about 10% between a graphite and water reflected core. The value of $\alpha$ varies with temperature and is also sensitive to core reflection properties. Values applied in Chapter 7 of the Safety Analysis Report represent conservative exptremes of core reflection and temperature coefficient for purposes of accident analysis. For values expected to be

typical of the operational TRIGA core a peak power of 1500 MW is calculated.
( $\alpha$ = 1.3 x $10^{-4}$ ($\Delta$ k/k)/$^{o}$C; $\ell$ = 45 x $10^{-6}$ sec;
C = 8.9 x $10^{4}$ watt-sec/core; $\Delta$k = .014).

Slightly lower power levels are predicted if a modified pulse model is applied that includes the temperature variation of the fuel element heat capacity. A description of the dependence of various TRIGA fuel parameters has been documenced in GA-7882- Kinetic Behaviour of TRIGA Reactors by G.B. West and others (March 31, 1967).

21. Analysis of the $Ar^{41}$ concentration for air in the reactor room (Safety Analysis Report section 6.5.1.1) provides a conservative analytical calculation. Equation (15) page 6-24 of this analysis, however, assumed a very low (1.14 x $10^{5}$ $cm^{3}$/sec) room ventilation rate instead of the 2 air change/hour value of 2.29 x $10^{6}$ $cm^{3}$/sec. With the correction the concentration (eq. 16) become 3.34 x $10^{-7}$ $\mu$C/$cm^{3}$.

Subsequent examination of the $Ar^{41}$ analysis and knowledge of observed $Ar^{41}$ concentrations at similar facilities suggested an alternate analysis that is expected to estimate more accurately the concentration levels in the room (see enclosed supplement).

22. Data obtained from the National Flood Insurance Program indicates that no portion of the Balcones Research Center is located within either the 100 or 500 year flood zone. The 100 year flood base elevation line is 756 feet at the highest point of the Shoal Creek watershed that drains the Balcones Research Center site. Mean elevation of the area at the location of the proposed facility is ~795 feet. Ground water flooding of the site is to be prevented by control of site runoff and building or site design features. The gentle sloping characteristics of the immediate site vicinity provide ample gradient for establishing good water runoff. The site slopes downward to the west with a 3 to 4 foot change over the general area proposed for the reactor facility.

23. Facility design locates the base of the reactor tank 6 to 7 feet below the mean site grade level. Flooding of the first level area is possible if extreme conditions of local rainfall persisted such that designed ground water runoff was either

insufficient or failed. The actual drainage area available for water accumulation is 2400 square feet compared to the 7000 square feet of first level floor space. Facility features are designed to prevent local flooding for a specified expected rainfall. However, in the event that some flooding should occur of the facility first level, no safety problem effecting the reactor will occur.

Hazards of water on the floor of the reactor room consist of flooding of the storage wells and damage to equipment or systems located below the level of the experiment beam tubes. The beam tube center lines are about 2 feet 10 inches above the reactor room floor and water levels that deep are not considered credible. Although most installed equipment will be mounted on raised pads (6 inch height), components of the pool water cooling system or experimental equipment at reactor beam tubes may be subject to water damage. Electrical shorts and deposition of crud are the major concerns. No other reactor systems outside the pool are located below the beam tubes.

Other areas such as the building utility supply area on the first level are susceptible to water damage primarily as damage or shorts to pump, motors or electrical equipment. Loss of building utility services do not effect safety of the reactor since no forced flow systems are required for cooling and reactor shutdown is automatic with any loss of power.

24. The Nuclear Engineering Teaching Laboratory building has been designed for 70 mile per hour winds in accordance with requirements of the Uniform Building Code. Design conditions include factors for wind gust conditions in excess of the 70 mph value. The building is constructed in a zone 0 seismic zone and is designed only for combined gravity and wind loads. Seismic forces in zone 0 are such that combined gravity and wind load stresses exceed those anticipated by seismic events.

