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UNIVERSITY OF MARYLAND

CENTRAL ADMINISTRATION
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December 12, 1983

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

Dear Mr. Thomas:

We are submitting herewith, answers to the 39 questions concerning the MUTR Safety Analysis Report requested by your staff on October 31, 1983.

If you require any additional information please call Dr. Frank Munno or Dr. Ralph Belcher at 454-2436.

Sincerely,

John S. Toll

John S. Toll
President

JST:jf

Enclosures

cc: Dr. David S. Sparks
Dr. John B. Slaughter
Dr. Samuel Price
Dr. Frank Munno
Dr. Ralph Belcher

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Q-1.

(a) What are the principal uses of the MUTR?

(b) What is the current utilization in megawatt-hours per year?

Ans-1.

(a) The principal uses of the MUTR are training, nuclear reactor engineering education and a source of neutrons for activation analysis.

(b) The current utilization is between 10 and 15 megawatt-hrs. per year.

Q-2.

The SAR states that the fuel/moderator diameter is 1.41 in. and the corresponding outer rod diameter is 1.45 in.

However, Figs. 4-1 and 4-2 show the outer rod diameter to be 1.41 in., with an implied fuel meat diameter of 1.37 in. Confirm the dimensions of your fuel rods.

Ans-2

The data shown in Fig. 4-1 are correct. The outer rod diameter is 1.41". The S.S. cladding is 0.02". Thus the fuel (meat) diameter is 1.37".

The diameter listed on page 4-1 under section 4.1,2 is in error and will be corrected.

Q-3.

Specify the weight and ^{235}U content of a fuel rod.

Ans-3

The overall weight of each fuel rod is 7.5 lbs., and the ^{235}U content is 37 grams.

Section 4.1.2 FSAR page 4-2 erroneously gives the weight of a 4-cluster bundle. This will be corrected.

Q-4.

Describe the graphite reflector assemblies.

Ans-4

The graphite reflector consists of a solid slug of reactor grade graphite, sealed in a No. 27 Gage (0.016") 1100 aluminum

alloy rectangular can, 2.812" on an edge and 25-1/8" long. A bottom adaptor is designed to fit into and rest on the existing MTR-type grid plate which provides vertical support and spacing.

A foldback aluminum bail is attached to the top of the reflector can to facilitate handling by the existing fuel cluster handling tool.

Ref: Allis-Chamers, Graphite Reflector

University of Maryland, Drawing No. 15 C-05003, 11/3/59.

Q-5

Describe the support method for the bottom grid plate. Provide a figure if available.

Ans-5

The grid plate support for the TRIGA fuel is the original MTR fuel support. We are submitting the original plans for the grid plate and the grid plate supports. (See attachment 1)

Q-6.

Describe the two individual safety channels and specify the detector type of each. Discuss the functional operation of these channels.

Ans-6

Detector Type

Linear Power (Safety) Channel I: Fission Chamber

Linear Power (Safety) Channel II: Ionization Chamber

Functional Description

Monitor the neutron power level to provide linear indication in percent of full power (scale 0 to 150%) and to provide a signal to the reactor scram circuit to scram the reactor when reactor power exceeds 120% of full power.

Design Description

The Safety Channel is a D.C. current measuring instrument specifically designed for use with nuclear reactors. It consists of two identical channels, "1" and "2", receiving signals from separate ionization chambers located in the neutron field to be measured, followed by various special subcircuits to process and evaluate the information received. All signals, except those from the detectors, that enter or leave the assembly are isolated from essential internal circuits to prevent external fault currents from destroying vital circuit functions.

The Linear Amplifier is the input stage of each subchannel. Within this circuit the ionization chamber current is converted to a proportional voltage output signal. Full scale output of 10V (equivalent to 150% reactor power) may be obtained with input currents between 3×10^{-3} amperes and 2×10^{-5} amperes with 1/2% accuracy.

Test and calibration signals are built into each channel and are controllable from the front panel.

The circuits are all solid state utilizing high quality epoxy glass plug-in printed boards. Many use only monolithic integrated circuits as the active elements.

Primary power for the channel is obtained from the 115V A.C. mains. Modular ± 15V power supplies are included within the drawer assembly along with an adjustable modular high voltage supply.

Circuit boards and switches are interlocked to indicate removal of a board or incorrect position of a switch.

Q-7.

What are the measured excess reactivity above cold, clean critical and control rod worths in your current core?

Ans-7

Control rod worth

Shim 1 - \$2.80

Shim 2 - \$3.00

Reg. Rod - \$2.42

$\beta = 0.007$

Excess reactivity 1.21% $\Delta k/k$

Ref. Measurements performed July 19, 1983

Q-8

Comment on the ability of the reactor components and systems to continue to operate safely and withstand prolonged use over the term of the requested license renewal. Include the potential effects of aging on fuel elements, instrumentation, and safety systems. Discuss potential impact on fuel-cladding strength assumed in your safety analysis, Chapter 11 of your FSAR.

Ans-8.

The reactor pool tank is made of 3/8" Al 6061 T6 plates on the sides and 1/2" aluminum plate on the bottom. There are five flanged nozzles on the tank. Four are for the beam tubes and through tube and the fifth is for the

thermal column. [See Sections 3.4 and 3.5 of the FSAR].
Prior to start-up the tank is visually inpsected from the reactor bridge for obvious defects. The through and beam tubes as well as the thermal column are inspected for obvious water leaks during the walk through inspection prior to start-up.

Fuel elements are inspected routinely every two years [See Section 4.1 Tech Specs) and the Control rods on the same basis. [See Section 4.2 Tech Specs]

Each safety channel's ability to perform according to specifications is verified prior to start-up.

TRIGA fuel is known to operate in excess of 15 years at power greater than one megawatt. In addition the GA Mark F has operated for longer than 750 megawatt-days without significant fuel degradation. At the licensed power of the MUTR this amounts to over 20 years of operation on an 8 hr/day schedule at 250 kW.

