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DEPARTMENT OF NUCLEAR ENGINEERING AND ENGINEERING PHYSICS
NUCLEAR REACTOR FACILITY
SCHOOL OF ENGINEERING AND APPLIED SCIENCE
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Telephone: 804-924-7136

December 13, 1984

Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Attn: Mr. Cecil O. Thomas

Dear Mr. Thomas:

We are pleased to respond to your letter dated November 26, 1984. At this time, we are submitting the written responses to the formal questions which arose from the additional safety evaluation reviews performed by your group, during their November visit to our facility. In answer to question #7, please find enclosed a modified Table 3-1 of the CAVALIER SAR. Table 9-1 of the SAR has also been modified in response to question #8, as more up-to-date data was used in the recalculations. Also, with regard to question #22, the U.Va. Reactor Facility staff has reviewed the CAVALIER Technical Specifications and made necessary changes, which are included in the replacement pages to be found in annex to our response to the questionnaire.

The staff answers to the questionnaire have been presented to the U.Va. Reactor Safety Committee (RSC) and were discussed at its meeting on December 12, 1984. The Reactor Safety Committee is continuing its review of this lengthy document and its approval is pending. The next RSC meeting will take place on December 19, 1984. Formal approval of this document is expected on that date. You will be formally notified of the Committee's decision shortly thereafter. Unfortunately, the short 20 day response time and the many (24) questions did not permit formal RSC approval at this time.

However, the Reactor Safety Committee was able to review and discuss the revised CAVALIER Technical Specifications. The new definitions for Safety Limit and Limiting Safety System Setting were approved.

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It is our hope that the information provided will permit your staff to continue and complete its safety evaluation. If you have any further questions, please have the Project Manager for our facility contact Pres Farrar, Reactor Supervisor, at (804)924-7136.

Sincerely,

Robert U. Mulder

Robert U. Mulder, Director
UVA Reactor Facility

Sworn to and subscribed before me this 13th
day of December, 1984
Witness my hand and official seal.
Delores E. Van Notary Public
My Commission Expires October 14, 1985

U.VA. Formal Questions

1. Question: When the CAVALIER is again loaded with fuel, describe the expected core configuration (number and location of fuel elements, control rods, and graphite or aluminum reflector elements). What is the total expected U-235 inventory in the CAVALIER core?

Answer

The expected core configuration for the CAVALIER reactor is a 4x4 array of fuel elements with the control rods at the four corners, water reflected, with an aluminum wire mesh around the core to prevent any movement of objects near the fuel elements (see attached diagram). This is the same configuration that has been used in the CAVALIER for the past 9 years except the previous loading had flat plate fuel elements (approximately 2.2 Kg. U-235) and the planned loading will have curved plate fuel elements (approximately 2.7Kg. U-235). An experimental element with removable fuel plates may be used to adjust the reactivity of the core. The final configuration will be such that it is well within the Technical Specification limits concerning shutdown margin and excess reactivity. The expected worth of each of the control rods is approximately 1.0% $\Delta k/k$, excess reactivity approximately 1.0% $\Delta k/k$ and the shutdown margin with the highest worth rod withdrawn approximately 1.8%.

CAVALIER REACTOR LOADING CHART

Loading No. ______________ Date Approximately December 1984

Total Mass 2726 gram U-235

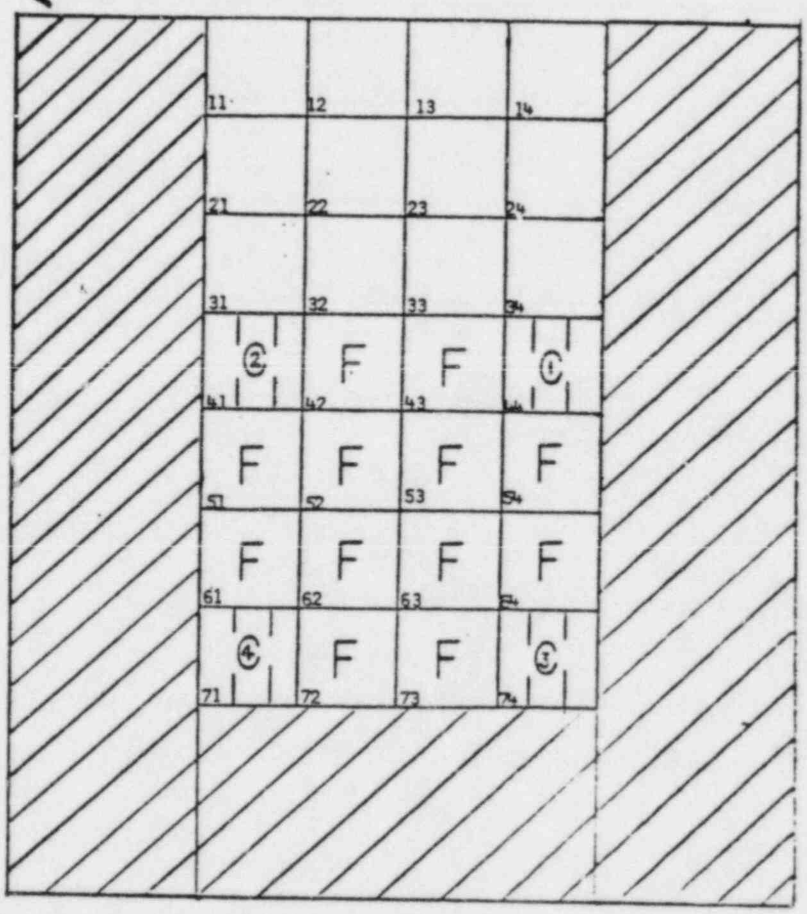
F - Normal Fuel Element
P - Partial Fuel Element
C - Control Rod Fuel Element
E - Removable Plate Fuel Element

Excess Reactivity ~1.0% Δk/k

Shutdown Margin ~1.8% Δk/k
One Rod Withdrawn

Anticipated Loading
Rods Worth ~1.0% Δk/k Each

Al Wire Mesh



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2. Question: Confirm that the data in Table 3-1 of the CAVALIER SAR are correct. Submit a modified Table 3-1 if necessary.

Answer:

A modified Table 3-1 is attached.

Table 3-1
REACTOR DATA

Data is given for a water reflected core only since there are no plans to operate a graphite reflected core in the CAVALIER at this time. In the first ten years that the reactor was operated no graphite was used but this should not preclude the possibility of using a graphite reflector in the future.

Active Core Dimensions	
Length (in.)	23.5
Width (in.)	12.1
Depth (in.)	13.1
Active Core Volume	
Liters	61.04
Cubic Inches	3.75×10^3
Number of Standard Elements	12
Number of Control Rod Elements	4
Mass U^{235} (Kg)	2.73
Mass Aluminum (Kg)	42.84
Mass H_2O at 100^oF (Kg)	37.80
Metal to Water Ratio (vol Al)/vol H_2O)	0.50
Atomic Ratios in Active Core	
Atoms U^{235}	1
Atoms Aluminum	137
Molecules H_2O	181
Average Thermal Flux @ 10 W	$\sim 1.0 \times 10^8$

<u>Fuel Elements</u> - $U\text{-}Al_x$ clad with aluminum

Overall dimensions	
Length (in.)	34.375
Width (in.)	2.94
Depth (in.)	3.248
Standard element	
Number of plates	18
Width (in.)	2.775
Thickness (in.)	0.050
Length (in.)	
Inner plates	24.625
Outer plates	28.625

TABLE 3-1

(Continued)

Fuel thickness ($U-Al_x$) (in.)	0.020
Fuel width (max. variance) (in.)	2.22 - 2.40
Fuel length (max. variance) (in.)	22.5 - 24.0
Uranium weight % in $U-Al_x$	69%
Weight U-235 per element (g)	194.94 ± 1.36 (95%)
Water space between plates (in.)	0.126
Side plates - overall	
Length (in.)	28.625
Width (in.)	0.188
Depth (in.)	3.150
Number of Grooves	18
Depth of grooves (max.)(in.)	0.091
Width of grooves (in.)	0.057

Partial Elements - Same as standard elements except every other plate is aluminum only, producing an element with one-half the U-235 content of a standard element.

Control Rod Element - Only half the number of fuel plates that are in a standard element with a central gap to allow for insertion of absorber.

Control Rods

Shim Safety Rods

Number	4
Absorber Material	Boron Steel 1.5% Boron

Dimensions, Overall

Width (Approx.)	in.	1
Depth (Approx.)	in.	2.38
Length (Approx.)	in.	27.5
Travel (Approx.)	in.	24
Weight (Dropping Section)	kg.	5.5

TABLE 3-1

(Continued)

Drive - Electric motor, 115 V 60 cycle, split phase, through reduction gear and rack and pinion drive

Drive Speed	in/min	Approx. 3.7
Release - Magnetic Clutch		
Typical Reactivity, fully inserted $\% \frac{\Delta K}{K}$		~ 1.0
Typical Reactivity per inch $\% \frac{\Delta K/in}{K}$		~.05
Typical rate of Reactivity Increase in up travel $\% \frac{\Delta k/sec}{K}$		~.003

<u>Reactivity Coefficients</u>

Temperature Coefficients	$< -1.74 \times 10^{-4}$ $\Delta k/k/^{\circ}F$
Void Coefficient	-1.90×10^{-3} $\Delta k/k/\%$ void

3. Question: Describe the method used to determine the moderator temperature coefficient and void coefficient.

Answer: Temperature Coefficient

The temperature coefficient was measured in the UVAR reactor, which uses the same type of fuel as the CAVALIER, by R.J. Palabrica[1]. He compared the plot of a power rise that resulted from a ramp reactivity insertion while at low power where there is no temperature feedback with a similar transient at high power where the temperature change with changing power is significant. The difference in slopes of the power versus time plots for the two cases is a result of the temperature coefficient. By unfolding the difference and calculating the metal temperature rise during the transient, Palabrica was able to determine the moderator temperature coefficient and the fuel temperature coefficient. The results were:

moderator temperature coefficient	$-1.47 \pm 0.09 \times 10^{-4}\ \Delta k/k/^{o}F$
fuel temperature coefficient	$-0.27 \pm 0.02 \times 10^{-4}\ \Delta k/k/^{o}F$
Total temperature coefficient	$-1.74 \pm 0.11 \times 10^{-4}\ \Delta k/k/^{o}F$

The core configuration for which the above values were measured had 21 fuel elements. The CAVALIER typically has 16 fuel elements so the moderator temperature coefficient has a larger negative value than that presented above.

Void coefficient

The void coefficient for a similar core (Ford Nuclear Reactor at the University of Michigan) was measured by J.L. Shapiro[2] by measuring the change in reactivity that resulted from bubbling air through the fuel elements. The measurements were performed using a highly enriched core in 1961. The results of the measurements are summarized below:

Condition	Void Coefficient
center of core	$-3.2 \pm 0.2 \times 10^{-3}\ \Delta k/k/\%$ void
edge of core	$-1.20 \pm 0.08 \times 10^{-3}\ \Delta k/k/\%$ void
overall	$-1.90 \pm 0.12 \times 10^{-3}\ \Delta k/k/\%$ void

References:

1) R.J. Palabrica, Determination of the temperature coefficients of reactivity of a reactor by analysis of its response to ramp inputs. Phd Thesis, University of Virginia, Charlottesville, Va. June 1968.

2) J.L. Shapiro, "The Void Coefficient in an enriched, water reactor". Nuclear Science and Engineering No. 12, pages 449-456 (1962).

4. Question: What is the maximum hypothetical accident (MHA) at the CAVALIER facility? Provide the calculated results of this accident including source terms and exposures and the assumptions and methodology used in the calculation.

Answer: The maximum hypothetical accident at the CAVALIER facility is assumed for calculational purposes to be the failure of a fueled experiment and the subsequent release of gaseous fission products into the air and ultimately to the outside of the Reactor Facility building. Several other kinds of accidents can be postulated that would have more serious consequences than a fueled experiment failure but they are so incredible in nature that they will not be considered as plausible scenarios. The analysis for such postulated accidents such as Ramp Reactivity Insertions and Large Reactivity Excursions are included in Section 9 of the CAVALIER SAR.

The following items are to be considered when calculating the consequences of the maximum hypothetical accident.