Data in Chapter 2 of the Safety Analysis Report provide expected local wind and seismic conditions. The peak wind recorded in Austin, Texas was 57 mph (Table 2-2) in February, 1947. Tornadic activity at the site is roughly one event per year per 1000 square miles (Figure 2-9) or 4 x $10^{-6}$ per year for an area that is 333 foot square (roughly equal to the general site area). Judgements related to the site area, potential structural damage, and possibility for fuel element damage indicates that the rate per year is probably at least 2 orders of magnitude less for an event that might cause a significant release of radioactivity. The largest intensity earthquakes within the state (excluding one event in far West Texas) are of magnitude VI on the Modified Mercalli scale. Accelerations for this magnitude are about 0.7 g with no damage to buildings of good construction expected.

SUPPLEMENT TO FINAL QUESTIONS

Argon-41

Analysis of Concentration in Reactor Room
from Pool Water Releases

The University of Texas at Austin

TRIGA Reactor

January 1985

## Release of Argon-41 from Reactor Pool Water

Dissolved gases in the reactor coolant contain the radioactive noble gas argon-41. The release of argon-41 activity from the coolant depends on the gaseous exchange rate at the air-water interface and the change in gas solubility as a function of temperature.

According to Henry's law for gases in contact with liquids the equilibrium concentration in the liquid is proportional to the partial pressure of the gas. The saturated concentration of argon in water at one atmosphere of standard air is given in Table 6-1.

Table 6-1
SATURATED ARGON CONCENTRATION IN WATER [2]

| Temperature ($^{o}C$) | S(atoms A-40/$cm^3$ $H_2O$) |
|---|---|
| 10 | $1.14 \times 10^{16}$ |
| 20 | $0.94 \times 10^{16}$ |
| 30 | $0.79 \times 10^{16}$ |
| 40 | $0.69 \times 10^{16}$ |
| 50 | $0.62 \times 10^{16}$ |
| 60 | $0.56 \times 10^{16}$ |
| 70 | $0.52 \times 10^{16}$ |
| 80 | $0.48 \times 10^{16}$ |

Concentrations at equilibrium conditions for argon in air that is in contact with the water depends on the air volume, air exchange rate, water surface volume and the water-air exchange rate. Argon-41 activities at conditions of equilibrium concentration are functions of the pool water volume, reactor core volume, water flowrates through the core, and the production and decay rates for the radioisotope.

The argon-40 concentration in the water at the core inlet temperature (38°C) is

$$N_1^{40} = 7.1 \times 10^{15} \text{ atoms/cm}^3,$$

and the concentration of argon-40 in the water at the core exit temperature (68°C) is

$$N_1^{40} = 5.3 \times 10^{15} \text{ atoms/cm}^3.$$

From the reactor power the cycle times for argon exposed in the core and circulated in the pool are estimated. The volume flowrate is given by

$$\bar{v} = Q/(P_w C_p \Delta T),$$

where:

$\bar{v}$ = Volume flowrate of water through the core
$Q$ = Reactor power level ($10^6$ watts)
$P_w$ = Pool water density (0.96 gms/cm$^3$)
$C_p$ = Specific heat of water (4.2 watt sec/gm °C)
$\Delta T$ = Temperature rise in core (30°C).

thus:

$$\bar{v} = 10^6/(0.96 \times 4.2 \times 30) = 8.3 \times 10^3 \text{ cm}^3/\text{sec}$$

The exposure, t, and cycle, T, times in the core are calculated from

$$t = V_c/\bar{v} = A_f L_c/\bar{v},$$

and $$T = V_p/\bar{v} = (\pi h(w/2)^2 + h(1-w)/\bar{v},$$

where:

$A_f$ = flow channel area (485 $cm^2$),
$L_c$ = length of flow channel (38.1 cm),
$V_c$ = volume of water in the core,
w = width of reactor pool (200 cm),
l = length of reactor pool (300 cm),
h = height of pool water (750 cm).

