UNIVERSITY OF VIRGINIA
SCHOOL OF ENGINEERING AND APPLIED SCIENCE
CHARLOTTESVILLE, 22901

DEPARTMENT OF NUCLEAR ENGINEERING AND ENGINEERING PHYSICS
REACTOR FACILITY

TELEPHONE: 804-924-7136

May 18, 1982

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

Re: University of Virginia Reactor (License R-66, Docket 50-62)

Dear Sir:

Enclosed is our response to the questions identified in your letter dated March 29, 1982. These answers are submitted for your information.

In addition we have included a change to Section 3.1 the proposed technical specifications which were submitted for approval in March 1982. The previous section 3.1 should be deleted and replaced with the attached version. Approval of the revised technical specifications is requested.

Sincerely,

Bryce L. Shriver, Director
Nuclear Reactor Facility

BLS:ph

cc: Mr. Robert Carter, NRC
Reactor Safety Committee

Encl.

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UNIVERSITY OF VIRGINIA
NUCLEAR REACTOR FACILITY
ANSWERS TO NRC QUESTIONS
APRIL, 1982

1. Question: What provisions are made to ensure that shield plugs are in place in unused beam ports before the reactor is started up?

   Response: The shield plugs are locked in place when the beam ports are not in use. The keys to the plugs are controlled by authorized personnel as discussed in the Security Plan. Prior to each operation a visual check of the plugs or the doors covering the plugs is made. Periodic radiation survey's are also used to confirm that the plugs are in place in unused ports.

2. Question: Describe the methods used to control/prevent personnel overexposures to radiation streaming or scattering from open beam ports at the experimental level? Include discussion of typical measured radiation levels in the building and in the unrestricted driveway outside of the building.

   Response: When the beam port is in use, shielding is used to reduce the radiation to personnel. The area near the beam is also roped off and an electric eye installed which alarms locally and in the UVAR reactor room to detect people entering the area. In addition, an area radiation monitor installed near the beam ports will trip the reactor if the detector setpoint (2 mr/hr) is exceeded.

   The dose rates at the shield used to protect radiation workers or outside the direct beam is generally less than 5 mrem/hr. Dose rates outside the building are less than 1 mrem/hr.

3. Question: Describe the program to ensure that personnel radiation exposure and releases of radioactive material are maintained at levels that are "as low as reasonably achievable" (ALARA).

   Response: The University of Virginia has had a formal program for maintaining occupational radiation exposures as low as reasonably achievable since December 4, 1979. This program includes the Reactor Facility and is a part of NRC Byproduct Materials License No. 45-00034-26. For your reference and information we enclose as Attachment I a copy of the University of Virginia ALARA program.

4. Question: Outline the minimum qualifications (training and/or previous experience) for each of the "Health-Physics"-related positions at your facility.

   Response: The University of Virginia is a state institution and the Department of Personnel and Training, Commonwealth of Virginia has set forth specifications regarding Health Physics related positions. A radiation safety specialist is normally assigned to the Facility with additional support provided by the Radiation Safety Office as needed. The qualification standards are:

A. RADIATION SAFETY SPECIALIST

Qualification Standards

Graduation from an accredited college or university with a degree in health physics or radiological physics; or graduation from an accredited college or university with a degree in chemistry, physics, engineering, general science or closely related fields, and one year of professional experience in radiological health physics work. Graduate education in health physics, nuclear engineering, or radiological health may be substituted for the experience on an equivalent time basis. Certification by the National Registry of Radiation Protection Technologists may be substituted for the degree and experience. A year or more of military experience related to the duties of the position may be substituted for the degree on an equivalent time basis.

Thorough knowledge of the uses and applications of radioactive material and radiation producing devices; thorough knowledge of radiological health principles and procedures; good knowledge of laws, standards and regulations controlling radioactive material and radiation producing devices; good communication skills; ability to work effectively with all levels of intra and inter-agency personnel.

B. RADIATION SAFETY TECHNICIAN B

Qualification Standards

Graduation from high school or successful completion of a high school equivalency test with some courses in physics, chemistry, or other related general sciences, and five years of responsible work experience in the maintenance of electronic and radiological equipment. College education in a field related to the duties of the position may be substituted for up to three years of experience on an equivalent time basis. Certification as a Radiological Technologist or Nuclear Medicine Technologist may be substituted for up to two years of experience.