(1) Maximum fission product inventory available for release. Iodine, because of its ability to deliver a substantial internal dose from the inhalation of gaseous iodine, will be considered to be the most critical fission product.
(2) Transport mechanism from the reactor tank to the outside air including hold-up time, radioactive decay, in-air dilution and all other factors affecting transport.

To calculate the number of atoms of iodine produced by a certain number of fissions up to the equilibrium value the following equation is used.

$I = (P)(Y)(K)(\lambda)^{-1}$ where

P = Power produced by the experiment, in watts.
Y = Fractional yield of iodine isotope per fission.
K = A constant, 3.2×10^{10} fissions-sec^{-1}-watt^{-1}.
λ = The decay constant, (ln 2) (half-life)$^{-1}$; units of sec^{-1}.
I = Number of atoms of iodine produced.

Table 1 contains the necessary nuclear data and the results of the computation of the equilibrium amounts of iodine produced by the long operation of a fueled experiment at one watt.

TABLE 1

Iodine Data and Fission Product Production

Isotope	Half-Life	Yield (%)	Atoms Produced	mCi Produced
I-131	193 hr	3.1	9.94E14	26.8
I-132	2.28 hr	4.4	1.67E13	38.1
I-133	20.9 hr	6.6	2.29E14	57.1
I-134	0.88 hr	7.8	1.14E13	67.5
I-135	6.86 hr	6.1	6.96E13	52.8

TABLE 2

From the U.S. Nuclear Regulatory Commission Regulatory Guide 1.109 the following information relating the inhalation of radioactive iodine to the absorbed dose in the thyroid can be obtained. The specific information is extracted from Tables E-7 and E-10.

Inhalation Dose Factors (Rem/mCi inhaled)

Isotope	Adults	Infants
I-131	1.49E3	1.06E4
I-132	1.43E1	1.21E2
I-133	2.69E2	2.54E3
I-134	3.73E0	3.18E1
I-135	5.60E1	4.97E2

To calculate how much iodine is inhaled several assumptions must be made as to how much iodine escapes the failed experiment, migrates out of the reactor tank and then is carried by the wind to the site boundary where the radionuclide is inhaled.

For iodine isotopes, 50% of the total inventory is assumed to escape the experiment and 50% of this amount is available to be released from the reactor tank. Thus, the amount of iodine in the CAVALIER room is assumed to be one-quarter of the inventory which was present in the experiment at the time that it failed.

The concentration in the room at a time t after the release is equal to:

$$C = C_o e^{-(a+\lambda)t}$$

where C = current concentration
C_o = initial concentration
λ = ln(2)/isotope radioactive half life
a = ln(2)/length of time for half the material to escape from the CAVALIER room to outside the building.

The source term is proportional to the current concentration of gas so

$$Q = aC = aC_o e^{-(a+\lambda)t}$$

The total dose to the thyroid to a person standing on the site boundary is calculated by the following equation.

$$\text{Dose} = \int_{o}^{t} (BR)(x/Q)QK \, dt$$

where BR = breathing rate = 3.47×10^{-4} m^3/sec
X/Q = meteorological attenuation factor
Q = source term
K = rem per mCi inhaled

The calculation of the meteorological attenuation factor is the most complicated part of the overall calculation. With the assumptions that the release is made at ground level and that the person inhaling the gas is in the center of the plume the factor is equal to:

$$\frac{X}{Q} = \frac{1}{E_y E_z U\pi}$$

where U = wind speed (m/sec)
E_{y}, E_{z} = total diffusion factors (m)

The diffusion factors are a functio of the standard deviation of the distribution of material in the plume, the cross sectional area of nearby buildings and pressure wakes which exist around those buildings. Using the most conservative assumptions and the data suggested in TID-14844 "Calculations of Distance Factors for Power and Test Reactor Sites" the value of x/Q = 1.4E-4.

Combining all the above information together with the assumption that half of the material in the CAVALIER room is released every 30 minutes and that a person is standing on the site boundary for two hours following the release, the following doses to the thyroid are calculated.

Infant	5.18 mRem
Adult	0.67 mRem

If all five iodine isotopes existed at their respective MPC's for unrestricted areas at the site boundary for the entire time (2 hours) that a test subject is projected to have remained there he would receive the following doses to the thyroid.

Infant	7.8 mRem
Adult	0.95 mRem

Thus, the individual standing at the site boundary after the failure of a fueled experiment would receive a lower dose to the thyroid than if he were standing in a stable cloud existing at the unrestricted maximum permissible concentration listed in 10CFR, Part 20.

5. Question: What is the basis for your value for β_{eff} for the CAVALIER?

Answer:
β effective (0.8%) was established from measurements made at the Bulk Shielding reactor at Oak Ridge. The probability of escaping fast leakage while slowing down is greater for delayed neutrons than for prompt neutrons. The data showed that delayed neutrons are approximately 1.25 times as effective as prompt neutrons.
References: (Two Group Reactor Theory, by J.L. Meem);(Measurement of Effective Delayed Neutron fraction, Nucl. Sci. Eng. Volume 12, No. 4, April 1962)

6. Question: What is the anticipated worth of each control rod in the next core configuration?

Answer: The anticipated worth of each control rod in the next CAVALIER core configuration is approximately 1.0% $\Delta k/k$.

7. Question: Are there any scram functions or interlocks in addition to those identified in the Technical Specifications?

Answer:
Yes, the additional Scram functions not listed in Technical Specifications are:

1) Source Range channels - coincidence circuit will scram reactor if both channels measure period < 10 seconds.

2) Initiation of Evacuation alarm from any of four locations: (1) UVAR Control Room, (2) CAVALIER Control Room, (3) First Floor Hallway, (4) UVAR Experimental Area, Lower Level

3) Initiation of Fire Alarm system:
   (a) Sensors located in UVAR control room, CAVALIER control room, Heat Exchanger Room and Demineralizer Room.
   (b) Pull boxes located at corridor outside UVAR room, corridor on 2nd floor level, experimental area on lower level, and machine shop on lower level.

4) Key switch at CAVALIER console.

8. Question: The results of the CAVALIER loss of moderator accident are summarized in Table 9-1 of the SAR. Provide the basis and assumptions used in arriving at the results presented in Table 9-1.

Answer:

The maximum credible loss of moderator accident is a complete rupture of a drain line which exits the tank at the bottom while the reactor is operating and has one of the power histories listed in Table 9.1. Shortly after the pipe rupture the moderator drops to the scram set point level which is at its minimum limiting safety system setting of 7.25 feet above the core causing a scram and subsequent shutdown. In the column of Table 9-1, time after shutdown heading, the 17 minutes entry indicates a minimum time minus a safety margin required for the moderator (which serves also as a biological shield) to be lost so that the dose rates calculated represent a maximum credible dose rate. The entry of 28 hours represents what the maximum estimated dose rates would be at some later time.

The basis supporting the power histories are approximations of operating scenarios approaching the integrated power limit of 200 watt-hours per day. The basis for the figure of 17 minutes for loss of moderator in "Time after shutdown" column is the time required to drain the tank from 7.25 feet above the core to the core top. The inside diameter of the drain pipe is 1.25 inches and assuming no restriction to the drain flow rate due to boundary layer friction, 21 minutes would be required to drain the moderator down to the core top.

The dose rates for the distances given are based on a seven group dose rate calculation of the gamma emanations from fission product decay. The fission product inventory is based on the power history listed in Table 9-1. The geometry for the calculations are given in the first column with no attenuation due to any self shielding, air, or external shielding accounted for. The calculations for a single fuel element are based on a core with 16 elements and the fission product inventory is just one sixteenth of the total inventory. The figures are given for an average fuel element meaning that no account was taken for the flux distribution in the core.

A new Table 9-1 is attached. The dose rates were re-calculated in Dec. 1984 using more up to date data. The resulting dose rates are slightly higher than the original data.

TABLE 9-1

DOSE RATES FROM SHUTDOWN CORE AND FUEL ELEMENTS

Fission Product Source and Model Geometry	Reactor Power History	Time After Shutdown	Dose Rate at Distance Shown, R/hr		
			150 cm	220 cm*	550 cm**
REACTOR CORE	100 hr-at 10 watts	17 min (10^3 sec)	3.3	1.5	.24
AS A	100 hr-at 10 watts	28 hr (10^5 sec)	.45	.21	.034
POINT SOURCE	1 hr -at 100 watts	17 min (10^3 sec)	13.7	6.4	1.02
	1 hr-at 100 watts	28 hr (10^5 sec)	.10	.047	.008
AN AVERAGE	100 hr-at 10 watts	17 min (10^3 sec)	.20		
FUEL ELEMENT	100 hr-at 10 watts	28 hr (10^5 sec)	.028		
AS A	1 hr-at 100 watts	17 min (10^3 sec)	.85		
LINE SOURCE	1 hr-at 100 watts	28 hr (10^5 sec)	.006		

*Top of Tank
**Ceiling of Student Lab

9. Question: What are the set point values for the various scram parameters?

Answer: The present set point values for the various scram parameters are as follows:

Channel	Scram Setpoint *
Tank Water Level	7 feet 9 inches
Tank Top Radiation Monitor	15 mr/hr
Reactor Power Level (CIC)	60 watts
Reactor Power Level (GAMMA)	60 watts
Reactor Period (CIC)	10 seconds
Reactor Period (GAMMA)	10 seconds
Reactor Period (Source Range)	10 seconds (coincidence circuit)

*above the fuel

10. Question: Can the control rods be withdrawn as a gang?

Answer: No the control rods cannot be withdrawn electrically as a gang. The operating procedures allow any two rods to be withdrawn at a time up to 10 inches. Beyond 10 inches, only one rod can be withdrawn at a time. We cannot have a configuration where the critical rod positions would be less than 10 inches and meet the Technical Specification limits on shutdown margin and excess reactivity.

11. Question: Describe the CAVALIER fire protection system.

Answer: The CAVALIER control room has a fire detector that is actuated by rate of rise in temperature or maximum temperature of 135°F. This system alarms throughout the building and at the UVA Police dispatchers office which is manned 24 hours per day. This system also shuts down both reactors if they are in operation. There are manual pull-boxes at various locations around the building to actuate an alarm. A portable fire extinguisher is kept in the CAVALIER control room at all times.

12. Question: What are the maximum temperatures reached by the fuel elements for the two ramp reactivity insertion accidents discussed in the safety analysis of the CAVALIER? What assumptions were made for the coolant inventory during ramp insertions.

Answer:

The maximum fuel temperatures for the 1×10^{-4} $\Delta k/k$-s and 2×10^{-4} $\Delta k/k$-s ramp insertions are 1.4 ^{o}F and 0.46 ^{o}F rise from original tank temperature respectively. The coolant inventory assumed is the water in the CAVALIER moderator tank.

These results were obtained by summing two temperature rises. One temperature rise was due to coolant warming from the integrated power and the second rise calculated as a steady state temperature rise of the hot channel fuel plates at the peak power of the ramp insertion. This steady state temperature rise is very conservative since the temperature has not reached equilibrium at this transient peak.

The fuel temperature rise calculation was based on the results obtained for the UVAR since the UVAR in free convection is substantially similar to the CAVALIER. The power levels as well as the free convection heat transfer coefficient were scaled down to obtain a fuel plate peak temperature for the CAVALIER.

13. Question: How are the log-G and linear channel gamma chambers calibrated? Demonstrate that the calibration errors could not allow the CAVALIER to operate above the licensed power limit.

Answer: The Log-G and linear power channels on the CAVALIER receive their signals from two gamma chambers located on opposite sides of the moderator/shield tank. The signals from these two chambers are summed and then divided between the two instruments. The linear channel provides an accurate power level indication from its linear display over the several decades of reactor power. This instrument has no automatic reactor shutdown mechanism. The Log-G channel, on the other hand, initiates an automatic shutdown at a level determined from the power calibration but does not provide a very accurate power level indication due to its logarithmic display.

The Log-G and linear power channels are calibrated as follows:

(1) A gold foil is irradiated in a "known" flux using a one curie Pu-Be source in a graphite (sigma) pile.

(2) A number of gold foils are irradiated in one quadrant of the reactor within stringers used to place several foils in each element. These foils are inserted with the reactor shutdown and then irradiated for a set period of time at a particular power level as indicated on the linear power instrument.