Exposure time, t, is about 2.2 seconds and cycle time, T, is about 2.8 x $10^3$ seconds.

Argon atoms exchanged at the water-air interface depend on a water thickness depth that is small relative to the pool dimensions and, therefore, a small fraction of the available saturated argon is exchanged with the air. During the time required for the pool water to circulate once through the reactor core (less than one hour), the argon equilibrium concentration is depleted to the lowest solubility value for equilibrium concentration. The argon release as a function of temperature and solubility thus approaches zero. This depletion occurs as the activity of the argon radioisotope increases but is substantially complete (about 45 minutes) before the argon-41 activity reaches half the equilibrium value (about 110 minutes).

Evaluation of the water-air interface exchange rate for argon is related to an air and water thickness depth that depends on the argon atom diffusion coefficient. The total exchange rate then is a function of the pool surface area, $A_s$, and an effective release volume $V_i'$. The two terms are related by

$$\beta_i A_s = f_{i \to j} V_i' \quad ,$$

where

$\beta_i$ is a surface exchange coefficient (cm-$sec^{-1}$), and $f_{i \to j}$ is the fraction of atoms exchanged from volume i to j ($sec^{-1}$).

Estimates of the surface exchange coefficient (i.e., the gas in a unit volume that is exchanged at the surface per unit time per unit surface area) for

Solving for $f_{j \to i}V'$ gives

$f_{j \to i}V_j' = f_{i \to j} V_i' (N_i/N_j)$

The following calculations were performed to evaluate the rate of Ar-41 escaping from the reactor pool water into the room enclosure. The calculations show that the Ar-41 decays while in the water, and most of the radiation is safely absorbed in the water. The changes in Ar-41 concentration in the reactor, in the pool water external to the reactor, and in the air of the room enclosure are given by

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \Phi N_1^{40} \sigma^{40} - N_1^{41} (v_1 + V_1 \Phi \sigma^{41} + \lambda^{41} V_1) + N_2^{41} v_1 \quad (1)$$

$$V_2 \frac{dN_2^{41}}{dt} = - \lambda^{41} N_2^{41} V_2 + v_1^{41}(N_1^{41} - N_2^{41}) - (f_{2 \to 3}N_2^{41} V_2 - f_{3 \to 2}N_3^{41} V_3) \quad (2)$$

$$V_3 \frac{dN_3^{41}}{dt} = (f_{2 \to 3}N_2^{41} V_2' - f_{3 \to 2}N_3^{41} V_3') - N_3^{41} (\lambda^{41} V_3 + q) \quad (3)$$

where

subscript 1 = Reactor region (water region in core)
subscript 2 = Reactor-tank water region external to the reactor
subscript 3 = Room enclosure region
superscript 40 = Ar-40
superscript 41 = Ar-41
superscript A = Ar-40 plus Ar-41
$V$ = Volume of region ($cm^3$)
$N$ = Atomic density ($atoms/cm^3$)
$\lambda$ = Decay constant ($sec^{-1}$)
$\sigma$ = Absorption cross section ($cm^2$)
$q$ = Volume flow rate from room enclosure exhaust ($cm^3/s$)
$v_1$ = Volume flow rate through region No. 1 ($cm^3/s$)
$\bar{\phi}$ = Average thermal neutron flux in region No. 1 ($n/cm^2 \times s$)
$f_{i \to j}V_i'$ = Fraction of Ar-41 atoms in region that escape to region j per unit time ($s^{-1}$)