REF. Private communication: Gordon West, Nov. 1983.

With respect to radiation embrittlement, sources of information dealing with radiation damage to materials that were reviewed in connection with this issue, indicate that below a fast neutron fluence of 1.0×10^{20} n/cm^2 the mechanical properties of stainless steels and aluminum (the materials that provide structural support within the MUTR TRIGA reactor) are essentially unaffected. In fact, at a fast neutron fluence level of 1.0×10^{21} n/cm^2, both aluminum and stainless steel have ductilities that are said to be "reduced but not greatly impaired."

Moreover, the yield strength of stainless steel actually increases at an NVT of approximately 3.0×10^{20} fast n/cm^2.

Using a conservative value for fast neutron fluence, at which the mechanical strength properties of both aluminum and stainless steel are assumed to begin degrading, of 1.0×10^{20} n/cm^2 and using a conservative annual burnup of 15 MW-hours per year, and finally using a conservative value for the MUTR reactor fast flux of 5.0×10^{12} $n/cm^2/sec$ while operating at 250 KW(t), indicates that the MUTR reactor facility structures exposed to fast neutron irradiation would not accumulate 1.0×10^{20}, fast $neutrons/cm^2$ until the year 2074 AD, i.e., not for another 90 years, based on initial criticality of the MUTR reactor in 1974.

Ref. 1. Etherington, H. (ED), Nuclear Engineering Handbook, First Edition, McGraw-Hill, New York, NY, 1958.

2. Murray, R.L., Introduction to Nuclear Engineering, Second Edition, Prentice-Hall, Englewood Cliffs, NJ, 1961.

Q-9.

1. Describe the primary heat exchanger currently used in the MUTR.

2. Explain how the primary side of this heat exchanger is maintained at a lower pressure than the secondary side, to preclude possible leakage of primary coolant to the unrestricted environment.

3. Discuss the potential consequences if outleakage were to occur.

Ans-9.

9.1 The system currently in use for cooling and purification of pool water has a heat-removal capacity of 2280 BTU/min. There are two heat exchangers, one a tube and shell type, the other is a "U" type.

9.2 The primary water is circulated by a pump with a capacity of 120 gal/min. Throttling of this flow produces a line pressure less than 70 psi on the primary side. The city water pressure (secondary) is on the order of 80 psi. This tends to preclude leakage of the primary water into the secondary side.

9.3 Check valves are installed in the secondary water lines to prevent primary water from flowing into the city water supply should the city water pressure fail. The radioactivity in the primary (pool) water has always been less than the maximum value given in 10CRF.55 for release. Thus primary water leakage would have no impact if leakage were to occur, since the output of the pool (suction of the pump) is less than 2 feet from the top of the pool, the exit would self-terminate when approximately 700 gallons had leaked. This assumes no operator intervention.

Q-10.

How many demineralizer units are in the primary coolant system? Specify type ("disposable cartridge" or "rechargable"), resin volume, and normal exchange or recharge frequency. Provide data on radioactivity retained by the demineralizers, based on operational experience.

Ans-10.

There is one demineralizer, having a disposable cartridge, in the purification system. The cartridge is filled with approximately 1 ft^3 of a mixed ion exchange resin provided by the Continental Water Corporation. The throughput capacity is about 10 ft^3/min. The resin is replaced approximately once every two years. There is no direct data to show any accumulation of radioactivity on the resin. Externally the resin is monitored for radiation and has never exceeded a level greater than 3 to 5 mr/hr when operating at full power (250kW). The resin is replaced when its deionizing ability is degraded and not because of radioactive material content.

Q-11.

Under what conditions do you normally use the heat exchanger system? What is the usual flow rate in the primary loop? What is the flow rate through the purification system?

Ans-11.

The heat exchanger is normally turned on when there is a possibility that the pool water temperature will approach 120°F. This is the upper temperature recommended for use of the ion exchange resin in the demineralizer cartridge. The flow rate in the primary loop through the heat exchanger is 120 gpm. Ten gallons of H_2O per minute is by-passed through the demineralizer cartridge.

Q-12.

Under what conditions is the ^{16}N diffuser system routinely used? Discuss the reasons for this use of the diffuser. Provide operational data showing the effect on exposure rate of diffuser operation.

Ans-12.

There are no operational situations under our technical specification or FSAR that would require the use of the diffuser system. However, for training purposes we usually activate the diffuser at powers in excess of 50kW. Data taken for laboratory illustration show that the radiation level at the surface of the pool (at 250kW directly above the core) is reduced from 150 mr/hr to

~50 mr/hr by the use of the diffuser. Bridge level radiation level is reduced from ~5 to 2 mr/hr.

Q-13.

Provide an up-to-date drawing of the secondary coolant system.

(Replacement for FSAR Figs. 5-1 & 5-2)

Ans-13.

The changes in the secondary coolant system are shown in the attached drawings and the FSAR has been changed accordingly. The primary system remains the same as shown in the original FSAR.

Q-14.

1. Describe the heating and air conditioning systems at the MUTR.
2. What prevents the air conditioners on the west wall of the reactor building from becoming air outlets? When the dampers in the ventilation system are "closed", there might still be some air leakage out of the room.
3. Discuss this in connection with your Fuel Element Cladding Failure analysis.

Ans-14.

1. The heating system to the reactor building is provided by means of steam from the University central heating plant. By means of heat exchangers, the steam is converted to hot water, this is circulated to heat the reactor building.

LEGEND FOR FIGURES 5-1 and 5-2.