(3) All foils are counted in a low background gas flow proportional counter. The counting data is then corrected for decay, detector efficiency, foil non-saturation and self shielding. The counting data is then used to compute the neutron flux required to produce the observed activity. The symmetry of the core is taken into account to make the assumption that the flux in each element in one quadrant of the core is the same as the similar element position in each of the three remaining quadrants.

(4) Fluxes are converted to fission rates which are in turn converted to units of thermal power production.

(5) The detector positions are changed as necessary in order to have 0.1 V on the linear power instrument correspond to one watt of reactor power. The automatic shutdown initiation on Log-G is set such that it is below the limiting safety system setting of 100 watts. Usually this set point is between 50 and 60 watts.

There are several potential sources of error in this calculation and some built in conservatism. The conservatism is part of the calculation which converts the fluxes seen by the foils to the fission rate in the core. Since the flux at each fuel element is measured only by several foils, the flux seen by each foil is, for calculational purposes, assumed to exist throughout the section of the element in which it is contained. This means that if there are three foils in an element then the element volume is divided among the foils based on their positions. The flux seen by a central foil is assumed to be constant throughout the element cross-section between "points" located halfway between the foils. The flux determined by the foil on each end of the stringer is considered to extend to the end of the fueled region on that end. Thus, the actual sinusoidal shape of the flux and the assumption that the flux at each foil is the same throughout the entire volume of a section of the element, lead to an overestimation of the fission rate at a particular power level. The precise degree of this overestimation is not known.

All other potential sources of error have an equal probably of leading to either an overestimation or underestimation of the reactor power level. Counting statistics, half life, cross section, foil mass, timing of the activation, detector efficiency, foil self-shielding correction and the assumption that the flux in one quadrant of the core is the same as in the other quadrants could all potentially err in either the positive or negative direction. It is not likely that there will be more in a non-conservative direction than in a conservative direction. The estimation that these potential errors will approximately cancel is not considered unreasonable. Some of the potential sources of error are not quantifiable, making a strict statistical analysis difficult, if not impossible.

In conclusion, the CAVALIER power calibration has been checked several times over the last ten years and has varied from the initial calibration by only a small percentage each time. These results give confidence that the current calibration is correct within a small margin of error. In order to insure that the licensed power limit of 100 watts is not exceeded, all automatic shutdown mechanisms which use reactor power level for their signal input are set at no greater than 60 watts as indicated by the reactor power calibration. This setpoint allows for an error of up to 40% in the power calibration and there is no indication from any previous calibrations that the potential error is anywhere near that magnitude.

14. Question: Provide the basis and assumptions used in the fission product release analyzed in Sec. 9.4.4 of the SAR.

Answer:

There is no specific basis for the fission produce release analyzed in Section 9.4.4 of the CAVALIER Safety Analysis Report. The accident analyzed is the maximum excursion used during the SPERT tests which produced no release of fission products. No accident scenario was postulated which would produce this excursion but certainly a 10 MW-second pulse is greatly in excess of any accident which might be reasonable expected to occur on the CAVALIER.

All assumptions used in the calculation of the doses received by off-site and on-site personnel from a release from the accident postulated are contained in the SAR.

15. Question: Potentially radioactive material can be released from the CAVALIER moderator tank to the pond and then from the pond to the environment. How do you ensure that releases to unrestricted areas are controlled and that radioactivity in the effluent does not exceed the Maximum Permissible Concentration (MPC)?

Answer: The tank of water which provides both moderation and shielding for the CAVALIER reactor is not normally drained for any reason but if necessary it can be released to the facility's hold-up pond through a valve at the bottom of the tank and a drain in the reactor pit. Before release from the tank the water is analyzed to insure that no unusual levels of radioactivity are present. The usual level of activity in the tank should not be much different than tap water due to the reactor's low power level, the low number of operating hours and because the water is continuously circulated through a demineralizer. Any release would be under the supervision of a senior reactor operator with directions from the reactor supervisor and reactor health physicist.

After water has been released to the pond (a restricted on-site area), release of water from the pond to unrestricted off-site areas is made according to standard operating procedures. In the pond, dilution with rainfall or fire hydrant water and radioactive decay with time serves to lower the effluent artificial activity significantly. Three samples of the pond are taken prior to release to assure that the contained activity is below unrestricted area MPC's. Additional samples of the pond effluent are taken during each release to verify the pre-release sampling and to record the actual radioactive concentration of the released water.

16. Question: Describe the capability for detecting and measuring radiation at the CAVALIER facility. Include the following information (1) the generic type of detectors (for example, ionization chamber), (2) tne type of radiation monitored, and (3) the operable range of each experiment.

Answer:
The fixed area monitors associated with the CAVALIER reactor are 3 Victoreen model 555 monitors. They incorporate Geiger-Muller detectors and have a useful range from 0.01 to 10^3 mr/hr. Personnel monitoring includes film badges and ring badges for measuring β, γ, and neutron exposure, pocket dosimeters (ion chambers) with a range from 0 to 200 mr. and digital personnel monitors that have a range from 0.01 to 1000 mr.

In addition, the following portable monitors are available at the facility.

Make	Model	Detector	Serial #	Type of Radiation	Range
Keithley (5)	36100 Digital	Ion Chamber	11695	β,γ	0.5mR/hr-20/R/hr
			11547	β,γ	
			11699	β,γ	
			11181	β,γ	
			11181	β,γ	
Vic. Radector(3)	2035	Ion Chamber	725	β,γ	0.1mR/hr-1000 R/hr
			1326	β,γ	
			1327	β,γ	
Victoreen (2)	490	G.M. Tube	339	β,γ	0.1mR/hr-2 R/hr
			389	β,γ	
Victoreen	493	G.M. Tube	148	β,γ	0.1 mR/hr-50 mR/hr
Jordan	AGB-10KG-SR	ion chamber	1060	β,γ	0.1 mR/hr-10 kR/hr
Kaman	D-300		134	n	1.0 mR/hr-10 R/hr
Tech.Assoc.	2202D		6059	n	0.1 mR/hr-10R/hr
Vic. Frisker	495	G.M.Pancake	162	β,γ	0-500000 cpm
Tennelec	LB5100	Gas-Proportional	-	α,β	
Eberline(3)	RM-14	G.M. Pancake	4267	β,γ	0-50 k cpm
			4242		
			4920		
Eberline	E-520E	G.M.	7744	β,γ	0-50 k cpm
Eberline	HFM-3	G.M.	-	β,γ	0-500 cpm
Packard Liq. Scint. Spec.	Tri-Carb	Scintilation	07065	α,β	
Nuc. Supplies	RM-100	G.M	L-00327-3	β,γ	0-50000 cpm
Eberline	E-520	G.M.	3103	β,γ	0-2000 cpm

Also have various Ge-Li, NaI, detectors to be used with multichannel analyzers for isotopic analysis

17. Question: Describe the capability for measuring radioactivity levels in the reactor moderator and any effluents. Include the following information: (1) generic type of detectors (for example, low background proportional counter), (2) the types of radiation or radioactive material monitored, and (3) the minimum detectable activities or concentrations.

Answer: Radioactivity levels of the reactor moderator and any effluents are analyzed for specific activity (µci/ml) using a low background proportional counter for checking gross Beta activity. Isotopic analysis is performed using a Ge-Li detector and multichannel analyzer to identify individual isotopes. The minimum detectable activities vary depending on the isotope involved, counting efficiency, counting geometry, count time, sample composition, etc. However our equipment is relatively modern and similar to equipment used throughout the nuclear industry and the MDA's achieved are sufficient to meet regulatory standards.

18. Question: The Radiation Safety Office and the reactor staff are shown to be organizationally separate. Describe the interaction between the groups that assures adequate communication of radiation safety information.

Answer:
A Health Physicist from the Radiation Safety Office is assigned to the Reactor Facility and either he or a health physics technician is present at the facility on a day to day basis to perform surveys, handle waste disposal or transfer of radioactive material. The Health Physicist is informed and is present for the conduct of any unusual experiments, shipment of waste or spent fuel, etc. that might involve radiation safety. The Health Physicist also attends weekly staff meetings of the Reactor Staff and is kept informed of the daily operations of both the UVAR and CAVALIER reactors.

The Radiation Safety Officer, to whom the Reactor Health Physicist reports, has a permanent seat on the University of Virginia Reactor Safety Committee which meets at least semi-annually. The Radiation Safety Officer is also the liason between the Reactor Safety Committee and the Radiation Safety Committee.

19. Question:

Provide a summary of the facility's annual personnel exposures (The number of persons receiving a total annual exposure within the designated exposure ranges similar to the report described in 10 CFR 20.407(b) for the last 5 years of operation.) Estimate what fraction of these exposures may be attributed to the operation of the CAVALIER.

Answer: A summary of personnel exposure for the last five years is as follows:

Whole Body Exposure Range (Rems)	Number of Individuals in Each Range				
	1980	1981	1982	1983	1984 (as of 11/84)
No measureable exposure	73	44	60	45	33
measureable exposure < 0.1	46	52	44	54	56
0.1 to 0.25	0	4	12	3	4
0.25 to 0.5	0	3	3	2	5
0.50 to 0.75	0	0	0	0	0
0.75 to 1.0	0	0	0	0	0
>1.0	0	0	0	0	0

Due to the low power level of the CAVALIER we estimate that less than 1.0% of personnel exposures are due to the operation of the CAVALIER. Most of the exposures are due to handling samples for activation analysis work performed with the 2 MW UVAR reactor.

20. Question: Describe the radiation safety training program for the reactor operations staff and other personnel who may work around the CAVALIER. Include a description of the process for verifying that training has been accomplished.

Answer: An upgraded training program in radiation safety has recently been developed by the Reactor Health Physicist. This new program will be implemented in January, 1985. An outline of the program is as follows:

1. Objectives

The primary objectives of the training program will be to accomplish at least the following:

A. to instruct personnel on the biological effects of radiation, the risks associated therein and the basis for risk estimates.

B. to instruct individuals concerning UVa Radiation Safety rules and procedures.

C. to provide information to individuals that will enable them to keep their own radiation exposures ALARA.

D. to provide the information needed to enable each person to comply with NRC regulations and license conditions.

2. Procedures

A. Current Personnel Training

1. Current workers and frequent visitors will be individually interviewed to determine the extent of prior HP training at the UVa Reactor Facility. These interviews will be documented and voids in training history will be corrected.

2. New workers and frequent visitors will receive an HP orientation to include familiarization with the University of Virginia Reactor Facility General Radiological Health Training Manual. They will receive an oral, written and/or practical test on their knowledge in this area. Permanent training records on each individual will be kept by the reactor HP.

B. Future Training Schedule

1. All workers and frequent visitors will receive ongoing HP training on at least an annual basis. Records of this training will be kept.

2. Quarterly topics of interest will be presented and workers and visitors will be strongly recommended to attend.

3. Oversight

A. The program will be reviewed and revised as needed to meet changing conditions.

B. Each individual's training will be reviewed in order to ensure that they are adequately trained for each task to which they are assigned.
C. The program will be reviewed to ensure that training does not become needlessly repetitive.

4. Additional Training Requirements
   A. The reactor HP will continue to provide the UVA Safety Guide Lecture to users of Radioactive Materials under AU #40=80.
   B. The reactor HP will work with the reactor supervisor to develop a training program to cover the health physics topics included in the reactor operator's requalification exam.

Specific topics to be covered are as follows:

A. Measurement & Control of Exposure to Radiation & Radioactive Material

   1. Controlling Exposure
   2. Sources (Origins) of Radioactive Materials (contamination and radiations at the facility)
   3. Source Identification and Control
   4. Types and Forms of Radioactive Materials (contamination)
   5. Detection and Control of Contamination
   6. Radiation Measurement and Survey Instruments
   7. Radioactive Wastes, Their Origins, Storage, Handling & Disposal

B. Radiation Protection Program

C. Biological Effects of Radiation

D. Preparation for Emergencies & Incidents

Individuals with little or no previous training in radiation protection will receive a one-hour orientation lecture prior to the training session covering the following topics:
   A. Radiation Fundamentals
   B. Types of Radiations & their Characteristics
   C. External Dosimeter

Individuals with extensive training and/or experience in radiation safety and health physics may be exempted from the training program by submitting a written request to the Radiation Safety Office. This request should include the dates of the applicable training and experience, a description of that training and experience and a description of current job duties. If a review of each individual request shows clearly appropriate background in the material, that individual will not be required to attend the full three-hour presentation, but will be required to attend a one-hour lecture on HP concerns specific to this facility.