Considerable knowledge of radiology principles and practices; considerable understanding of the technical and operational characteristics of radiological equipment; considerable knowledge of the standards and regulations on radiation safety; ability to direct and supervise the activities of others; ability to communicate and deal effectively with others.

C. RADIATION SAFETY TECHNICAL A

Qualification Standards

Graduation from high school or successful completion of a high school equivalency examination with some course work in physics, chemistry or other related general sciences, and two years of ex-

perience, in the maintenance of electronic equipment or radiological instruments. Certification as a Radiological Technologist or Nuclear Medicine Technologist may be substituted for the required experience.

Some knowledge of radiological principles and the operational and technical characteristics of radiological equipment; some knowledge of radiation safety standards and regulations; ability to interpret and follow technical instructions; ability to work well with others.

5. Question: For the fixed-position radiation monitors, specify the generic type of detectors, their operable ranges, and methods and frequency of calibrations and operational checks.

Response: The area radiation monitors presently used for the UVAR are Victoreen model 845 wide range monitors. They use two ion chambers to detect radiation levels from 0.1 mr/hr to 10^7 mr/hr displaying three decades at a time on a meter face. Prior to reactor operation, the instruments response is checked with an internal CL-36 check source. The alarm set points and alarm and trip functions are also checked at this time. Once every six months power supply voltages are checked and the overall calibration of the instrument is checked with a Victoreen 848-8 field calibrator which places a 100 mci Cs-137 source in a fixed geometry with the detector. Curves supplied with the calibrator indicate what the proper readings should be for several shield positions and decay times.

The argon monitors are thin window G-M tubes whose signals are processed by analog pulse counters. The counters have alarms that are checked prior to reactor operation. The background radiation is such that the counters always read a few hundred counts per minute to indicate that the detectors and readout are working. Once every six months the electronics are calibrated against a time reference and the response of the detectors are checked with a known source.

The constant air monitor draws 20 LPM of air through a fixed filter which in turn is counted by a thin window pancake G-M tube. The flow rate of the air is checked daily when the reactor is operated. Once every six months the electronics is calibrated against a time standard and the efficiency of the detector is checked with known sources. The air flow device is also calibrated against a known standard.

6. Question: For the portable radiation monitors routinely available, specify the numbers, generic types, operable ranges, and methods and frequency of calibration.

Response: Portable monitors are available at the Reactor Facility. The primary portable monitors presently used at the Facility are 4 Keithley model 36100 ion chambers which read 20 R/hr full scale.

The following backup instruments are also available:

a) Three "Radectors" using high pressure ion chambers that read out from 0.1 mr/hr to 1000 R/hr.

b) Four "cutie pie" of various makes using air ion chambers allowing readings from 0.1 mr/hr to 1000 R/hr.

c) Two "thyac" G-M tube monitors that read out from 0.01 mr/hr to 100 mr/hr.

Other portable monitors are available through the Radiation Safety Office including the following:

	INSTRUMENT	MANUFACTURER	NUMBER
a.	GM Survey Meter	Victoreen	2
b.	Ion Chamber Survey Meter	Victoreen	1
c.	"RADECTOR III"	Victoreen	1
d.	"TATTLER"	Victoreen	2
e.	"V.I.P."	Victoreen	1
f.	GM Area Monitor with Local and Remote Readout and Alarm	Victoreen	3
g.	"Vic-Chek"	Victoreen	1
h.	"Frisker" w/GM "pancake" probe	Victoreen	1
i.	Alpha/Gamma Scintillation Counter	Eberline	1
j.	Portable Lin-Log Gas Proportional Alpha/Beta Counter	Eberline	3
k.	Exposure Dosimeter	Reactor Experiments	2
l.	Multi-Channel Analyzer with NaI (Tl) Crystal Detectors	Tracor Northern	1
m.	"Triton" Airborne Tritium Monitor	Johnson Laboratories	1
n.	Condenser R-Meter with 5 Chambers	Victoreen	1
o.	Gas Proportional Counter	Nuclear Measurements Corporation	1

The portable instruments are calibrated every three months with a commercial portable monitor calibration source using a Cs-137 source.