	MEASURED VARIABLE (First letter)	INSTRUMENT FUNCTION (Second letter)
Local Instrument	Temperature T	Indicator I
Control Room Instrument	Pressure P	Element E
Gate (open) (closed)	Flow F	Transmitter T
Butterfly (open) (closed)	Conductivity C	
Particulate Filter		

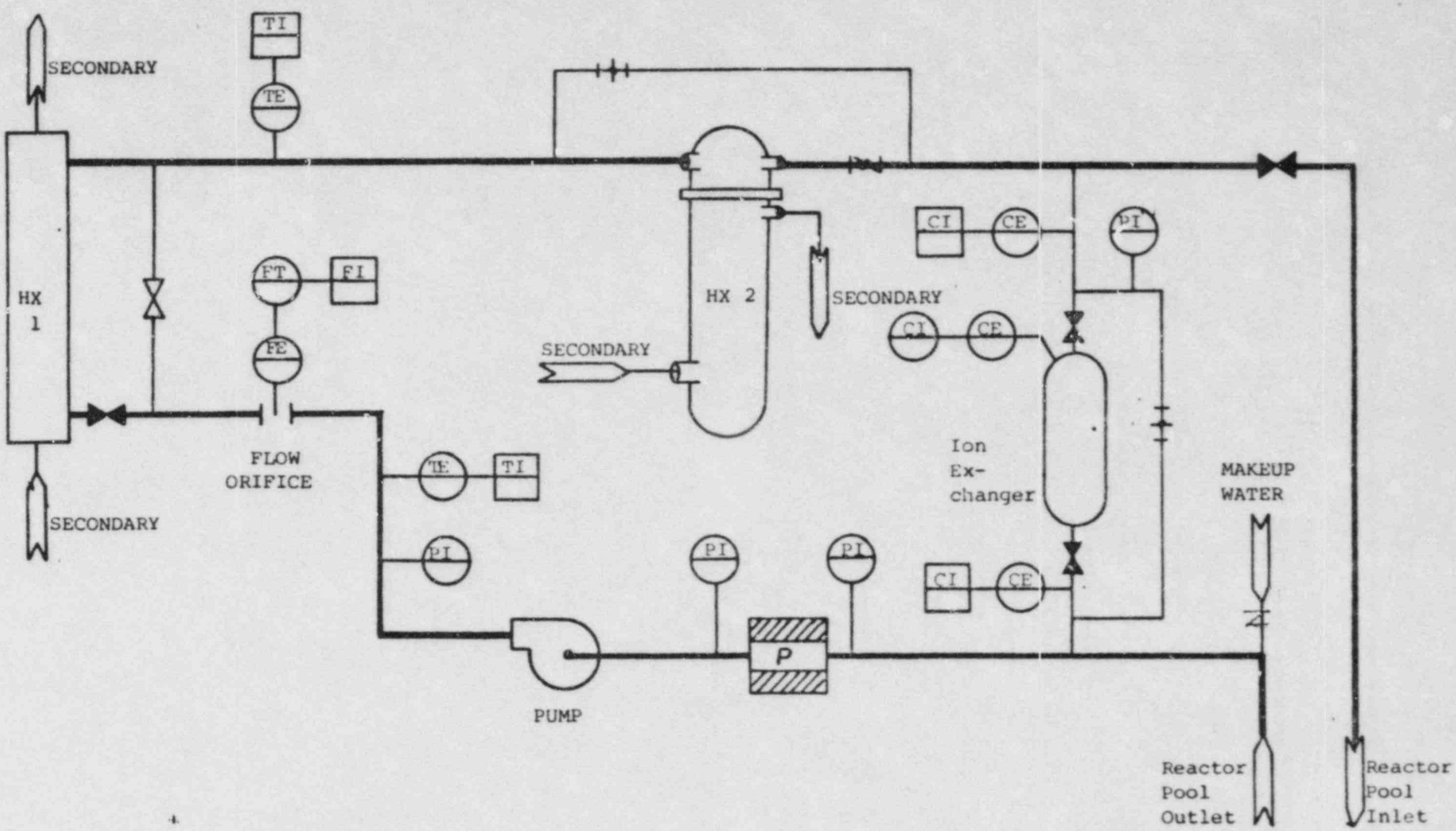


Figure 5-1 PRIMARY COOLANT SYSTEM

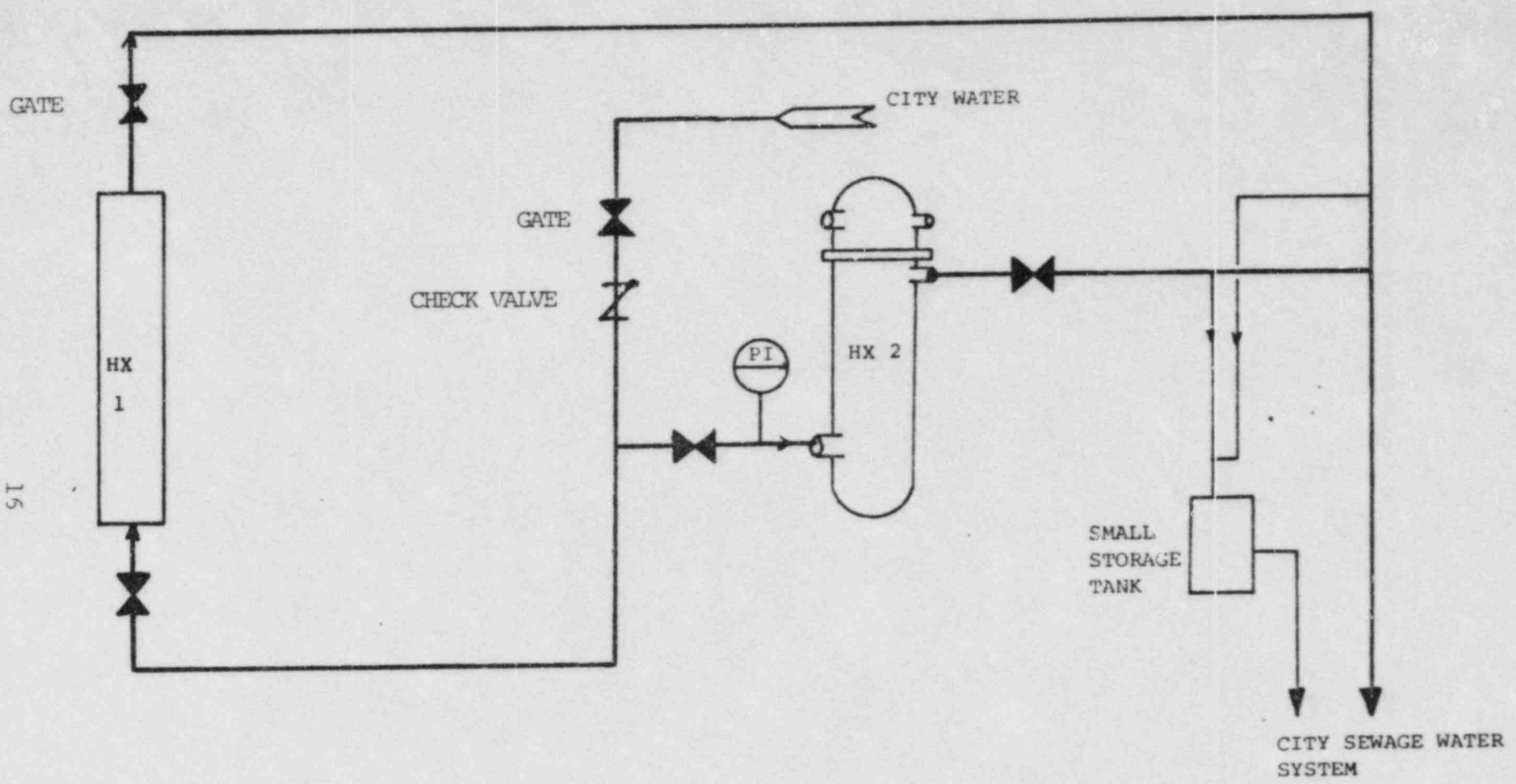


Figure 5-2 SECONDARY COOLANT SYSTEM

The air conditioning in the reactor building is provided by a closed system wherein reactor air is circulated over coils cooled by chilled water from the Chemical Engineering central air conditioning unit.

2. The two west balcony wall air conditioners have no means of venting air to the outside. The vents are mechanically closed and permanently sealed.
3. If the A/C vents were to be opened and the louvers to the west balcony were inoperative, the air flow out of the air conditioners would still be less than that from the reactor building main exhaust. This would present no additional hazard and is still bounded by the analysis presented in the FSAR.

Q-15.

Describe the radioactive liquid waste collection system at the MUTR. Summarize sa .pling procedures and analytical techniques. What is the normal discharge rate from this waste collection system to the unrestricted environment?

Ans-15.

Radioactive liquid at the MUTR consists of a neglible quantity of "hand wash" after removing protective gloves. These washes, (two lavatories on west balcony) are collected in the sump tank. This water along with two other water sources:

(1) Overflow from the reactor tank - when refilling due to evaporation.

(2) Scrub water from normal floor clean-up

is collected in a 1400 gallon sump tank. This sump water is measured for radioactivity by the Radiation Safety Office and if it meets 10CFR20 requirements for sewage disposal it is then pumped into sewage system. Thus far we have always been below 10CFR20 limits.

Q-16.

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

Ans-16.

The MUTR, during the last 5 years, has never released any reportable or significant quantity of solid or liquid radioactive waste.

All activation samples are normally retained in a lead cave until the activity has decayed to a level below that prescribed as an "unlicensed quantity" in 10CFR20. They are disposed of after being surveyed by the Rad. Safety Dept.

Other "solid" waste consists of disposable gloves, paper towels and plastic containers. These are stored in a 55 gallon drum until filled. Approximately two drums per year are removed by the CRSO. They estimate that over the last five years less than 5 millicuries of radioactivity have been collected.