All lectures will consist of an oral presentation as well as relevant audio-visual materials. All lectures will end with a brief written exam covering the lecture material.

The training program to begin in January, 1985 will include the following personnel.

1. Faculty, Reactor Staff, Grad Students, Engineering Science Students. One hour lecture covering pertinent HP topics specific to the facility. This lecture will include the rules and regulations applicable to our license, as well as risks and personal rights associated with occupational radiation exposure (to be taught during the first seminar class period).

2. Engineering Science and EP Students. In addition to the one-hour HP lecture, this group will be required to attend a three-hour short course covering a broader range of HP topics which these students may not have been exposed to (to be taught during one evening session).

3. First and Second year students frequenting the Facility. This group will attend the three-hours short course along with the ES and EP students. They will also attend a special one-hour introductory lecture prior to the short course. (this intro. lecture will also be a makeup session for those people in category one who missed that session.

4. Physical Plant Personnel. This will be handled separately.

5. Non-operating staff. A special three hour short course will be given to those individuals (secretaries, maintenance, part-time student help) who do not routinely enter the controlled areas. (It will be taught during work hours).

21. Question: Cite documentation that demonstrates the commitment of UVA to the ALARA principle.

Answer:
A program of commitment by the University of Virginia to the ALARA principle was established 1979. This commitment was forwarded to the NRC in 1982 as part of the License renewal for the UVAR reactor (License R-66). A copy of that commitment is attached and is still valid. Mr. R.O. Allen is still chairman of the Radiation Safety Committee but Mr. B.G. Copcutt is now Radiation Safety Officer.

22. Question: In accordance with our discussion during the site visit the week of November 5, and with Section 2 of ANSI/ANS 15.1 (1982), the Safety Limits for a nonpower reactor should be closely related to protecting the integrity of principal barriers that guard against the uncontrolled release of radioactivity. Please review your Technical Specifications and include with the responses to these questions any changes necessary to achieve consistency with the guidance noted above.

Answer:
The proposed Technical Specifications have been revised to define the Safety Limits in conformity with the generally accepted definitions as outlined in ANS 15.1. These revisions are included as an appendix to the answers to these questions as substitute sheets for the Technical Specifications volumes previously remitted to the NRC.

23. Question: During our visit we pointed out some other areas in your Technical Specifications where clarity, internal consistency, or conformance with current NRC practice could be improved. Please consider and propose appropriate changes with the responses to these questions.

Answer:
Other proposed revisions to the Technical Specifications which address these areas are included in the appendix to the answers to this questionnaire, in the form of substitute pages for the Technical Specification volumes previously remitted to the NRC.

24. Question: Please provide copies of the figures listed in the attachment that will be of suitable quality for use as originals in our Safety Evaluation Report.

1) the expected core loading (similar to SAR figure 3.5)

2) a drawing of the standard fuel element in the next core loading

3) same as item 2 for a control rod fuel element.

4) a drawing of a control rod drive unit.

Answer:
These drawing's are included as part of our response to these questions.

PROGRAM FOR MAINTAINING OCCUPATIONAL
RADIATION EXPOSURES AT THE UNIVERSITY OF VIRGINIA
AS LOW AS REASONABLY ACHIEVABLE

DECEMBER 4, 1979

I. Management Commitment

a. We, the management of this University, are committed to the program described in this paper for keeping exposures (individual and collective) as low as reasonably achievable (ALARA). In accordance with this commitment, we hereby establish an administrative organization for radiation safety and develop the necessary written policy procedure instructions to foster the ALARA concept within our institution. The organization will include a Radiation Safety Committee (RSC), and a Radiation Safety Office (RSO). We are committed to following the guidance provided by U. S. Nuclear Regulatory Guides 8.10 and 8.18.

b. In addition to maintaining doses to individuals as far below the limits as is reasonably achievable, the sum of the doses received by all exposed individuals will also be maintained at the lowest practicable level. It would not be desirable, for example, to hold the highest doses to individuals to some fraction of the applicable limit if this involved exposing additional people and significantly increasing the sum of radiation doses received by all involved individuals.

II. Radiation Safety Committee (RSC)

a. Review of Proposed Users and Uses

1. The Radiation Safety Committee will thoroughly review the qualifications of each potential Authorized User with respect to the types and quantities of materials and uses for which he has applied to assure that the user will be able to take appropriate measures to maintain exposure ALARA.

Amended 8/7/81

2. When considering a new use of byproduct materia', the Radiation Safety Committee will review the efforts of the Authorized User to maintain exposure ALARA. The user should have systematized procedures to ensure ALARA, and should have considered the use of special equipment such as syringe shields, rubber gloves, etc., in his proposed use.

3. The Radiation Safety Committee will ensure that the user justifies his procedures and that they will result in ALARA doses (individual and collective).

b. Delegation of Authority

1. The Radiation Safety Committee will delegate sufficient authority to the Radiation Safety Office for enforcement of the ALARA concept.
2. The Radiation Safety Committee will support the Radiation Safety Office in those instances where it is necessary for the Radiation Safety Officer to assert his authority. Where the Radiation Safety Officer has been overruled, the Committee will record the basis for its action.

c. Review of ALARA Program

The Radiation Safety Committee will review all instances of deviations from the ALARA philosophy and, at least annually, will review the entire radiation safety program in order to evaluate the University of Virginia's overall efforts for maintaining exposures ALARA. Information in support of the review will normally be supplied by the Radiation Safety Office.

d. Public Statement of Commitment by the Radiation Safety Committee to ALARA.

All elements of our institution will be informed of the Radiation Safety Committee's commitment to the ALARA philosophy.

1. The Radiation Safety Committee will ensure that employees are aware of the Radiation Safety Committee's commitment to the ALARA philosophy.

Amended 8/7/81

2. The Radiation Safety Committee will demonstrate its commitment to the ALARA concept through the methods employed in its review of proposed users and uses.

III. Radiation Safety Office (RSO)

a. Periodic Review and Audit of the Radiation Safety Program for Compliance with ALARA Concepts. Frequent reviews of procedures will be conducted.

1. The Radiation Safety Office will review and audit, on a regular basis (at least annually), the effectiveness of its own radiation protection program in maintaining doses (individual and collective) ALARA.

2. The Radiation Safety Office will review exposures of authorized users and occupational workers to determine that their exposures are ALARA.

b. The Radiation Safety Office's Education Responsibilities for an ALARA Program

1. The Radiation Safety Office will assure that authorized users understand the ALARA philosophy and know that management, the Radiation Safety Committee, and the Radiation Safety Office are committed to implementing the ALARA concept.

c. Cooperative Efforts for Development of ALARA Procedures

Individuals who must work with ALARA concepts will be given opportunities to participate in formulation of the procedures that they will be required to follow.

1. The Radiation Safety Office will maintain close contact with all users in order to develop ALARA procedures for working with radioactive materials.

d. Reporting and Reviewing Instances of Deviation from Good ALARA Practices

1. The Radiation Safety Office will investigate all instances of deviation from good ALARA practices; and, if possible, determine the causes.

When the cause is known, the Radiation Safety Office will propose changes in the program to maintain exposures ALARA.

2. The Radiation Safety Office will report all significant instances of deviation from ALARA concepts to the Radiation Safety Committee for review.

IV. Authorized Users

a. New Procedures Involving Potential Radiation Exposures

1. The Authorized User will consult the Radiation Safety Office and/or the Radiation Safety Committee before using radioactive materials for a new procedure.

2. The Authorized User will consider all procedures thoroughly before using radioactive materials to ensure that exposures will be kept ALARA. This may be enhanced through the application of trial runs.

b. Responsibility of the Authorized User to Those He Supervises

1. The Authorized User will thoroughly explain the ALARA concept and his commitment to maintain exposures ALARA to all of those he supervises.

2. The Authorized User will ensure that his occupational workers are trained and educated in good health physics practices and in maintaining exposures ALARA.

3. The Authorized User will be responsive to the radiation safety concerns of the individuals that he supervises.

c. Continuing Review of ALARA Concepts by the Authorized User

1. The Authorized User will continuously review his procedures to ensure that his ALARA program is optimal.

2. The Authorized User will maintain contact with the Radiation Safety Office to ensure that he is aware of and employs the most current methods to maintain exposures ALARA.

V. Occupational Worker

a. What the Occupational Worker Must Consider about ALARA

1. The worker will implement ALARA procedures developed by the Authorized User and the Radiation Safety Office.

2. The occupational worker will know what recourses are available if he feels that ALARA is not being promoted on the job.

3. The occupational worker will understand the ALARA concept and will review his own working conditions for the implementation of ALARA principles.

VI. Establishment of Action Levels in Order to Achieve Reductions in Individual Occupational Exposures

This institution hereby establishes exposure action levels for specific kinds or classes of operations which, when exceeded, will trigger investigation by the Radiation Safety Officer. The exposure action levels that we have established are listed in Section VII below. These levels apply to the exposure of individual workers. The exact levels have been determined based on our institution's radiation exposure history and a thorough analysis of our current program. We will maintain on file at our institution an account of the considerations used in establishing action levels.

We will investigate the causes of personnel exposures that exceed our established exposure action levels. In the event of a personnel exposure that exceeds our established action levels or 10% of Maximum Permissable Dose (MPD), whichever is higher, we will maintain accounts of our investigation for inspection by the NRC. As a minimum, these accounts will include the cause of the exposure, the action taken to correct the situation and the follow-up action taken.

VII. Action Levels

The specific action levels established by this institution are as follows:

Kind or Class of Operation	Action Level
1. Department of Radiology	10% of MPD
2. All other Departments	10% of MPD

Amended 8/7/81

VIII. Signature of Certifying Officials

We hereby certify that this institution is committed to the ALARA Program set forth above.

Ralph O. Allen	Harold W. Berk
(Signature)	(Signature)
Ralph O. Allen	Harold W. Berk
Name	Name
Chairman, Radiation Safety Committee	Radiation Safety Officer
Title	Title

Institution name and address:

Rectors and Visitors
University of Virginia
Radiation Safety Officer
Box 262 - Attention: H. W. Berk, Radiation Safety Officer
Charlottesville, Virginia 22908