The values of constants in equations (1), (2) and (3) are

$$\bar{\phi} = 1.2 \times 10^{13}\ n/cm^2 \cdot s$$

$$\sigma^{40} = 0.47 \times 10^{-24}\ cm^2$$

$$\sigma^{41} = 0.060 \times 10^{-24}\ cm^2$$

$$\lambda^{41} = 1.06 \times 10^{-4}\ sec^{-1}$$

$$q = 2.29 \times 10^{6}\ cm^3/s$$

$$v_1 = 8.13 \times 10^{3}\ cm^3/s$$

$$V_1 = 1.85 \times 10^{4}\ cm^3$$

$$V_2 = 1.03 \times 10^{8}\ cm^3$$

$$V_3 = 4.12 \times 10^{9}\ cm^3$$

Equation (1) can be reduced to

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \bar{\Phi} N_1^{40} \sigma^{40} - (N_1^{41} - N_2^{41}) v_1 \qquad (4)$$

by $v_1 + V_1 \bar{\Phi} N_1^{41} + \lambda V_1 \simeq v_1$.

During equilibrium conditions the three equations reduce to:

$$V_1 \bar{\Phi} N_1^{40} \sigma^{40} = (N_1^{41} - N_2^{41}) v_1 \qquad (5)$$

$$N_2^{41} [\lambda^{41} V_2 + f_{2\to 3} V_2] = (N_1^{41} - N_2^{41}) v_1 + f_{3\to 2} N_3^{41} V_3 \qquad (6)$$

$$N_3^{41} [\lambda^{41} V_3 + q + f_{3\to 2} V_3] = f_{2\to 3} N_2^{41} V_2 \qquad (7)$$

Combining equations (5) and (6) gives

$$N_2^{41} = \frac{V_1 \bar{\Phi} N_1^{40} \sigma^{40}}{\lambda^{41} V_2 + f_{2\to 3} V_2'} + \frac{f_{3\to 2} N_3^{41} V_3'}{\lambda^{41} V_2 + f_{2\to 3} V_2'} \qquad (8)$$

argon vary considerably. One method of arriving at a value for this parameter is through the diffusion coefficient of the gas in water. The mean square distance traversed by a molecule is

$$\langle \Delta X \rangle^2 = 2Dt$$

where D = diffusion coefficient ($cm^2/sec$),

t = time (sec).

The exchange coefficient is assumed to be evaluated for 1 sec. as

$$\beta = (\langle \Delta X \rangle^2)^{1/2}/t = (2D/t)^{1/2}$$

The diffusion coefficient at 40°C is about $1.1 \times 10^{-5}$ $cm^2/sec$, and, if one assumes that only one-half of the argon atoms within one diffusion length of the surface escape,

$$\beta = 1/2 \quad (2 \times 1.1 \times 10^{-5})^{1/2} = 2.35 \times 10^{-3} \text{ cm/sec}$$

Values for the surface exchange coefficient have been reported by Dorsey [3] for air, $O^2$, and $N^2$. The values for these three gases are all about equal. Assuming argon behaves as do these gases, a value is obtained of $5.7 \times 10^{-3}$ cm/sec for $\beta$.

Measurements have been made of the argon-41 activity in a TRIGA Mark III reactor pool and from the data acquired from these measurements it was possible to construct a value for the surface exchange coefficient. This value at 40°C is about $2.9 \times 10^{-4}$ cm/sec.

During equilibrium conditions and assuming no difference in the rates of escape fractions for Ar-40 and Ar-41, the number or argon atoms that escape from the water into the air equals the number of argon atoms that enter the water from the air, i.e.,

$$f_{i \to j} V'_i N_i = f_{j \to i} \, V'_j N_j$$

where $N_j = 2.1 \times 10^{17}$ argon atoms/$cm^3$ of air $\sim N^{40}$

$N_i = 7.1 \times 10^{15}$ argon atoms/$cm^3$ of water $\sim N^{40}$

which inserted into equation (7) for $N_2$ yields

$$N_3^{41} \left[ \frac{\lambda^{41} V_3 + q + f_{3\to2} V_3'}{f_{2\to3} V_2'} - \frac{f_{3\to2} V_3'}{\lambda^{41} V_2 + f_{2\to3} V_2'} \right] = \frac{V_1 \bar{\Phi} N_1^{40} \sigma^{40}}{\lambda^{41} V_2 + f_{2\to3} V_2'}, \quad (9)$$