Approximately 1000 gallons/month of sump water is pumped into the city sewage disposal system. The measured radioactivity in this water has never exceeded 3.6×10^{-7} $\frac{\mu Ci}{m\ell}$ of γ activity or 8.9×10^{-7} $\frac{\mu Ci}{m\ell}$ of β activity.

Q-17.

Identify the assumptions used in estimating the average annual release of ^{41}Ar from the MUTR to be on the order of 100mCi. What operational data do you have for verification? What is your best estimate of annual release if you operated on a maximum schedule consistent with your license limits?

Ans-17.

To be submitted by December 22, 1983

Q-18.

What calibration procedures are used for the radiation monitors in the reactor room? What is the function and purpose of the monitor near the exhaust vent at the "stack"? At what exposure rate and room concentration of airborne radioactivity does it initiate alarms and/or protection action? How does this compare with 10 CFR 20?

Ans-18.

To be submitted by December 22, 1983.

Q-19.

Describe the current arrangement of the various components of the thermal column and its shielding.

Ans-19.

The thermal column as described in Section 6.2 and shown in Fig. 6-2 of the FSAR represents the current arrangement and shielding in the MUTR.

Q-20.

Describe the current fire protection program at the MUTR. Who is responsible for ensuring adequate implementation?

Ans-20.

Fire Protection Program

The Fire Protection Program at the University of Maryland, College Park Campus, involves the practice of fire protection, fire prevention, and fire safety education.

The following standards, guidelines and recommended practices are utilized by the fire protection staff.

Title 29 CRF (U.S. Department of Labor) Occupational Safety and Health Standards
State of Maryland Fire Prevention Code
The National Fire Codes
The BOCA Building Code (1981)

Fire Protection Bureau

The primary responsibility for the campus fire protection program has been assigned by the Chancellor to the Department of Environmental Safety, Fire Protection Bureau, which is part of the Administrative Affairs Division. At present, the Bureau consists of three full-time employees, two of them have degrees in fire protection or safety; two part-time employees, who are fire protection students; and is assisted as needed by the Assistant Director who is a fire protection engineer. All Bureau employees have fire service experience, in addition to academic or technical training.

The fire protection service is provided by the Prince George's County Fire Department. The station that services the reactor is the College Park Volunteer Fire Dept., Inc.