CAVALIER REACTOR LOADING CHART

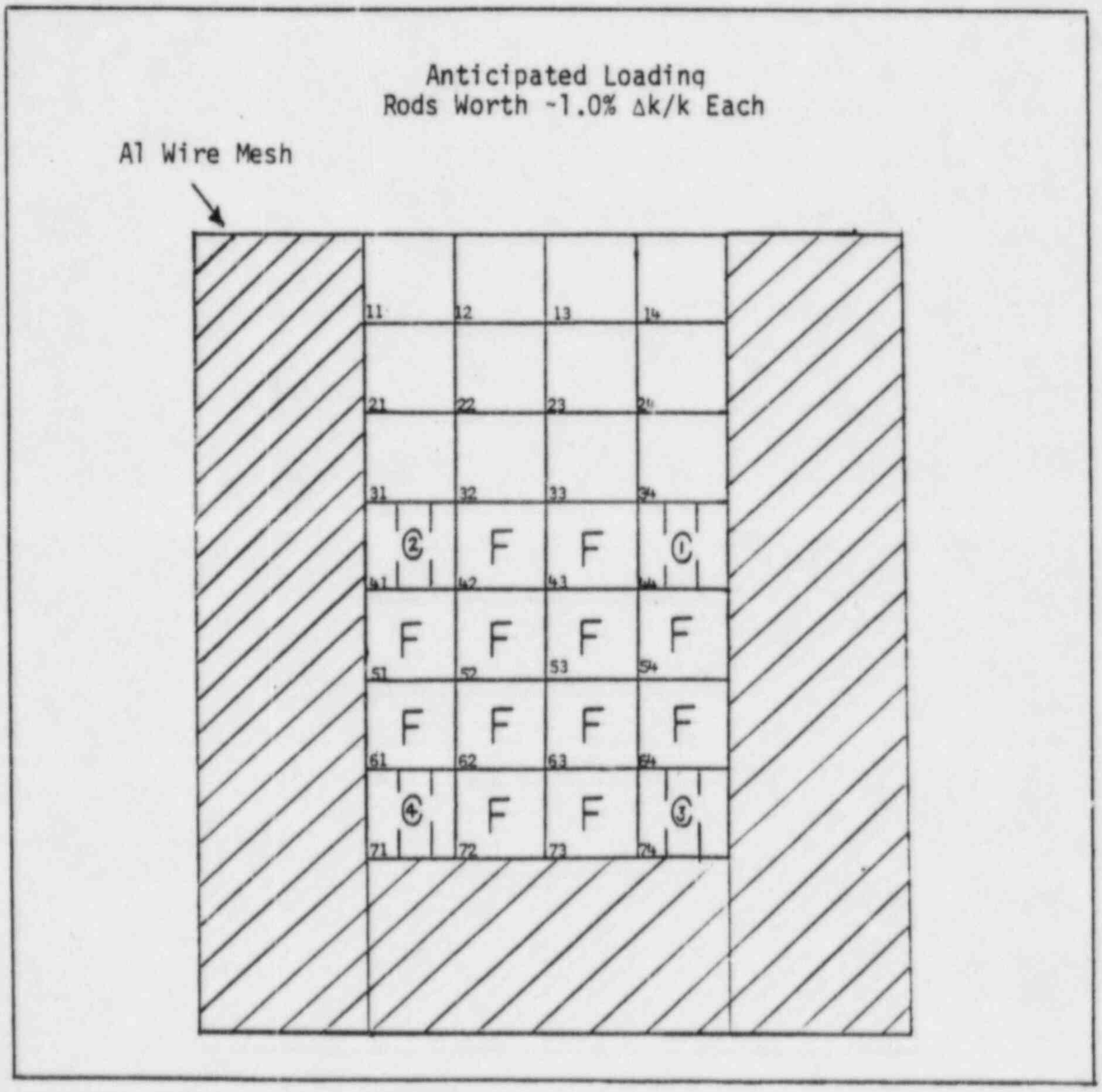
Loading No. ______________ Date Approximately December 1984

Total Mass 2726 gram U-235

F - Normal Fuel Element
P - Partial Fuel Element
C - Control Rod Fuel Element
E - Removable Plate Fuel Element