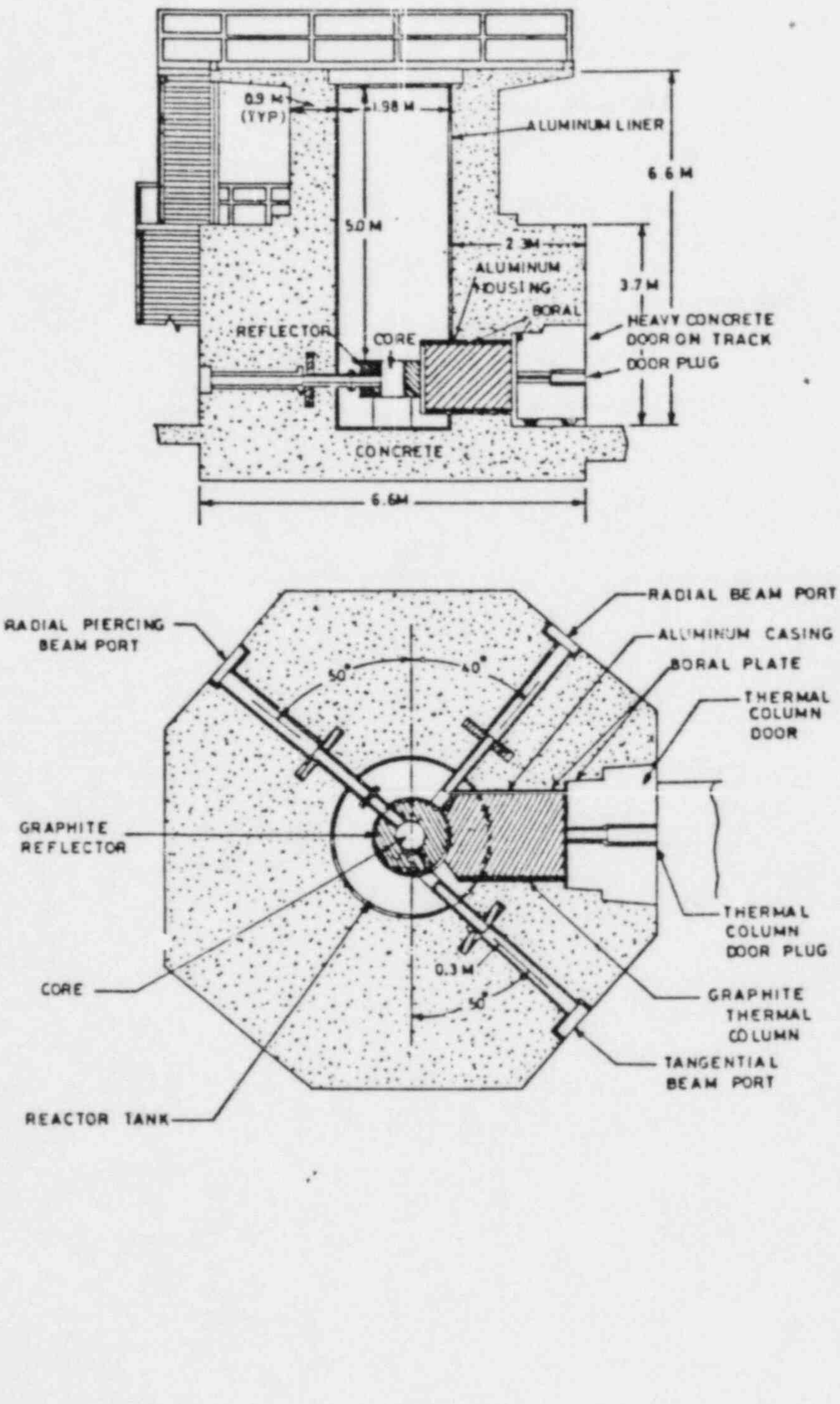
Solving for $N_3$ yields 7.4 atoms/cm$^3$. This corresponds to a concentration of Ar-41 activity of

$$A^{41} = \frac{\lambda^{41} N_3^{41}}{C} = \frac{1.06 \times 10^{-4} \times 7.4}{3.7 \times 10^{4}} = 2.12 \times 10^{-8}\ \mu Ci/cm^3$$