Q-21.

Identify the assumptions used when estimating the releases of radioactive aerosols and/or gases from experiments. What would be the resulting concentrations within the reactor building from the failure of a maximum "fueled experiment?"

Ans-21.

The assumption made is that all gases produced are released into the building and that approximately 5% of the aerosols would be released through our absolute filter.

A maximum fueled experiment as defined by Technical Specifications is such that no more than 5 millicuries of I-131 thru I-135 are produced. The free air volume of the reactor as given on p. 3-4 of the 1980 FSAR is $1.7 \times 10^9 cm^3$. Of the iodines, I-131 is most hazardous and will be considered here. With this assumption the resulting concentration is $3 \times 10^{-6} \mu C/cm^3$. Such an incident would require shutting down the reactor and the initiation of the Emergency Plan.

Q-22.

Your analysis of the safety aspects is minimal. Section 10CFR50.59 provides for licensee actions under certain conditions. Discuss how your safety committee determines if a proposed action or experiment constitutes "an unreviewed safety question."

Ans-22.

The proposed experiment is presented to the Reactor Safety Committee for evaluation. The experiment will be evaluated according to parts e, f of 10.9 of the FSAR. Next, any potential safety problems that are revealed, will be assigned to that member of the RSC best qualified to handle it. He will then report his results to the RSC. The RSC can now accept, reject or recommend further investigation. The items of principal concern are reactivity additions in the context of potential excursions, chemical reactivity and potential radioactive material expulsion.

Q-23.

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

Ans-23.

The administrative organization of the MUTR radiation protection program is as follows:

Reactor Safety Committee (RSC)

The Committee whose members are appointed by the Chairman, Dept. of Chemical and Nuclear Engineering, establishes policies concerning radiation protection including the ALARA policy. The RSC reviews operating procedures and experiments for reactor and radiological safety prior to approval.

The RSC conducts audits to determine if facility operations are being conducted in accordance with policies and procedures.

Director, Nuclear Reactor

The Director is responsible for radiological safety at the reactor facility. He maintains the policies and procedures reviewed by the RSC and ensures that facility operations involving radiological safety are in accordance with policies and procedures. The director is a member of the RSC.

Licensed Operators

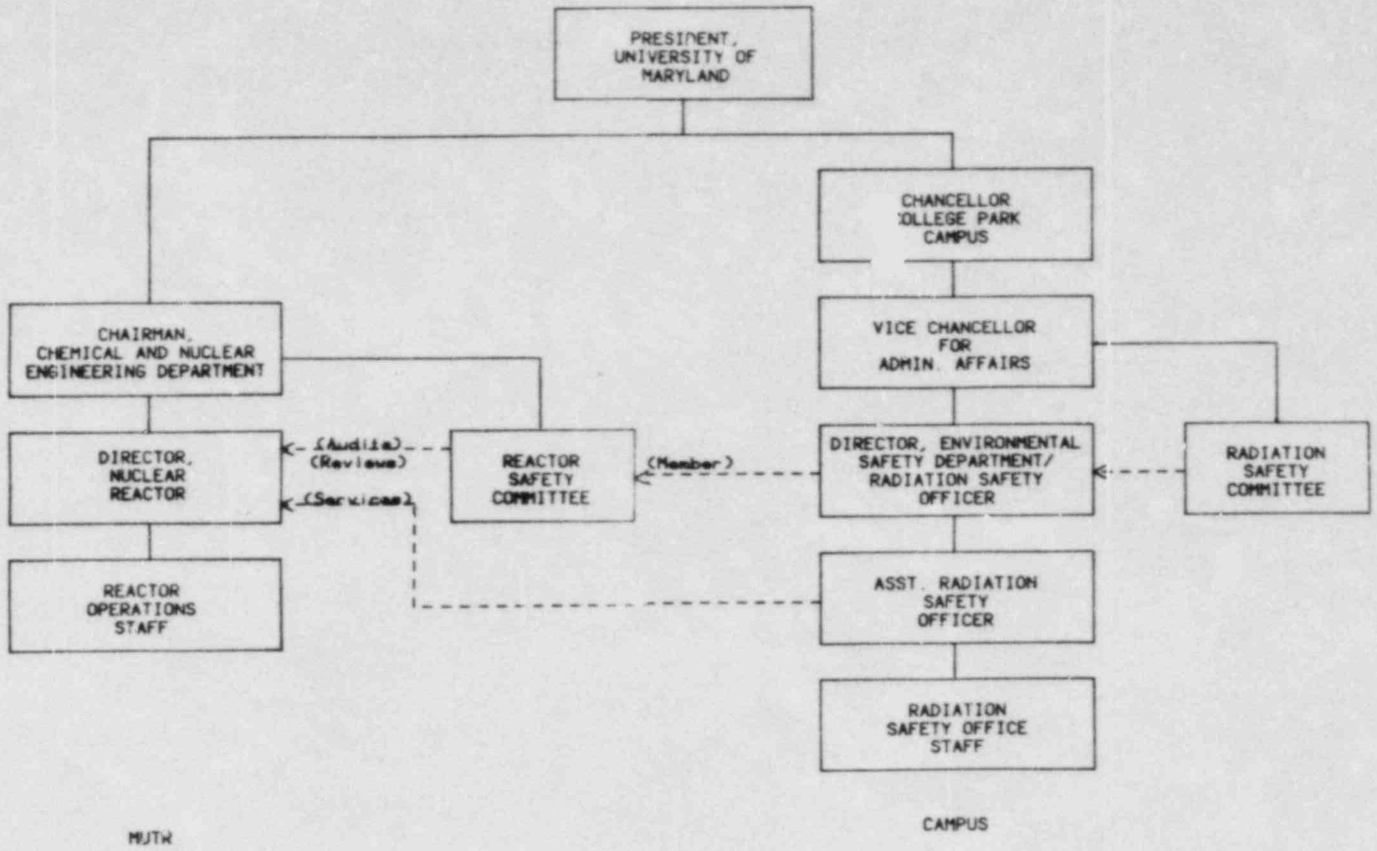
The operators are responsible for assuring that reactor operation is conducted in accordance with procedures and policies. Routine health physics activities at the facility are performed by the operations staff.