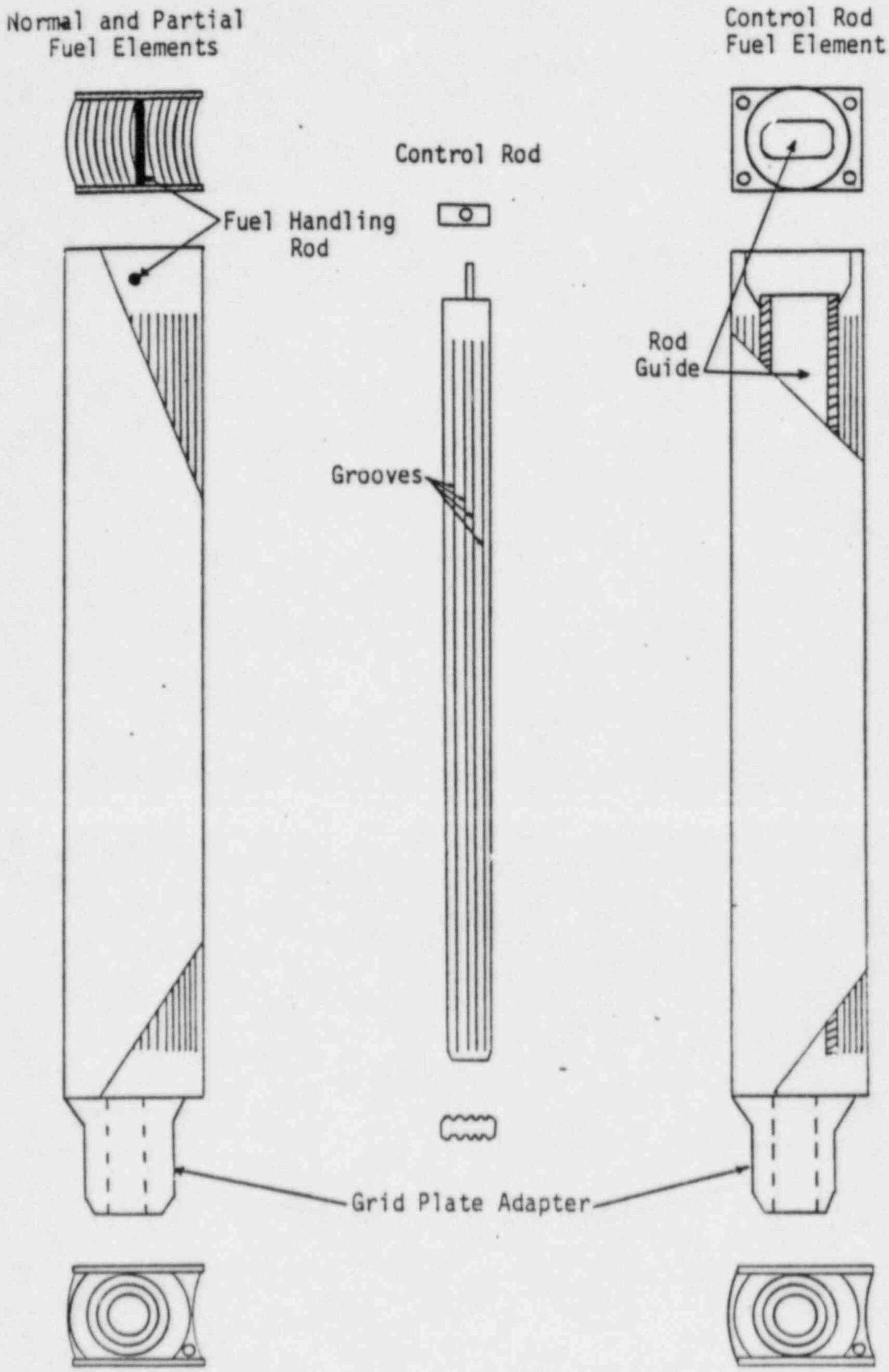
Excess Reactivity ~1.0% Δk/k

Shutdown Margin ~1.8% Δk/k
One Rod Withdrawn

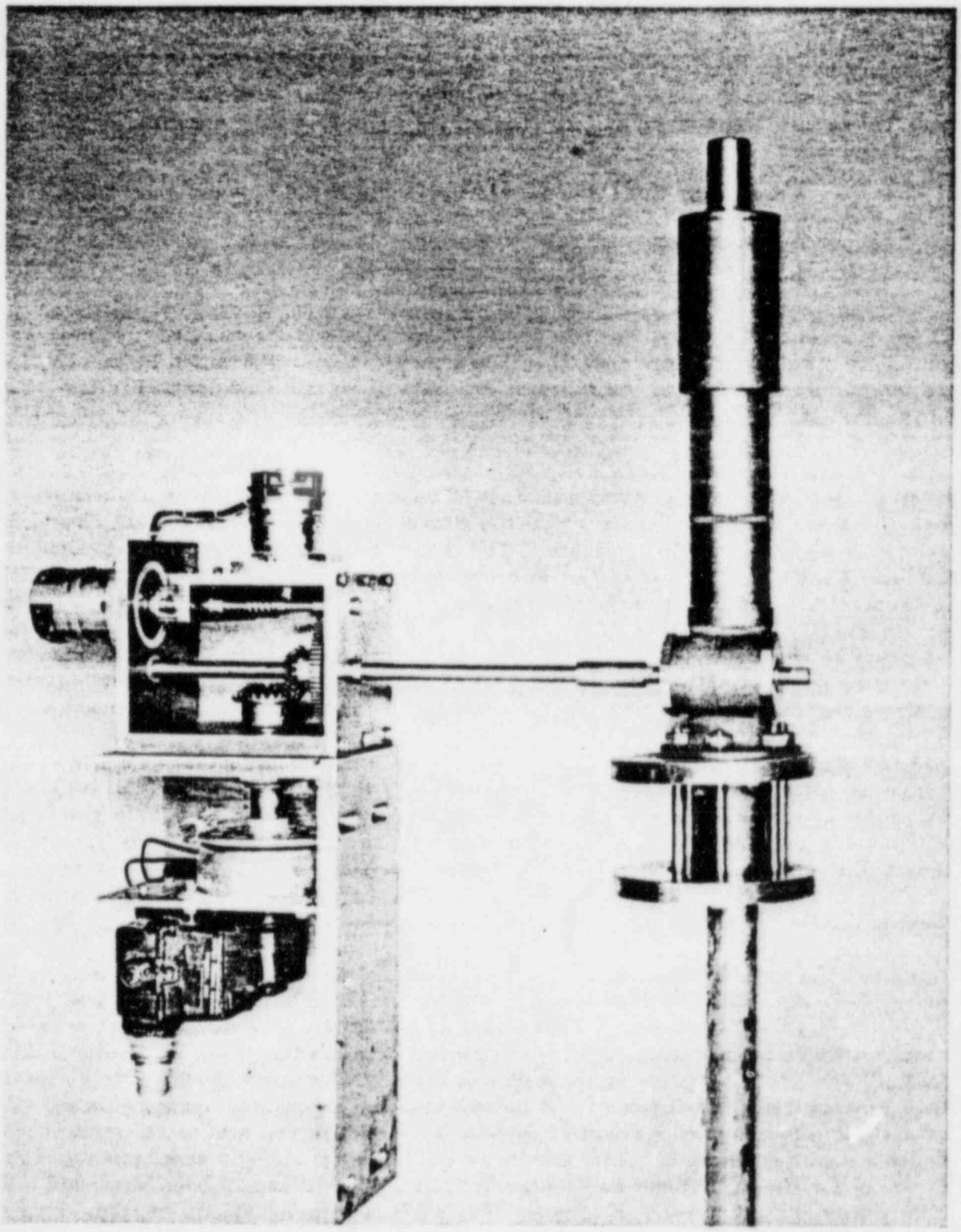


N

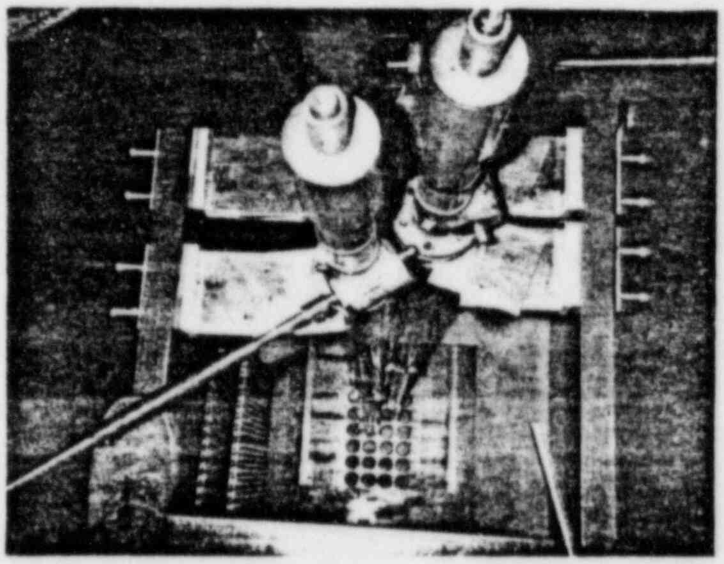
CONSOLE



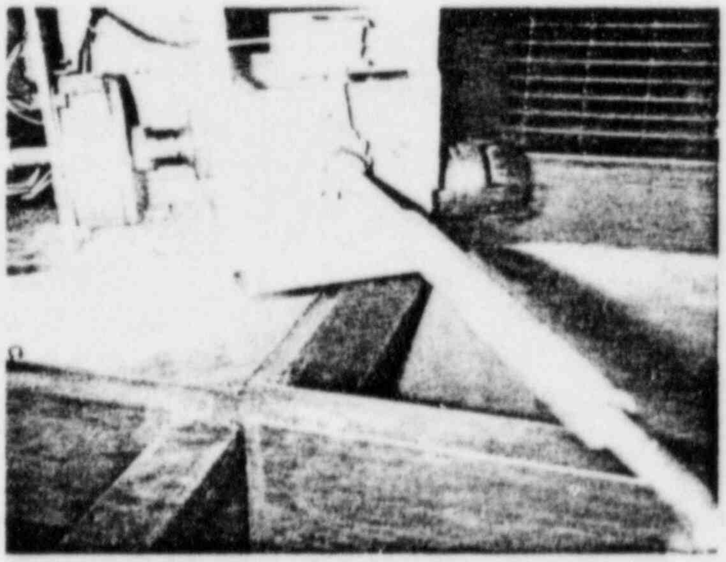
FUEL ELEMENTS



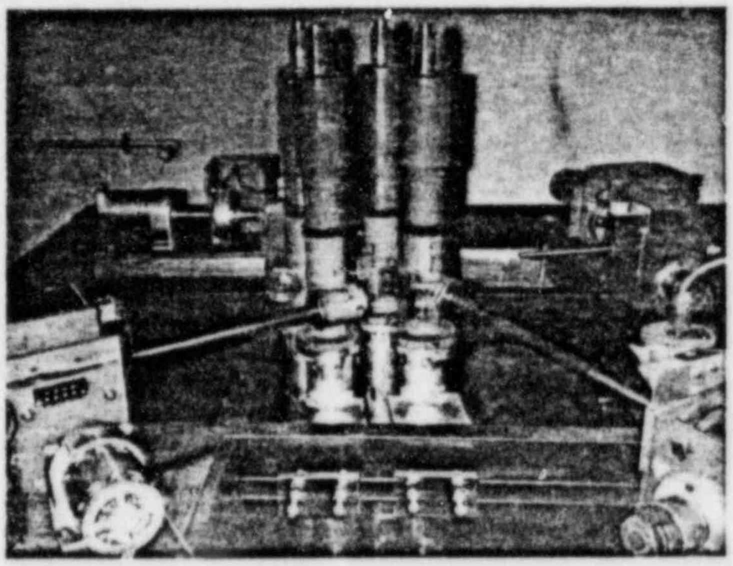
Rod Drive Unit



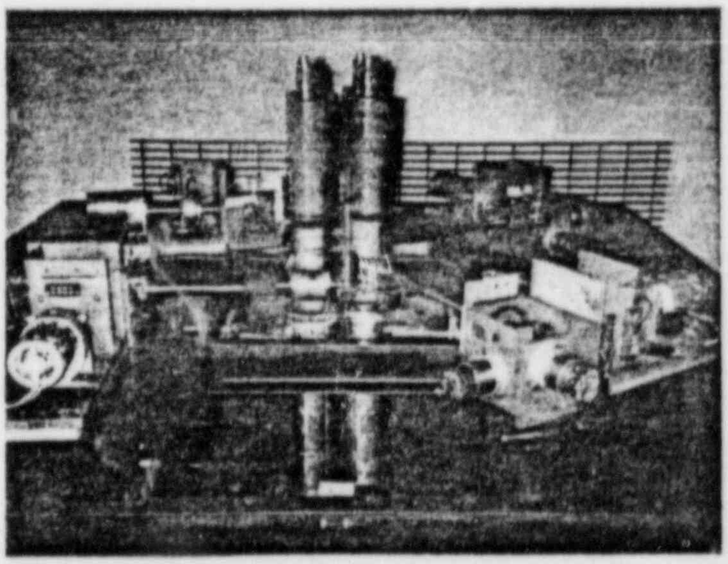
Top View of Typical Core Loading



Outmotion Latch



A. Side View



B. End View

Rod Drive Arrangement for Typical Core Loading

FACILITY LICENSE R-123

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF VIRGINIA

CAVALIER REACTOR

DOCKET NO. 50-396

TABLE OF CONTENTS

1.0 Definitions

The terms Safety Limit (SL), "Limiting Safety System Setting" (LSSS), "Limiting Condition of Operation" (LCO), "Surveillance requirements," and "design features" are as defined in 10 CFR 50.36.

Channel Calibration: A channel calibration is an adjustment of the channel so that its output responds, with acceptable range and accuracy, to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification should include comparison of the channel with other independent channels or methods of measuring the same variable, where this capability exists.

Channel Test: A channel test is the introduction of a signal into a channel to verify that it is operable.

Experiment: An experiment is (1) any apparatus, device, or material placed in the reactor core region (in an experimental facility associated with the reactor, or inline with a beam of radiation emanating from the reactor) or (2) any incore operation designed to measure reactor characteristics.

Experimental Facility: An experimental facility is any structure or device associated with the reactor that is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

Explosive Material: Explosive material is any solid or liquid that is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in "Dangerous Properties of Industrial Materials" by N.I. Sax, or is given an Identification of Reactivity (stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, "Identification System for Fire Hazards of Materials," also enumerated in the "Handbook for Laboratory Safety" published by the Chemical Rubber Company.

Fueled Experiment: A fueled experiment is any experiment that contains U-235 or U-233 or Pu-239. This does not include the normal reactor core fuel elements.

Measured Value: The measured value of the process variable is the value of the variable as it appears on the output of a measuring channel.

Measuring Channel: A measuring channel is the combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.

Movable Experiment: A movable experiment is one that may be inserted, removed, or manipulated while the reactor is critical.

On Call: To be on call refers to an individual who (1) has been specifically designated and the designation is known to the operator on duty, (2) keeps the operator on duty informed of where he may be contacted and the phone number, and (3) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g. approximately 30 minutes).

Operable: A component or system is operable when it is capable of performing its intended function in a normal manner.

Operating: A component or system is operating when it is performing its intended function in a normal manner.

Reactivity Limits: Quantities are referenced to ambient tank water temperature with the effect of Xenon poisoning on the core activity accounted for if greater than or equal to 0.05% $\Delta k/k$. The reactivity worth of Samarium in the core will not be included in reactivity limits. The reference core condition will be known as the cold, xenon free critical condition.

Reactor Operation: The Reactor is in operation when not all of the shim rods are fully inserted and six or more fuel elements are loaded in the grid plate.

Reactor Safety System: The reactor safety system is that combination of measuring channels and associated circuitry that forms the automatic protective system of the reactor.

Reactor Secured: The reactor is secured when (1) all shim rods are fully inserted, (2) the console key is in the off position and is removed from the lock, and (3) no work is in progress in core involving fuel or experiments or maintenance of the core structure, control rods, or control rod mechanisms.

Reactor Shutdown: The reactor is in a shutdown condition when all shim rods are fully inserted.

Reportable Occurrence: A reportable occurrence is any of the conditions described in Section 6.4.2 of these specifications.

Secured Experiment: A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be sufficient to overcome those to which the experiment might be subjected by hydraulic, pneumatic, buoyant or other forces that are normal for the operating environment of the experiment.

Shim Rod: A shim rod is a control rod fabricated from borated stainless steel, which is used to compensate for fuel burnup, temperature, and poison effects. A shim rod is magnetically coupled to its drive unit allowing it to perform the function of safety rod when the magnet is de-energized.

Surveillance Time Intervals

Annual - Interval not to exceed 15 months

Semi-annually - Interval not to exceed 7 1/2 months

Quarterly - Interval not to exceed 4 months

Monthly - Interval not to exceed 6 weeks

Weekly - Interval not to exceed 10 days

Daily - must be done during the calendar day

Tried Experiment: A tried experiment is (1) an experiment previously performed in this reactor or (2) an experiment for which the size, shape, composition, and location does not differ significantly enough from an experiment previously performed in this reactor to affect reactor safety.

True Value: The true value of a process variable is its actual value at any instant.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit

Applicability: This specification applies to the maximum temperature of the fuel or fuel cladding that could cause the uncontrolled release of fission product activity.

Objective: To assure that the reactor is operated in such a manner that the fuel cladding integrity is maintained to prevent an uncontrolled release of fission product activity that could adversely affect facility personnel and the general public.

Specification: The fuel consists of a U-AL alloy clad in aluminum. The safety limit is specified as the melting point of the fuel or cladding which is 1220°F or 660°C.

Basis: The melting point of aluminum is that temperature at which the fuel integrity would be breached, thereby causing an uncontrolled release of fission product activity. With the low power operating restrictions of the CAVALIER and considering the consequences of abnormal events as analyzed in the SAR, there is virtually no possibility that this temperature could ever be reached.

2.2 Limiting Safety System Settings

Applicability: This specification applies to limitations on setpoints pertaining to the thermal power level of the reactor and the water level above the fuel which would initiate an automatic shutdown of the reactor.

Objective: To assure that automatic protective actions are initiated in a manner consistent with maximizing safety for the reactor operators and minimizing the chance for their exposure, or the exposure of the public, to ionizing radiation.

Specification:

Maximum Reactor Power Level	100 watts
Minimum Tank Water Level	6.25 feet above top of fuel

Actual set-points may be set at more conservative values than those specified above.

Bases: The limitations on reactor power level and water height above the fuel was established by calculated radiation levels above the water level of the moderator tank as developed in section 3.2 of the CAVALIER SAR. The water height of 6.25 feet would lead to a dose rate of about 60 mr/hr above the reactor tank, at a power level of 100 watts, which would produce a radiation level in the control room work area which is significantly less than 60 mr/hr (and 10 CFR Part 20 limits). The actual set-points for these parameters are normally set much more conservatively than the specification limits. Operating experience over the past 10 years with the power level at approximately 50 watts and the water level at approximately 8 feet has indicated a dose rate at the top of the tank at approximately 4 mr/hr and less than 1.0 mr/hr in the control room area.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Power Operation

Applicability

This specification applies to the average power rating of the CAVALIER.

Objective

To assure that the reactor is operated in a manner consistent with maintenance of a low level of residual radioactivity in the fuel elements.

Specification

The Average Power Rating shall be less than 200 watt-hours/day where the averaging period shall not exceed 24 hours.

Bases

This rating will limit production of fission products to a level less than that analyzed in the Fission Product Released Section 9.4.4 of the CAVALIER SAR. This analysis indicates that the 2 hour doses at the site boundary after a very unlikely release of fission products from the fuel are within 10 CFR Part 20 averaged over a period of a year.

3.2 <u>Reactivity</u>

<u>Applicability</u>

These specifications apply to the reactivity condition of the reactor, and the reactivity worths of control rods and experiments.

<u>Objective</u>

The objective is to assure that the reactor can be shut down safely at all times, even with an experiment failure.

<u>Specifications</u>

The following specifications apply to the reactivity conditions for reactor operation.

(1) The minimum shutdown margin provided by control rods with secured experiments in place and referred to the cold, xenon free condition with the highest worth control rod fully withdrawn, is greater than 0.4% $\Delta k/k$.

(2) Any experiment with a reactivity worth greater than 0.35% $\Delta k/k$ must be a secured experiment.

(3) The total reactivity worth of all experiments is less than 1.6% $\Delta k/k$ and the reactivity worth of a single experiment is limited to 0.5% $\Delta k/k$.

(4) The excess reactivity including experiments in the core at any time shall be less than 1.6% $\Delta k/k$.

(5) The Alternate Reactivity Insertion System is operable.

These conditions must be met at all times with the following exceptions.

(a) With the ARIS system operable, the reactor may be operated up to 5 watts to measure the reactivity worth of experiments.

(b) The reactor may be operated up to 60 watts to calibrate control rods, after a major core configuration change, to determine if specifications 3.2.1 through 3.2.4 are met. The ARIS system must be operable during all operations.

Bases

The shut down margin required by Specification 3.2(1) is necessary so that the reactor can be shut down from any operating condition and that it will remain shut down without further operator action.

The reactivity limitations in Specifications 3.2 (2) and (3) are based on the guidelines for "Development of Technical Specifications for Experiments in Research Reactors" given in Regulatory Guide 2.2 as developed in the CAVALIER SAR. The reactivity worth limitations of specifications 3.2 (2) for a secured experiment and 3.2 (3) for any single experiment limit the reactor period to approximately 2 seconds.