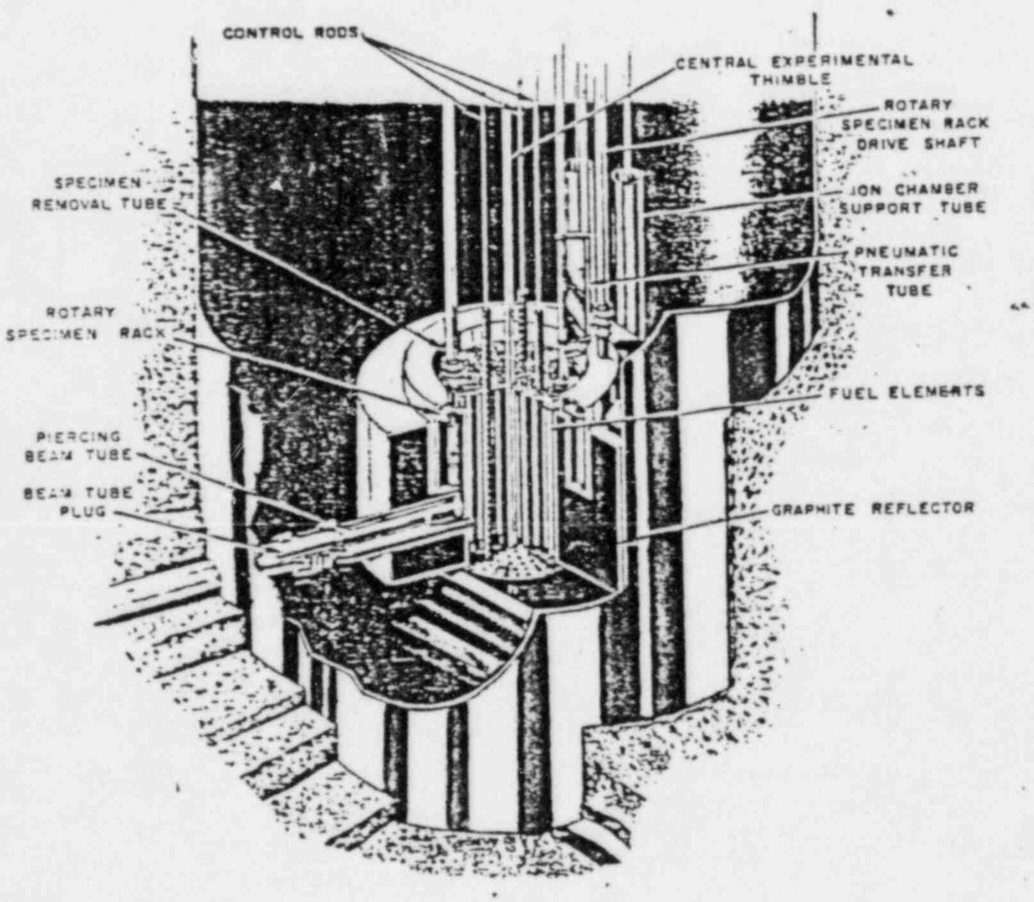
where

$A^{41}$ = Ar-41 concentration, $\mu$Ci/cm$^3$,
C = Conversion factor from disintegrations/s to $\mu$Ci