Campus Radiation Safety Officer (RSO)

The RSO reports to an organization completely independent of the MUTR organization. (See Figure 23-1) The RSO is a member of the Reactor Safety Committee.

Campus Radiation Safety Office

The Radiation Safety Office provides the facility general services such as personnel dosimetry, maintains exposure records and histories, routine radiation detector calibrations, routine radiation and contamination surveys, and radioactive waste removal. The Office is able to provide the facility with additional support as needed.



ADMINISTRATIVE ORGANIZATION
RADIATION PROTECTION PROGRAM

Fig. 23-1

Q-24.

Section 10.5.2C of your FSAR (p. 10-5) provides the duties and prerogatives of a Senior Reactor Operator. Make sure that this section is consistent with other documents, for example your Emergency Plan.

Ans-24.

We see no inconsistencies between the duties of an SRO as given in 10.5.2C of the FSAR and the original Emergency Plan. We assume that you refer to section 3.1.1 of the Emergency Plan, where the SRO (on duty) becomes the Acting Emergency Director in the event of an emergency. Since, as seen in section 3.1.4, the SRO functions as Emergency Director only until relieved by the Reactor Director or Nuclear Engineering Program Director.

Q-25.

Outline the minimum qualification (training or previous experience) for each of your Health Physics-related positions.

Ans-25.

1. All require BS/BA with 30 credits minimum in biological or physical sciences, mathematics, or health physics.
2. HP1 - no experience

   HP2 - 2 years

   HP3 - 3 years

   PH HP - 6 years

Q-26.

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

Ans-26.

The reactor radiation safety protection is provided as a service by the Campus Radiation Safety Office. Wipes of desk tops and working areas are taken on a monthly basis. If the wipes are found to contain 100dpm/100cm^2 (^{3}H-1000dpm) or greater we are notified and the area is then cleaned and resurveyed.

Air samples are taken on a monthly basis by means of a portable air sampler. Radioactivity in the air has never exceeded the limits set by 10CFR part 20.

Film badges are monitored on a monthly basis, notification level is 100mr. Exposures are reviewed each month for excessive dose.

Pocket dosimeters are issued by the Reactor Personnel to non-badged visitors and/or students. Dosimeters are calibrated by the CRSO.

Q-27.

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). At what level of the University Administration is the ALARA policy established? Provide a confirmation.

Ans-27.

Experimental facilities are utilized in accordance with procedures reviewed and approved by the Reactor Safety Committee. In addition, all (nonroutine) experiments must be reviewed and approved by the Committee for reactor and radiological safety. The Committee will recommend special precautions and instructions to maintain personnel exposures as low as reasonably achievable. The licensed operator on duty is responsible for ensuring that experiments and normal reactor operations are conducted in accordance with approved procedures and instructions. He will provide additional measures which will lower personnel exposure during facility operations. To close the loop, the Committee is required by Technical Specification to conduct audits on the radiological protection program to ensure that facility activities are being conducted in accordance with approved procedures and instructions and in a manner in which personnel radiation exposure and releases of radioactive materials including contamination are maintained at a level that is as low as reasonably achievable.

The ALARA policy which is the same as the facility radiation protection policy is established by the Reactor Safety Committee.

References: Technical Specifications

Operating Procedure 105 - Installation of Experiment

RSC Charter

Q-28.

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.

Ans-28.

Generic type	Quantity	Operable range	Calibration Method	Calibration Frequency
Geiger-Mueller	8	1) 0-2 R/hr	Cs-137, Electrical Pulse	3 months
		2) 0-2 R/hr	Cs-137, Electrical Pulse	3 months
		3) 0-2 R/hr	Cs-137, Electrical Pulse	3 months
		4) 0-500 MRem/hr	^{60}Co	3 months
		5) 0-500 MRem/hr	^{60}Co	3 months
		6) 0-500 MRem/hr	^{60}Co	3 months
		7) 0-1M cpm	Cs-137, Electrical Pulse	3 months
		8) 0-500k cpm	Cs-137, Electrical Pulse	3 months
Neutron detectors (BF_3)	2	1) 0-500k cpm	PuBe, electrical Pulse	3 months
		2) 0-500k cpm	PuBe, Electrical Pulse	3 months

Q-29.

Describe your personnel monitoring program.

Ans-29.

All personnel working within the reactor area wear film badges. These are exchanged on a monthly basis and all records are maintained by the CRSO. Badges are processed by a commercial service.

Q-30.

Describe calibration procedures for the in-house portions of the personnel monitoring program. Describe any Quality Assurance studies for the commercially supplied portions.

Ans-30.

Pocket dosimeters are calibrated using a Cs-137 source of known strength. The calibration is performed by the Campus Radiation Safety Office. A brief description of the procedure follows:

Each instrument is assigned an identification number and a history of its performance kept.

a. Instruments which read out in mR/hr and are used to detect gamma emissions are calibrated using the ^{137}Cs Calibration Range. Use of the Range is determined by referring to the calibration chart of the ^{137}Cs Range. For instance, to calibrate an instrument to read 10 mR/hr, the detector is placed 32 inches from the source, and an X800 attenuator placed in front of the exposure port. The instrument is turned on, and the source exposed. The instrument is checked at 20, 50 and 80% of the meter scale for each setting of the Function/Multiplier selector. The instrument is then adjusted to give a uniform reading throughout each scale, with the 50% mark being the most accurate. All settings and readings are recorded.

b. Instruments that read out in both mR/hr and CPM, and are used primarily to detect beta particles, will be calibrated for gammas as outlined above (a). The instrument is then checked for its efficiency using the appropriate beta emitter. No changes to the internal gamma ray calibration is done. All settings and readings are recorded.