7. Question: Describe your personnel monitoring program, including bio-assay and in vivo counting capabilities. Discuss calibration methods and any quality assurance studies on commercially supplied services.

Response: A. Personnel Monitoring: All personnel, students and visitors are required to wear personnel monitoring devices (film badges or pocket dosimeter) at the UVAR Facility. Permanent records are kept by the Health Physicist on individual exposures. Individual exposures are kept as low as reasonably achievable in accordance with ALARA program and shall in no case exceed the limits set forth in 10 CFR Part 20. Film badges are obtained commercially from the R.S. Landauer, Jr. and Company. For the purpose of quality assurance the Health Physicist exposes three film badges each month to a known radiation field. The calibration procedure used for pocket dosimeters is given in Attachment II.

B. Bioassays:

1. Bioassays fall into two general categories:

a. Assay of biological samples

In this method, biological samples (urine, breath, blood, etc.) are taken and assayed for radioactivity. The radiation dose to the critical organ is then estimated by relating the radioactive concentration to a radiation dose using appropriate mathematical models. This is most often done for tritium but can be performed for other radioisotopes as well.

b. External gamma counts

For some gamma-emitting radioisotopes it is possible to estimate the radiation dose to the critical organ by externally measuring the radiation levels using sensitive detectors. This method is most frequently used for radioactive iodine.

2. Guidelines for bioassays performed by the Radiation Safety Office

a. Bioassay Requirements for I-125 and I-131

(i) Except as noted in paragraphs (ii) and iii), in vivo bioassays (i.e., "thyroid counts") are required at quarterly intervals if I-125 and I-131 are used in levels which exceed the amounts listed in the following table:

(ii) Bioassay must be performed on all individuals within 72 hours (but more than six hours) after a laboratory first begins to work with I-125 or I-131 and every two weeks thereafter for a three month period. At the end of the three month interval, the Radiation Safety Office will make an evaluation to determine if the bioassay frequency should be reduced to that given in Paragraph (i).

(iii) If I-125 or I-131 are used on an infrequent basis, bioassay should be performed within 72 hours but more than 6 hours after use.

(iv) Bioassays are not required if the amounts used do not exceed 1/10 of the levels listed in Table 1. The Radiation Safety Office should be consulted on the necessity of bioassays if amounts used are between the levels listed in Table 1 and 1/10 of those levels.

b. Bioassay Requirements for Tritium

Conditions Requiring Bioassay

(i) Routine Bioassay is required when quantities processed by an individual at any one time, or total amount processed per month, exceed those for the respective forms of tritium as shown in the attached Table 11.

(ii) Above 0.1 of, but less than, the levels in Table 11, the need for bioassay will be decided by the Radiation Safety Committee on a case by case basis. Except as stated in (iii) below, bioassay is not required for process quantities less than 0.1 of those in Table 11.

(iii) Special bioassay measurements will be performed to verify the effectiveness of respiratory protection devices and other protective clothing when used.

(iv) Initial Routine Samples

Within 48 hours following entry of an individual into an area where operations require bioassay and then every two weeks or more frequently thereafter as long as the individual is working with tritium.

(v) After 3 Months

The sampling frequency may be changed to quarterly if, after 3 months, the Radiation Safety Office feels that the following conditions are met:

(1) The average urinary tritium concentration from specimens obtained during the 3-month period does not exceed 3 μCi/l.

(2) If measurements taken of the concentration (Ci/ml) of tritium in air to which workers are exposed when averaged over a quarter and multiplied by the factor 6.3×10^8 ml, does not exceed 0.8 mCi.