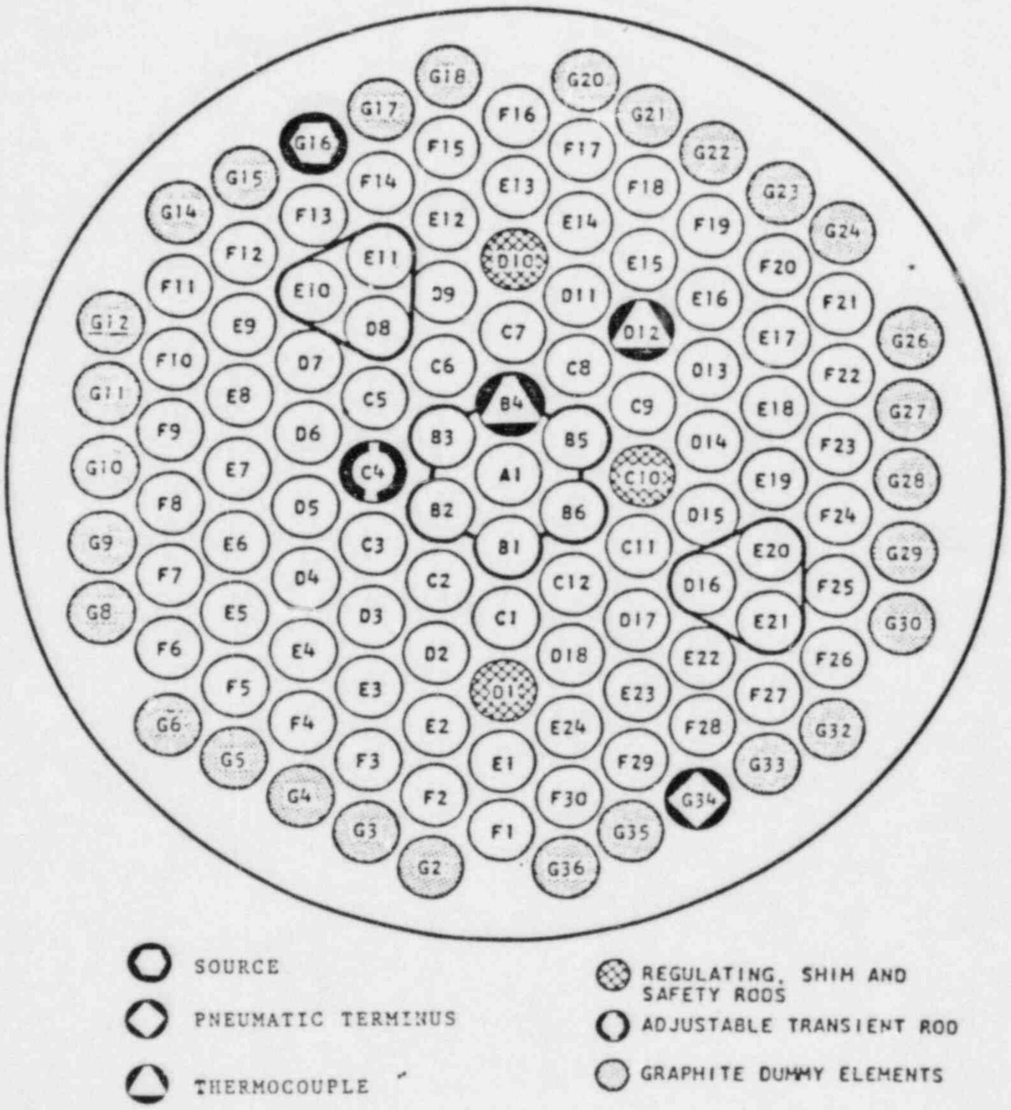
This is below the RCF limit recommended by 10 CFR Part 20 for unrestricted areas ($4 \times 10^{-8}$ $\mu$Ci/cm$^3$) and takes into account no dilution, which is conservative.