c. Instruments that read out only in CPM and are used to detect either beta particles or gamma rays, are electronically calibrated for CPM. A pulser is connected to the instruemnt and pulses corresponding to 20, 50, and 80% of meter scale and for each setting of the Function/Multiplier Switch is introduced. The instrument is then adjusted to give a uniform reading throughout each scale, with the 50% mark being the most accurate. The instrument is then checked for its efficiency using the appropriate beta or gamma emitter. No changes to the CPM calibration is made. All settings and readings are recorded.

d. Instruments that are designed to detect alpha emissions are calibrated using a Pu source set. Usually, only one meter reading per setting of the Function/Multiplier switch is obtainable with this source set. The instrument is adjusted to be as accurate as possible to this reading. All settings and readings are recorded.

e. At the present time, all neutron instruments are calibrated by an outside agency. [NBS]

f. After all calibration procedures are completed, a calibration sticker is affixed to the instrument. All instruments when received are assigned a log number and then calibrated according to the procedure outlined above. A record of the instrument performance is maintained by the CRSO.

Q-31.

Provide a summary of the reactor facility's annual personnel exposures [the number of persons receiving a total annual exposure within the designated exposure ranges, similar to the report described in 10CFR20.407(b)] for the last 5 years of operation.

Ans-31.

1978

Number of individuals monitored		11
$\leq$ 50 mrem	9	
>50 mrem <100 mrem	2	

1979

Number of individuals monitored		14
$\leq$50 mrem	13	
>50 <100 mrem	1	

1980

Number of individuals monitored		13
$\leq$ 50 mrem	13	
>50 $\leq$100 mrem	0	

1981

Number of individuals monitored		22
$\leq$50 mrem	22	
>50 <100 mrem	0	

1982

Number of individuals monitored		35
<50 mrem	32	
>50 $\leq$100 mrem	3	

Q-32.

Provide more details on the ramp reactivity insertions discussed on Pages 4-12 and 11-3 of your FSAR. Explain the different rates of insertion of reactivity. How much total reactivity was inserted before the scram signal? What reactor period was obtained? What happened to the period scram? How much total energy was generated, and how were fuel temperatures determined?

Ans-32.

On page 11-3, section 11.2.1.2 of the reactivity ramp rate for the simultaneous withdrawal of both shim rods and the regulating rod is given by assuming a withdrawal rate of 0.32 inches/second and a total reactivity of \$8.22. The rods are 15 inches long and thus are worth a total (simultaneous withdrawal) of \$0.55/inch. Thus the reactivity insertion rate is equal to \$0.175/second. Such an incident is extremely unlikely as it would involve the simultaneous failure of three wires - passive elements - which are not stressed.

The ramp discussed on page 4-12 was an assumed worst case calculation performed by the Vendor - General Atomic. The specific calculations are no longer available, however an alternate calculation is provided here. The ramp rate of \$0.30/second is retained as this imposes no hardship on the MUTR and is very conservative. The scenario selected assumes the reactor to be delayed critical at 10 milliwatts.

Then the 30¢/sec reactivity ramp is inserted continuously until the 300 Kilowatt power level scram is actuated. The approach to this calculation is to find the power and the period from the reactor at prompt critical. The method used is that of Hetrick, pp. 93,94. There is no period scram assumed in what follows. The reciprocal period at prompt critical is given by

$$\frac{1}{T^*} = -\lambda + \left(\frac{2\gamma}{\ell}\right)^{\frac{1}{2}} \frac{\Gamma\left(\frac{\mu+2}{2}\right)}{\Gamma\left(\frac{\mu+1}{2}\right)}$$

$\lambda \quad = .0767 \text{ sec}^{-1}$

$\ell \quad = 3.9\text{x}10^{-5}\text{sec}$

$\gamma/\beta \quad = 0.30$ ¢/sec

$\beta \quad = 0.007$

$\mu \quad = \frac{\lambda\ \beta}{\gamma}$

$$\frac{1}{T^*} = -\ .0767 + (10.4)\frac{\Gamma(1.13)}{\Gamma(.63)}$$

$T^* \quad = \quad 0.15$ seconds

The power at prompt critical is given by, P* where

$$\frac{P^*}{P_0} = \frac{e^{-\mu}}{2}\left(\frac{2\beta^2}{\gamma\ell}\right)^{\frac{\mu+1}{2}} \frac{\Gamma\left(\frac{\mu+1}{2}\right)}{\Gamma\ (\mu+1)}$$

with P_0 = 10 milliwatts

$$\frac{P^*}{P_0} = (.386)(87)\ \frac{\Gamma(.63)}{\Gamma(1.26)} \quad = 52.9$$

P* = 529 milliwatts

The assumption is now made that the power continues to rise with the period, T*. The elapsed time from the power at 529 milliwatts to 300 Kilowatts will be taken as the time that the ramp continues. This is conservative as the period after prompt critical will be significantly shorter than at prompt critical. Thus assume

$$\frac{P}{P_*} = e^{\frac{t}{T^*}}$$

P = 300 Kilowatts

Then

t = 2 seconds

The added reactivity, conservatively ignoring feedback, is then

$\Delta\rho$ = 2 x \$0.30/sec = 60¢

Thus the total reactivity addition before the rods are dropped assuming 30¢/second is \$1.60.

We then proceed by assuming that the \$1.60 reactivity addition is added as a fast pulse. A personal communication with AFRRI indictes that a \$1.60 pulse in their TRIGA results in a prompt energy of 8.2 MW-sec, a peak power of 200 Megawatts and a peak temperature of 240°C. As the cores are similar in loading this may be used as a conservative estimate for the MUTR.

Typographical errors found in Table 11.1 will be corrected and an errata submitted with all other known typographical errors.

References:

D.L. Heterick, Dynamics of Nuclear Reactors, Univ. of Chicago Press, 1971

M.T. Simnad, et al., "Fuel Elements for Pulsed TRIGA Research Reactors," Nuc. Tec., V. 28, Jan. 1976

Personal Communication, Mark Moore, AFRRI Research Reactor, 5 December 1983.