The reactivity of 1.6% $\Delta k/k$ in specification 3.2(4) corresponds to a 6.9 millisecond period. Reactor core DU-12/25 of the SPERT-I series of tests had 12 plate fuel elements containing 168 grams of U-235 substantially similar to the CAVALIER fuel elements (Reference - Thompson and Beckerly, "Technology of Nuclear Reactor Safety," Volume I, page 683 (1964)). A 6.9 millisecond period was non-destructive to the SPERT reactor when shut down immediately following the excursion. See Chapter 9 of the CAVALIER SAR.

The boron addition capability of the ARIS provides additional assurance that the reactor can be shut down and maintained subcritical in the event of all four control rods failing to respond to a scram signal. See section 9.4.6 of the CAVALIER SAR.

3.3 Reactor Instrumentation

Applicability

This application applies to the instrumentation which must be operable for safe operation of the reactor.

Objective

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the measuring channels described in the following table are operable and the information is displayed on the control console.

Measuring Channel	Minimum No. Operable	Operating Mode in Which Required
Startup Count Rate	2	Reactor Startup
Linear Power (Gamma-Ion Chamber)	1	All Modes
Log N and Period (CIC)	1	All Modes
Tank Top Radiation Monitor	1	All Modes
Tank Water Level	1	All Modes

Bases

The neutron detectors, and gamma monitors, provide assurance that measurements of the reactor power level are adequately covered at both low and high power ranges. The reactor tank water level indicator provides early warning of the possibility of a leak in the Moderator Tank.

The radiation monitor provides information to operating personnel of a decrease in tank water level, or of high reactor power, or of any impending or existing danger from radiation, contamination, or streaming allowing ample time to take necessary precautions to initiate safety action.

3.4 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to stipulate the minimum number of reactor safety system channels that must be operable during normal operation.

Specification

The reactor shall not be operated unless the safety system channels described in the following table are operable:

Measuring Channel	Minimum No. Operable	Function	Operating Mode in Which Required to be Operable
Tank Water Level Monitor	1	Scram	All Modes
Tank Top Radiation Monitor	1	Scram	All Modes
Startup Count Rate	2	To prevent control rod withdrawal when both channels read <2 CPS	Reactor Startup
Manual Switch	1	Scram	All Modes
Reactor Power Level (CIC)	1	Scram	All Modes
Reactor Power Level (Gamma)	1	Scram	All Modes
Reactor Period (CIC)	1	Scram at less than 5 second period	All Modes
Reactor Period (Gamma)	1	Scram at less than 5 second period	All Modes

Bases

The startup interlock which requires a neutron count rate of at least 2 CPS on at least one startup count rate channel before the reactor is operated, assures that sufficient neutrons are available for proper operation of the startup channel. Power level scrams are provided to assure that the reactor power is maintained within the licensed limits. The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition arises. The period scrams are provided to assure that the power level does not increase on a period less than 5 seconds. One period scram specified is the power level channel using the compensated ion chamber and the other period scram utilizes a gamma sensitive chamber. Specifications on the tank water level scram are included as safety functions in the event of a serious loss of moderator tank water. The reactor would be shut down automatically in the event that a major leak occurs in the tank. The analysis in Section 9.2 of the SAR for CAVALIER shows the consequences resulting from loss of this water; and in this event the area could be evacuated without difficulty before significant doses are received by personnel.

The tank-top radiation monitor provides a scram and gives audible and visual warning in the event of a high radiation level in the reactor room resulting from failure of an experiment, from a significant drop in tank water level, or a higher than planned power level.

3.5 Limitations on Experiments

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

The following limits on experiments shall be met at all times:

(1) The reactivity worths of all experiments shall be in conformance with specifications in Section 3.2.

(2) Movable experiment must be worth less than 0.1% $\Delta k/k$.

(3) Experiments worth more than 0.1% $\Delta k/k$ must be inserted or removed with the reactor shutdown except as noted in item (4).

(4) Previously tried experiments with measured worth less than 0.4% $\Delta k/k$ may be inserted or removed with the reactor 2% or more subcritical.

(5) If any experiment worth more than 0.4% $\Delta k/k$ is to be inserted in the reactor, a procedure approved by the Reactor Safety Committee shall be followed.

(6) All materials to be irradiated in the reactor shall be either corrosion resistant or encapsulated within corrosion resistant containers.

(7) Irradiation containers to be used in the reactor in which a static pressure will exist or in which a pressure buildup is predicted shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2.

(8) Explosive material shall not be allowed in the reactor unless specifically approved by the Reactor Safety Committee. Experiments reviewed by the Reactor Safety Committee in which the material is potentially explosive, either while contained or if it leaks from the container, shall be designed to prevent damage to the reactor core or to the control rods or instrumentation, and to prevent any changes in reactivity.

(9) Experimental apparatus, material or equipment to be inserted in the reactor, shall not be positioned so as to cause shadowing of the nuclear instrumentation, interference with the control rods, or other perturbations that may interfere with the safe operation of the reactor.

Bases

The above specified limitations on experiments are based on the guidance given in Regulatory Guide 2.2 "Development of Technical Specifications for Experiments in Research Reactors" as developed in Section 6 of the CAVALIER SAR and concern conservative requirements for protecting the reactor from materials to be used in experiments. The reactivity of less than 0.1% Δk/k which can be inserted or removed with the reactor in operation in specification 3.5(2) can be compensated for by manual operation of a control rod.

3.6 Operation with Fueled Experiments

Applicability

This specification applies to the operation of the reactor with any fueled experiment.

Objective

To assure that the fission product inventory in fueled experiments are within the limits used in the safety analysis.

Specification

The reactor shall not be operated with fueled experiments unless the following conditions are satisfied.

(1) The thermal power (or fission rate) generated in the experiment is less than 1 watt (3.2×10^{10} fission/second).

(2) The total exposure of the experiment is not greater than the equivalent of 6 years continuous operation at 100 watts.

Basis

In the event of the failure of a fueled experiment, with the subsequent release of fission products (100% noble gas, 50% iodine, 1% solids), the 2 -hour inhalation exposures to iodine and strontium 90 isotopes at the facility exclusion distance, 70 meters, are less than the limits set by 10 CFR Part 20, using an averaging period of 1 year.

The analysis supporting this specification assumes 100% exfiltration of fission products from the reactor building in 2 hours. The safety analysis is identical with that in Section 5.4 of the UVAR Safety Analysis Report for isotopes released to the reactor building in general (other than in the UVAR reactor room). The CAVALIER is in the same building as the UVAR. The UVAR Safety Analysis Report is on record with the Commission: UVAR-18 (October, 1970), License NO. R-66, Docket No. 50-62. Due to the limits on reactivity worth of experiments in the CAVALIER, i.e. 0.5% $\Delta k/k$ for a single experiment it is highly unlikely that a 1 watt fueled experiment could ever be run, however this is considered an upper limit for the purposes of analysis.

3.7 Rod Drop Times

Applicability

This specification applies to the time from the initiation of a scram to the time a rod starts to drop (release time), and to the time it takes for a rod to drop from the fully withdrawn to the fully inserted position (free drop time).

Objective

To assure that the reactor can be shut down within a specified interval of time.

Specification

The reactor shall not be operated unless:

(1) The release time for each of the shim rods is less than 100 milliseconds, and

(2) The free drop time for each of the shim rods is less than 700 milliseconds.

Bases

Rod drop times as specified are sufficiently short to be consistent with the reactor period and neutron level scram settings to assure that the LSSS will not be exceeded in a short period transient as shown in Section 9.3 of the CAVALIER-SAR.

3.8 Alternative Reactivity Insertion System (ARIS)

Applicability

This specification applies to the elemental boron in solution in the ARIS tank and to the ARIS isolation valve.

Objective

To assure that the ARIS is capable of providing an alternative means of reactor shutdown during all reactor operations.

Specification

The reactor shall not be operated unless the following conditions exist:

(1) The volume of solution in the ARIS tank is greater than 24 gallons.

(2) The concentration of the boron is greater than 0.129 lb/gal of solution.

(3) The ARIS valve is unlocked.

Bases

The boron solution in the ARIS tank will normally be kept at a volume of 25 gal. and a concentration of 0.144 lb of boron per gallon of solution. The combination of 24 gal. with a concentration of 0.129 lb of boron per gallon of solution will yield a total negative reactivity addition of 3.2% $\Delta k/k$ when uniformly mixed with the water in the moderator tank. The requirement that the ARIS valve be unlocked before reactor startups will preclude unnecessary delay in the system initiation in case of need.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Shim Rods

Applicability

This specification applies to the surveillance requirements for the shim rods.

Objective

To assure that the shim rods are capable of performing their function and that no significant physical degradation in the rods has occurred.

Specification

(1) Shim rod drop times shall be measured semi-annually. Shim rod drop times shall also be measured if the control assembly is moved to a new position in the core or if maintenance is performed on the mechanism.

(2) The shim rod reactivity worths shall be measured whenever the rods are installed in a new core configuration.

Bases

The reactivity worth of the shim rods is measured to assure that the required shutdown margin is available and to provide means for determining the reactivity worths of experiments inserted in the core. The rod drop times are measured to assure that they meet the requirements of section 3.7 of these Technical Specifications.

4.2 Reactor Safety System

Applicability

This specification applies to the surveillance requirements for the safety system measuring channels and associated circuits of the reactor safety system.

Objective

The objective is to assure that the safety system is operable and capable of performing its intended function.

Specification

(1) A channel test of each of the reactor safety system channels shall be performed prior to each day's operation or prior to each operation extending more than one day.

(2) A channel check of each of the reactor safety channels shall be performed daily when the reactor is in operation.

(3) A channel calibration of the reactor safety channels shall be performed semi-annually.

Bases

The daily channel tests and channel checks will assure that the safety channels are operable. The semi-annual calibration will permit any long-term drift of the channels to be corrected.

4.3 Radiation Monitoring

Applicability

This specification applies to the radiation monitor required by Section 3.3 of these specifications.

Objective

The objective is to assure that the radiation monitor is operating and to verify the appropriate alarm setting.

Specification

The operation of the radiation monitor and the position of its associated alarm set point shall be verified daily during periods when the reactor is in operation. Calibration of the radiation monitoring equipment shall be performed semi-annually.

Bases

Surveillance of the monitor equipment will provide assurance that it is operable and that sufficient warning of a potential radiation hazard is available to permit corrective action before tolerances are exceeded.

4.4 Maintenance

Applicability

This specification applies to the surveillance requirements following maintenance of control or safety systems.

Objective

The objective is to assure that a system is operable before being used after maintenance has been performed.

Specification

Following maintenance or modification of a control or safety system component, it shall be verified that the system is operable prior to its return to service.

Bases

The intent of the specification is to assure that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected.

4.5 Alternative Reactivity Insertion System (ARIS)

Applicability

This specification applies to the alternative reactivity insertion system.

Objective

To assure that the ARIS is operable and can provide sufficient reactivity to put the reactor in a subcritical condition.

Specification

(1) Prior to each day's operation the volume of solution in the ARIS tank shall be verified, and the leak detection trap will be observed for signs of leakage.

(2) The concentration of boron in the solution shall be determined semiannually or after each make-up addition to the ARIS tank.

(3) A flow test from the ARIS tank to the flanged tee will be performed annually and the results compared to similar tests run at initial startup.

(4) The section of pipe from the flanged tee to the bottom of the moderator tank will be blown out with air annually.

Bases

The daily verification and observation will provide a means of detecting leakage from the ARIS into the moderator tank which could cause unexpected reactivity fluctuations in the system. The concentration of the boron in the solution is determined periodically to assure that the ARIS is capable of providing a negative reactivity addition of 3.2% $\Delta k/k$. The flow tests and air tests will demonstrate that the ARIS valve is operable and that the pipes are free of obstructions.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements used in the CAVALIER are the same as those considered in the Safety Analysis Report.

Specification

The fuel elements shall be of the materials testing reactor (MTR) type consisting of plates containing highly enriched uranium alloy fuel, clad with aluminum. There shall be 12 fuel plates containing nominally 165 grams of U-235 per element or 18 fuel plates containing nominally 195 grams of U-235 per element in the standard fuel elements. There shall be six fuel plates containing nominally 82.5 grams of U-235, per element or nine fuel plates containing nominally 98 grams of U-235, per element in the control rod fuel elements. Partially loaded fuel elements in which some of the fuel plates do not contain uranium may be used. An experimental element in which individual fuel plates can be removed or inserted may also be used. The mass of U-235 listed above refers to the initial (zero burnup) loading.