Table 1

ACTIVITY LEVELS ABOVE WHICH BIOASSAY

FOR I-125 OR I-131 IS NECESSARY

Types of Operation	Activity Handled in Unsealed Form Making Bioassay Necessary*	
	Volatile or Dispersible*	Bound to Nonvolatile Agent*
Processes in open room or bench, with possible escape of iodine from process vessels.	1 mCi	10 mCi
Processes with possible escape of iodine carried out within a fume hood of adequate design, face velocity, and performance reliability.	10 mCi	100 mCi
Processes carried out within gloveboxes, ordinarily closed, but with possible release of iodine from process and occasional exposure to contaminated box and box leakage	100 mCi	1000 mCi

*Quantities may be considered the cumulative amount in process handled by a worker during a 3-month period; e.g., the total quantity introduced into a chemical or physical process over a 3-month period, or on one or more occasions in that period, by opening stock reagent containers from which radioactive iodine may escape. Quantities in the right-hand column may be used when it can be shown that activity in process is always chemically bound and processed in such a manner that I-125 or I-131 will remain in nonvolatile form and diluted to concentrations less than 0.1 mCi/mg of nonvolatile agent. Capsules (such as gelatin capsules given to patients for diagnostic tests) may be considered to contain the radioiodine in nonfree form, and bioassay would not be necessary unless a capsule were inadvertently opened (e.g., dropped and crushed). However, certain compounds where radioiodine is normally bound are known to release radioiodine when the material is in process, and the left-hand column may then be applicable. In those laboratories working only with I-125 in radioimmunassay (RIA) kits, the quantities of I-125 are very small and in less volatile forms; thus, bioassay requirements may be judged from the right-hand column. In field operations, where reagent containers are opened outdoors for simple operations such as pouring liquid solutions, the above table does not apply; bioassay should be performed whenever an individual employee handles in open form (e.g., an open bottle or container) more than 50 mCi at any one time.

Operations involving the routine use of I-125 or I-131 in an open room or bench should be discouraged. Whenever practicable, sealed bottles or containers holding more than 0.1 mCi or I-125 or I-131 should be open at least initially within hoods having adequate face velocities of 0.5 m/sec or more.

(3) The working conditions during the 3-month period, with respect to the potential for tritium exposure are representative of working conditions during the period in which a quarterly urinalysis frequency is employed, and there is no reasonable expectation that the criteria given in (1) and (2) above will be exceeded.

C. Special Bioassays:

1. In the event of a reported spill or other incident, bioassays will be performed as appropriate on personnel who are suspected of having ingested radioactive material. The Radiation Safety Officer will make the determination of the need for bioassays in accident situation.

2. Special tests for determining the presence of internal emitters in the body are desirable for persons handling intermediate or high-level quantities of unconfined radioactive materials. These tests may be ordered at the direction of the Radiation Safety Officer.

TABLE 11

ACTIVITY LEVELS OR CONCENTRATIONS ABOVE WHICH BIOASSAY SHALL BE REQUIRED*

TYPES OF OPERATION	HTO FORM (& Forms other than those on right-hand cols.)	HT OR T_2 GAS IN SEALED PROCESS VESSELS	NUCLEOTIDE PRECURSORS	HTO MIXED WITH MORE THAN 10 Kg OF INERT H_2O OR OTHER SUBSTANCE
PROCESSES IN OPEN ROOM OR BENCH, WITH POSSIBLE ESCAPE OF TRITIUM FROM PROCESS VESSELS	0.1 Ci	100 Ci	0.01 Ci	0.01 Ci/Kg
PROCESSES WITH POSSIBLE ESCAPE OF TRITIUM, CARRIED OUT WITHIN A FUME HOOD OF ADEQUATE DESIGN, FACE VELOCITY, AND PERFORMANCE RELIABILITY.	1 Ci	1000 Ci	0.1 Ci	0.1 Ci/Kg
PROCESSES CARRIED OUT WITHIN GLOVEBOXES, ORDINARILY CLOSED, BUT WITH POSSIBLE RELEASE OF TRITIUM FROM PROCESS AND OCCASIONAL EXPOSURE TO CONTAMINATED BOX AND BOX LEAKAGE.	10 Ci	10,000 Ci	1 Ci	1 Ci/Kg

*Quantities present (10 Kg) may be considered either the amount processed by an individual at any one time (when accidental intake is more likely), or the amount of activity entered into process (throughout) during any one month when routine handling of repeated batches is the more likely source of exposure). Concentrations in the right-hand column may be used when activity in process is always diluted in more than 10 Kg of other reagents, as in nuclear reactor coolant systems.