Q-33.

Provide more details for the transient pulse calculation discussed on pp. 11-4 to 11-6 of your FSAR. Justify the values used for heat capacity of fuel-moderator and prompt temperature coefficient of reactivity. Show how resultant values of prompt energy generated and fuel temperature rise were obtained. Explain the differences in some parameters between the FSAR (1980) and the 1973 FSAR on the same analysis. Which values were actually used?

Ans-33.

The analysis presented in Chapter 11 of the FSAR was intended to be the same as that in the 1973 FSAR. There are a number of typographical errors, especially in Table 11.1 that make this appear not to be the case. This will be corrected and released as an errata sheet.
The heat capacity in the FSAR (1980) is given as 1.01×10^{5} watt-sec/°C. If we make use of the data of Simnad (385 cm^3 for the fueled portion of a TRIGA element) and our loading of 93 elements, this results in 2.82 watt sec/cm^3-°C which corresponds to the heat capacity as given by Simnad for an element at 187°C.
The prompt temperature coefficient used is that given by General Atomics for use in the 1973 FSAR and is equal to $1.25 \times 10^{-4} \Delta\rho/°C$. This corresponds with the value given by Simnad of $9.5 \times 10^{-5} \Delta\rho/°C$.

Resultant values for prompt energy and fail temperature rise were generated by General Atomics. The calculation shows a prompt energy generated of 32.7 MW-sec and a Nordheim-Fuchs calculation

$$E = 10^5/\tau \qquad \tau = 2.2 \text{ milliseconds}$$

results in 45.4 MW-sec. As the Nordheim-Fuchs calculation is known to result in higher than actual peak power and energy the values submitted by General Atomics as displayed in our 1980 FSAR are acceptable.

The prompt temperature rise as reported is also consistent with this scenario and is given as

prompt fuel temperature rise (average)	266°C
prompt fuel temperature rise (peak)	486°C

Reference:

M.T. Simnad, "Fuel Elements for Pulsed TRIGA Research Reactors", Nuc. Tech., V. 38, Jan. 1976

Q-34.

On page 11-5 of your FSAR you mention 1000°C. Please give a more detailed discussion of its significance, or a more specific reference where a discussion can be found.

Ans-34.

The information presented on page 4-5 of our FSAR is the data from experiments performed by General Atomic on fuel integrity as a function of temperature and pressure.

Ref. 34. General Atomic Division of General Dynamics, Research in improved TRIGA Reactor Performance, Final Report, GA-5786, pages 17-18, 20 Oct. 1964.

Q-35.

Provide more details on the fuel element cladding failure accident described on pp. 11-7 to 11-10 of your SAR.

a. What fraction of the total inventory was assumed present in the centrally located fuel element?

b. Which specific isotopes were included in the source terms and what were their activities?

c. Show how the water activity value of 6.68×10^{-4} uCi/cm^3 was obtained.

d. Explain the method and provide the equations used for the calculation of atmospheric dilution and radiation exposure inside and outside the building.

e. What meteorological conditions, release height, building leakage rate, and dose conversion factors were assumed?

f. Where you refer to General Atomic research, give specific references.

g. Your indication of a release fraction of 1.5×10^{-5} at 600°C is not consistent with the latest reference work available to us. See, for example: Simand, et. al., "Fuel Elements for Pulsed TRIGA Research Reactors," Nucl. Tech., 28, 31-56, Jan. 1976.

h. Please consider the loss of cladding integrity of one fuel rod in air. (See Columbia University hearing on TRIGA Reactor).

Ans-35.

To be submitted by December 22, 1983

Q-36.

On page 4-4 of your FSAR you define a damaged fuel element. Please discuss in more detail how the damaged condition would be ascertained.

Ans-36.

The fuel elements are presently inspected every 15 months visually for obvious defects. If such a defect were discovered, measurements of the rod, particularly bow may be made.

Experience with TRIGA fuel suggests that at the low power and burn-up of the MUTR the most likely defect (although of low probability) would be a small breach of the cladding, this could escape visual detection but would be detectable by the routine beta-gamma analysis of the pool water. Detection of elevated fission product inventory would be an indication of cladding failure and intensive visual and physical inspection would be performed to ascertain the elements responsible.

Q-37.

On page 4-6 of your FSAR you mention routinely taking samples of pool water and analyzing them for fission products. Please give more details.

Ans-37.

Water samples are taken from the 6000 gallon reactor pool on a monthly basis or when requested by the Reactor Director. The pool water is thoroughly mixed by activating the diffuser prior to sampling. A liter sample is taken about 6' above the core. This sample is analyzed for beta activity using a Beckman LS0355 analyzer which separates the betas according to energy channels. Gross gamma content is found by means of a Beckman-310 gamma detector. The analyses are performed by the Radiation Safety Office personnel. Gross beta and gamma activity are measured. Typical results are $3.60 \times 10^{-7} \mu Ci/m\ell$ for gammas and $8.9 \times 10^{-7} \mu Ci/m\ell$ for betas.

Q-38.

There are several minor and/or typographical errors in your FSAR that we discussed with you during our review visit in September. Please take this opportunity to correct them.

Ans-38.

Thank you for pointing out these errors. Steps have been taken to make the necessary corrections.

Q-39.

As discussed with you previously, we are adopting the format of ANSI/ANS 15.1 (1982), "Standard Format for Technical Specifications for Research Reactors." Your proposed Technical Specifications submitted as part of your request for renewal of your license, by letter dated May 23, 1980, are not consistent with that format. Please use ANSI/ANS 15.1 and other examples provided to you to modify the format of your Technical Specifications. Be sure that the Bases provide a technical justification for the respective Specifications.

Ans-39.

We are presently revising our Technical Specifications to be more consistent with ANSI/ANS 15.1(1982). A revised version will be submitted in the near future.