TYPICAL MARK II TRIGA REACTOR
WITH THERMAL COLUMN
Figure 3-25




TYPICAL TRIGA MARK II REACTOR




CORE ARRANGEMENT
Figure 3-27

TRIGA System Console

Instrumentation, Control & Safety System
Plan for Evaluation and Checkout

I. System console calibration - Initial system configuration and installation is to require a console calibration that will subsequently be performed for periodic system checkout. The calibration will consist of specification and measurement of parameters at key test points for analog subsystems and the identification and execution of test algorithms for digital subsystems.

A. Test points for analog specifications will include parameters to be measured and tolerances.
   1) Crucial voltages
   2) Important waveforms
   3) Critical frequencies

B. Test algorithms for digital execution will include component identification and functions tested.
   1) CPU instructions and RAM memory storage
   2) Benchmark values for check sums of programmed ROMs and fixed parameters.
   3) Communication and function of all I/O interface systems and peripheral equipment.

II. Instrumentation system - Reactor measurement channels consisting of detector, electronics and readout are to be checked out by observation of channel response to signals verified by a system console calibration and indications compared to known conditions.

A. Parameter measurement channels and readout indication will be identified for each of the following:
   1) Neutron flux - two channels
   2) Fuel temperature - two channels
   3) Rod position - four channels
   4) Auxiliary measurements
      a) water systems - pool bulk temperature and others.

b) radiation monitors - area gamma and others.

c) other facility parameters.

5) Status conditions - switches, indicators, alarms.

B. Methods of comparison applied to establish correct indication of true conditions by each instrumentation channel.

1) Comparison to alternate measurement such as power by thermal response of bulk pool temperature.
2) Comparison to calibrated instrument such as current, voltage or frequency source.
3) Comparison to sensory observations such as state of system; up or down; closed or open; off or on.
4) Identification of point (linearity assumed) or points (linearity checked) at which indication is verified.

III. Control System - Operation of the control rod drive system is to be checked out by observation of rod drive response to control requests from the control switches.

A. Each operating mode will be specified and the logic conditions necessary for operation of the control rod drive tabulated and tested.

1) Transient rod - air supply and cylinder position.
2) Safety rod - electric motor operation.
3) Shim rod - electric motor operation.
4) Regulatory rod - electric motor operation.

B. Functional tests of the rods will be determined by observing operation of limit switches. Withdrawal times for each rod will be specified and measured.

1) Transient rod - air action or motor action.
2) Safety rod - motor action.
3) Shim rod - motor action.
4) Regulatory rod - motor action.

IV. Safety system - Operation of the safety system is to be checked out by actuation of control rod insertion and timing of control rod drop time.

A. A functional and circuit diagram of the safety system will be examined to determine the system function in response to each of the circuit components.
   1) A tabulation will be made of all trip functions.
   2) Points for each trip condition will be set.
   3) Removal of rod drive current will be documented for each trip.

B. Scram action on each of the control rods will include the measurement of rod drop time.
   1) Each rod will be tested for drive release by action of the scram circuit.
   2) Drop time for each rod of less than 1 second shall determine the rod scram function.

V. Special system features - The combined functions of analog and digital electronic equipment provide several features that improve the reactor console operation and assist operator's evaluation of system performance.

A. Routine system functions performed by programmable code will be executed and results verified.
   1) Programmable code functional checks will include system and subsystem components.
   2) Examples of system operations are display function, data printout, data retrieval from storage, and calculations such as reactivity.
   3) Examples of subsystem operations are set point checks and status conditions checks of subsystems or communications.

B. Readouts designated for backup indication or failure of the CRT display system are to be compared with CRT display system values.

C. Other system basic features such as clock frequency, battery backup power, and line power will be evaluated as appropriate for the function of the control system console.

D. Specific codes, standards, or guides applied to circuit design, component manufacturer or installation are to be identified and documented.

E. Each of the input detectors or types of sensors for the console control system are to be identified and performance specifications or other criteria established for their acceptance.