8. Question: List all parameters that are alarmed in the control room, and and the alarm trip settings.

Response: The parameters that have both audio and visual alarms in the control room are:

	Channel	Normal Setpoint*
a)	Bridge Radiation Monitor (Scram)	30 mr/hr
b)	Reactor Face Monitor (Scram)	2 mr/hr
c)	Hot Cell Monitor	2 mr/hr
d)	Demineralizer Room Monitor	300 mr/hr
e)	Room Argon Monitor	2000 cpm
f)	Duct Argon Monitor	700 cpm
g)	Core ΔT	<13.15°F (2.03 Mw)
h)	Core Gamma Monitor	110% Full Power
i)	CAVALIER Room-Fuel Storage Area	5 mr/hr
j)	Demineraliver Conductivity	2 μmho/cm
k)	Constant Air Monitor	Variable Setting to Show Increase Above Current Levels
l)	Power Range (Scram)	125% Full Power
m)	Intermediate Range (Period Scram)	3.5 Seconds
n)	Low Flow (Scram)	900 gpm
o)	Low Pool Level #1 (Scram)	19'3" Above Core
p)	Low Pool Level #2 (Scram)	19'3" Above Core
q)	Demineralizer Room Door (Open)	Status
r)	Heat Exchanger Room Door (Open)	Status
s)	Secondary Pump (Off)	Status
t)	Primary Pump On With Heater Down	Status
u)	Primary Pump Off (Scram)	Status
v)	Loss of Servo Rod Control	Status
w)	Auxiliary Scram Bus Trip	Status
x)	Console Manual Scram Trip	Status
y)	Pool Temperature (Scram)	105°F

*These may be changed depending on operational conditions, however the technical specification limits will not be violated.

9. Question: How does the operator determine the cause of each alarm or scram?

Response: The source of all alarms and several scrams are indicated by lights that must be manually reset to be extinguished. These are listed in response to question #8. The source of several other scrams can be determined by the status of the actuating device following a trip. These include the truck door, escape hatch, pump on, range switch, key switch, and evacuation trips. This leaves three scrams with no status indication. They are the manual scram switch in the reactor room, manual scram switch at the ground floor level, and air to header scrams. These require manual initiation so the source is evident.

The actual cause of each scram is investigated and corrected by a Senior Operator prior to resuming operations.

10. Question: Your thermal/hydraulic analysis for the core apply to either all flat-plate or all curved plate fuels. If you expect to use a mixture of these two types of fuel in one core loading, further analysis is required. Please provide such analysis, or conform that mixtures of these fuel types will not be used.

Response: Mixed cores consisting of flat plate and curved plate elements will not be used.

11. Question: In a 4x4 graphite-reflected core composed of all curved-plate fuel elements, (1) is the fraction of the flow that is diverted to the reflector still 10.9%? (2) is 48 gpm/element still a valid flow rate when that for the total core is 940 gpm? and (3) is the minimum channel flow rate still a factor of 0.835, or 16.5% less than the average channel flow rate? If not, how do these factors and the power-vs-flow curves change?

Response: (1) The original calculation of the fraction of the flow diverted to the reflector proceeds as follows (for flat plate fuel).

Two values are measured for a 4x4 graphite reflected core

940 gpm total primary flow
48 gpm flow through channels of element (i.e., flow per element)

The elements were designed such that the flow outside the outside fuel plates is the same as the flow in the channels. The control rod elements were also designed to have the same flow as the fully loaded elements. Therefore, to get the total core flow, the flow per element is multiplied by (12/11) and by the amount of fuel elements (16).

(48) (12/11) (16) = 837.8 pgm

The bypass flow through the reflector is the difference between

940 - 837.8 = 102.2 gpm or 10.9% of the total flow.

Assuming that the curved plate elements were also designed to allow equal flow on the inside and outside of the outermost fuel plates a ratio of the flow area per element can be determined. The calculated nominal flow area per channel on the curved plate elements is 0.317 in^2 while on the flat plate elements the area is 0.553 in^2. Multiplying these by the equivalent number of channels per element the flow area per element is obtained.

Curved plate: $0.317 \times 18 = 5.706\ in^2$
Flat plate: $0.553 \times 12 = 6.636\ in^2$

Thus, the total flow area through the core for the same number of elements is 5.706/6.636 or 86% of the area with flat plate elements when using curved plate elements.

The flow through the core is then reduced by 14%, from a 89.1% value to

$$\frac{(89.1)\ (.86)}{(89.1\ \ (.86) + 10.9} = 87.5\%$$

for curved plate elements and the bypass through the reflector is 12.5%.

Response: (2) Since the original calculations were performed a new primary pump has been installed such that the nominal total primary flow rate is now about 1050 gpm. For 16 element graphite reflected cores the following flows per element are obtained

Flat plate: (1050) (.891)/16 = 58.5 gpm/element total
Curved plate: (1050) (.875)/16 = 57.4 gpm/element total

If the total flow was still 940 gpm, the flow per element for the curved plate fuel would be:

(940) (.875)/16 = 51.4 gpm per element total
or 48.5 gpm through inside flow channels only

Thus, even though the flow bypassing the core is greater when using the curved plate fuel the actual flow through the flow channels of the element is slightly greater than when using the flat plate fuel due to the lesser amount of flow outside of the outer fuel plates.

Response: (3) No measurements have been made but it is surmised that the minimum channel flow rate is not as low as 0.835 of the average channel flow rate for the curved plate fuel. The fuel element handle which caused the restriction in the flat plate fuel is positioned directly over the middle fuel channel. In the curved plate fuel the handle spans portions of several channels due to the plate curvature. Therefore, the amount of blockage in each of these channels should not be as high as that of the central channel in the flat plate elements.

In summary, each of the factors in question change as follows when comparing the curved plate fuel to the flat plate fuel.

	Flat Plate	Curved Plate
Flow bypass to reflector	10.9%	12.5%
Flow per element (16 in core 940 gpm total flow)	48 gpm	48.5 gpm
Flow per element (16 in core, 1050 gpm total flow)	53.6 gpm	54.2 gpm
Minimum channel flow below the average	16.5%	<16.5%

Considering all of these adjustments to the original curved plate analysis it is apparent that the power versus flow curve will change slightly, but in the conservative (safer) direction. This is especially true when the higher flow rate provided by the new primary pump is taken into account.

12. Question: What is the maximum radial peaking factor in the water-reflected curved-plate fuel core? Is the peak power density in a graphite-reflected core still a factor of 1.12 larger than in the water-reflected core?

Response: The maximum radial peaking factor in a water-reflected, curved plate fuel core has not been directly measured or calculated since this configuration has not been used. Below is the data which is available.

	Radial peaking factor
Flat plate fuel, graphite reflected	1.37 (measured)
Flat plate fuel, water reflected	1.57 (measured)
Curved plate fuel, graphite reflected	1.45 (calculated)
Curved plate fuel, ½ water and ½ graphite reflected	1.53 (measured)

It appears from the trends in the above data that the radial peaking factor for a water reflected core of curved plate fuel is in the range of 1.6 to 1.65. Depending on the exact value within this range the peak power density in a graphite reflected core of curved plate fuel is a factor between 1.16 and 1.12 times larger than in the water-reflected core, nearly the same as the value for the flat plate fuel.

13. Question: Is the flow/channel that is given on page 62 of Amendment 1 to the SAR for natural convection cooling still valid for curved-plate fuel? If not, is the ratio of percent change in flow per channel to percent change in power per channel less than 1.303?

Response: No measurements or calculations have yet been performed concerning the natural convection cooling mode of operation with curved plate fuel. It is assumed, however, that because of the

lower power density per plate and the greater total heat transfer area the curved plate elements have higher safety limits than the flat plate fuel. This was discussed in a letter from J. L. Meem to K. R. Goller, A.E.C. Assistant Director for Operating Licenses on July 2, 1974.

Because a smaller amount of water per channel in the curved plate fuel is combined with a lower amount of heat being transferred per channel the flow rate established in natural convection cannot be simply obtained from the value for the flat plate fuel. The computer code which was used in the original flat plate fuel analysis and the forced convection, curved plate fuel analysis is not presently available.

14. Question: Update Table III-1 of the SAR to include the metal-to-water ratio for curved-plate fuel elements.

Response: The calculated metal-to-water ratio for curved plate fuel elements is 0.59.