Various core configurations consisting of any combination of the above fuel elements may be used to accommodate experiments, but the loadings shall always be such that the minimum shutdown margin and excess reactivity as specified in Section 3.2 of these specifications are not exceeded.

Bases

These same type fuel elements have been run in the UVAR reactor at 2MW for many years and would create no safety problems for the CAVALIER. These specifications are consistent with the description of the fuel in the UVAR SAR.

5.2 Fuel Storage

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel which is being stored will not become supercritical and will not reach unsafe temperatures.

Specification

(1) All reactor fuel elements not in the reactor core shall be stored in a geometric array where k_{eff} is less than 0.9 for all conditions of moderation.

(2) Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device surface temperature will not exceed the boiling point of water.

Bases

Within these specifications, the fuel can be stored safely under all conditions. the UVAR storage facility was constructed to meet these specifications and will be used to store the CAVALIER elements.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The reactor facility shall be an integral part of the School of Engineering and Applied Science of the University of Virginia. The organizational structure of UVA relating to the reactor facility is shown in Figure 6.1. The Chairman, Department of Nuclear Engineering will have overall responsibility for management of the facility (Level 1).

6.1.2 Responsibility

The Reactor Facility Director shall be responsible for the overall facility operation (Level 2). During periods when the Reactor Facility Director is absent, his responsibilities are delegated to the Reactor Supervisor (Level 3).

The Reactor Facility Director shall have at least a Bachelor of Science or Engineering degree and have a minimum of 5 years of nuclear experience. A graduate degree may fulfill 4 years of experience on a one-for-one time basis.

The Reactor Supervisor shall be responsible for the day-to-day operation of the UVAR and CAVALIER and for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Reactor Safety Committee. During periods when the Reactor Supervisor is absent, his responsibilities are delegated to a person holding a Senior Reactor Operator license (Level 4).

The Reactor Supervisor shall have the equivalent of a Bachelor of Science or Engineering degree and have at least 2 years of experience in Reactor Operations at this facility, or an equivalent facility, or at least 6 years of experience in Reactor Operations. Equivalent education or experience may be substituted for a degree. Within nine months after being assigned to the position, the Reactor Supervisor shall obtain and maintain an NRC Senior Operator license.

6.1.3 Staffing

When the reactor is operating the following conditions will be met:

(1) A licensed Senior Reactor Operator or a licensed Reactor Operator shall be present at the reactor controls, however, a trainee may be

present at the controls if under the direct supervision of Senior Reactor Operator or Reactor Operator in the control room. (2) A licensed Senior Reactor Operator shall be on call, but not necessarily at the facility.

(3) At least one other person, not necessarily licensed to operate the reactor, shall be present at the facility.

(4) Rearrangements of the core or other nonroutine actions shall be supervised by a licensed Senior Reactor Operator.

(5) A health physicist who is organizationally independent of the Reactor Facility Operations groups, as shown in Figure 6.1, shall be responsible for radiological safety at the facility.

6.2 Review and Audit

There shall be a Reactor Safety Committee that shall review and audit reactor operations to ensure that the facility is operated in a manner consistent with public safety and within the terms of the facility license. The Reactor Safety Committee shall report to the President of the University and advise the Chairman, Department of Nuclear Engineering, and the Reactor Facility Director on those areas of responsibility specified below.

6.2.1 Composition and Qualification

The Committee shall be composed of at least five members, one of whom shall be the Radiation Safety Officer of the University. No more than two members will be from the organization responsible for Reactor Operations. The membership of the Committee shall be such as to maintain a degree of technical proficiency in areas relating to reactor operation and reactor safety.

6.2.2 Charter and Rules

(1) A quorum of the Committee shall consist of not less than a majority of the full committee and shall include the Chairman or his designee.

(2) The Committee shall meet at least semiannually and shall be on call by the Chairman. Minutes of all meetings shall be disseminated to responsible personnel as designated by the Committee Chairman.

(3) The Committee shall have a written statement defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Committee.

6.2.3 Review Function

As a minimum the responsibilities of the Reactor Safety Committee include:

(1) review and approval of untried experiments and tests that are significantly different from those previously used or tested in the reactor, as determined by the Facility Director.

(2) review and approval of changes to the reactor core, reactor systems or design feature that may affect the safety of the reactor.

(3) review and approve all proposed amendments to the facility license, Technical Specifications, and changes to the standard operating procedures (discussed in Section 6.3 of these specifications).

(4) review reportable occurrences and the actions taken to identify and correct the cause of the occurrences.

(5) review significant operating abnormalities or deviations from normal performance of facility equipment that affect reactor safety.

(6) review reactor operation and audit the operational records for compliance with reactor procedures, Technical Specifications, and license provisions at least every two years.

6.3 Operating Procedures

Written procedures, reviewed and approved by the Reactor Safety Committee shall be in effect and followed for the items listed below. These procedures shall be adequate to ensure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

(1) startup, operation, and shutdown of the reactor.

(2) installation or removal of fuel elements, control rods, experiments, and experimental facilities.

(3) actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected system leaks and abnormal reactivity changes.

(4) emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.

(5) preventive and corrective maintenance operations that could have an effect on reactor safety.

(6) periodic surveillance (including test and calibration) of reactor instrumentation and safety systems.

Radiation control procedures shall be maintained and made available to all operations personnel.

Substantive changes to the approved procedures shall be made only with the approval of the Reactor Safety Committee. Changes that do not change the original intent of the procedures may be made with the approval of the Facility Director. All such minor changes to procedures shall be documented and subsequently reviewed by the Reactor Safety Committee.

6.4 Required Actions

6.4.1 Action To Be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is violated, the following actions shall be taken;

(1) The reactor shall be shut down and reactor operations shall not be resumed until authorized by the Commission.

(2) The occurrence shall be reported to the Reactor Facility Director and the Chairman of the Reactor Safety Committee, or their designee, as soon as possible, but not later than the next work day. Reports shall be made to the Commission in accordance with Section 6.6 of these specifications.

(3) A written safety limit violation report shall be made that shall include an analysis of the causes of the violation and extent of resulting damage to facility components, systems, or structures; corrective actions taken; and recommendations for measures to preclude reoccurrence. This report shall be submitted to the Reactor Safety Committee for review.

6.4.2 Action To Be Taken in the Event of a Reportable Occurrence

A reportable occurrence is any of the following conditions:

(1) any safety system setting less conservative than specified in Section 2.2 of these specifications.

(2) operating in violation of an LCO established in these specifications, unless prompt remedial action is taken.

(3) safety system component malfunctions or other component or system malfunctions during reactor operation that could, or threaten to, render the safety system incapable of performing its intended safety function, unless immediate shutdown of the reactor is initiated.

(4) an uncontrolled or unanticipated increase in reactivity in excess of 0.5% $\Delta k/k$.

(5) an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor.

(6) abnormal and significant degradation in reactor fuel, and/or cladding, coolant boundary, or containment boundary (excluding minor leaks) where applicable that could result in exceeding prescribed radiation-exposure limits of personnel and/or environment.

In the event of a reportable occurrence, the following action shall be taken:

(1) The Director of the Reactor Facility shall be notified as soon as possible and corrective action shall be taken before resuming the operation involved.

(2) A written report of the occurrence shall be made which shall include an analysis of the cause of the occurrence, the corrective action taken, and recommendations for measures to preclude or reduce the probability of reoccurrence. This report shall be submitted to the Director and the Reactor Safety Committee for review.

(3) A report shall be submitted to the Nuclear Regulatory Commission in accordance with Section 6.6 of these specifications.

6.5 Plant Operating Records

In addition to the requirements of applicable regulations, records (or logs) of the items listed below shall be kept in a manner convenient for review and shall be retained as indicated.

6.5.1 Records To Be Retained for a Period of at Least Five Years

(1) normal plant operation

(2) principal maintenance activities

(3) experiments performed with the reactor

(4) reportable occurrences

(5) equipment and component surveillance activity

(6) facility radiation and contamination surveys

(7) transfer of radioactive material

(8) changes to operating procedures

6.5.2 Records To Be Retained for the Life of the Facility

(1) gaseous and liquid radioactive effluents released to the environs

(2) offsite environmental monitoring surveys

(3) fuel inventories and transfers

(4) radiation exposures for all personnel

(5) changes to reactor systems, components, or equipment that may affect reactor safety

(6) updated and corrected drawings of the facility

(7) minutes of Reactor Safety Committee meetings

6.6 Reporting Requirements

In addition to the requirements of applicable regulations (such as described in Regulatory Guide 10.1 "Compilation of Reporting Requirements for Persons Subject to NRC Regulations" and NUREG-1022, "Licensee Event Report System"), reports should be made to the U.S. Nuclear Regulatory Commission as follows:

6.6.1 Special Reports

(1) A report as soon as possible, but no later than the next working day, to the Office of Regional Administrator, N.R.C. Region II, 101 Marietta Street, N.W. Atlanta, Ga. 30323:

(a) any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or exposure

(b) Any reportable occurrences as defined in Section 6.4.2 of these specifications

(c) any violation of a safety limit

(2) A report within 14 days in writing to the Director of the Office of Nuclear Reactor Regulation, U.S.N.R.C. Washington, D.C. 20555 ATTN: Document Control Desk with a copy to the Office of Regional Administrator, NRC Region II, 101 Marietta Street, N.W., Atlanta, Ga. 30323:

(a) any accidental offsite release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or exposure

(b) any reportable occurrence as defined in Section 6.4.2 of these specifications

(c) any violation of [illegible]

(3) A report within 30 da[illegible] to the Director of the Office of Nuclear Reactor Regulation [illegible] D.C. 20555 ATTN: Document Control Desk, 101 Mariet[illegible], [illegible] a copy to the Office of Regional Administrator, NRC, Region [illegible] Marietta Street N.W. Atlanta, Ga. 30323:

(a) any substantial variance from performance specifications contained in these specifications or in the SAR

(b) any significant change in the transient or accident analyses as described in the SAR

(c) changes in personnel serving as Chairman of the Department of Nuclear Engineering, Reactor Facility Director, or Reactor Supervisor

(4) A report within nine months after initial criticality of the reactor or within 90 days of completion of the startup test programs, whichever is earlier, to the Director, Office of Nuclear Reactor Regulation, US NRC, Washington, D.C. 20555 ATTN: Document Control Desk, upon receipt of a new facility license, an amendment to the license authorizing an increase in power level or the installation of a new core of a different design than previously used. The report will include the measured values of the operating conditions or characteristics of the reactor under the new conditions, including

(a) total control rod reactivity worth

(b) reactivity worth of the single control rod of highest reactivity worth

(c) minimum shutdown margin both at ambient and operating temperatures

6.6.2 Routine Reports

A routine report will be made by March 31 of each year to the Director, Office of Nuclear Reactor Regulation, US NRC, Washington, D.C. 20555, ATTN: Document Control Desk, with a copy to the Office of Regional Administrator, NRC, Region II, 101 Marietta Street, N.W., Atlanta, Ga. 30323 providing the following information:

(1) A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics, and operating procedures related to the reactor safety occurring during the reporting period.

(2) A tabulation showing the energy generated by the reactor (in watt hours) and the number of hours the reactor was critical each quarter during the year.

(3) A report of the results of the safety-related maintenance and inspections. The reasons for corrective maintenance of safety-related items will be included.

(4) A report of the number of emergency shutdowns and inadvertent scrams, including their reasons and the corrective actions taken.

(5) A summary of changes to the facility or procedures, which affect reactor safety, and performance of tests or experiments carried out under the conditions of Section 50.59 of 10 CFR 50.

(6) A summary of the nature and amount of radioactive gaseous, liquid and solid effluents released or dischanged to the environs beyond the effective control of the licensee as measured or calculated at or prior to the point of such release or discharge.

(7) A description of any environmental surveys performed outside the facility.

(8) A summary of radiation exposures received by facility personnel and visitors, including the dates and time of significant exposures (greater than 500 mrem for adults and 50 mrem for persons under 18 years of age) and a summary of the results of radiation and contamination surveys performed within the facility.

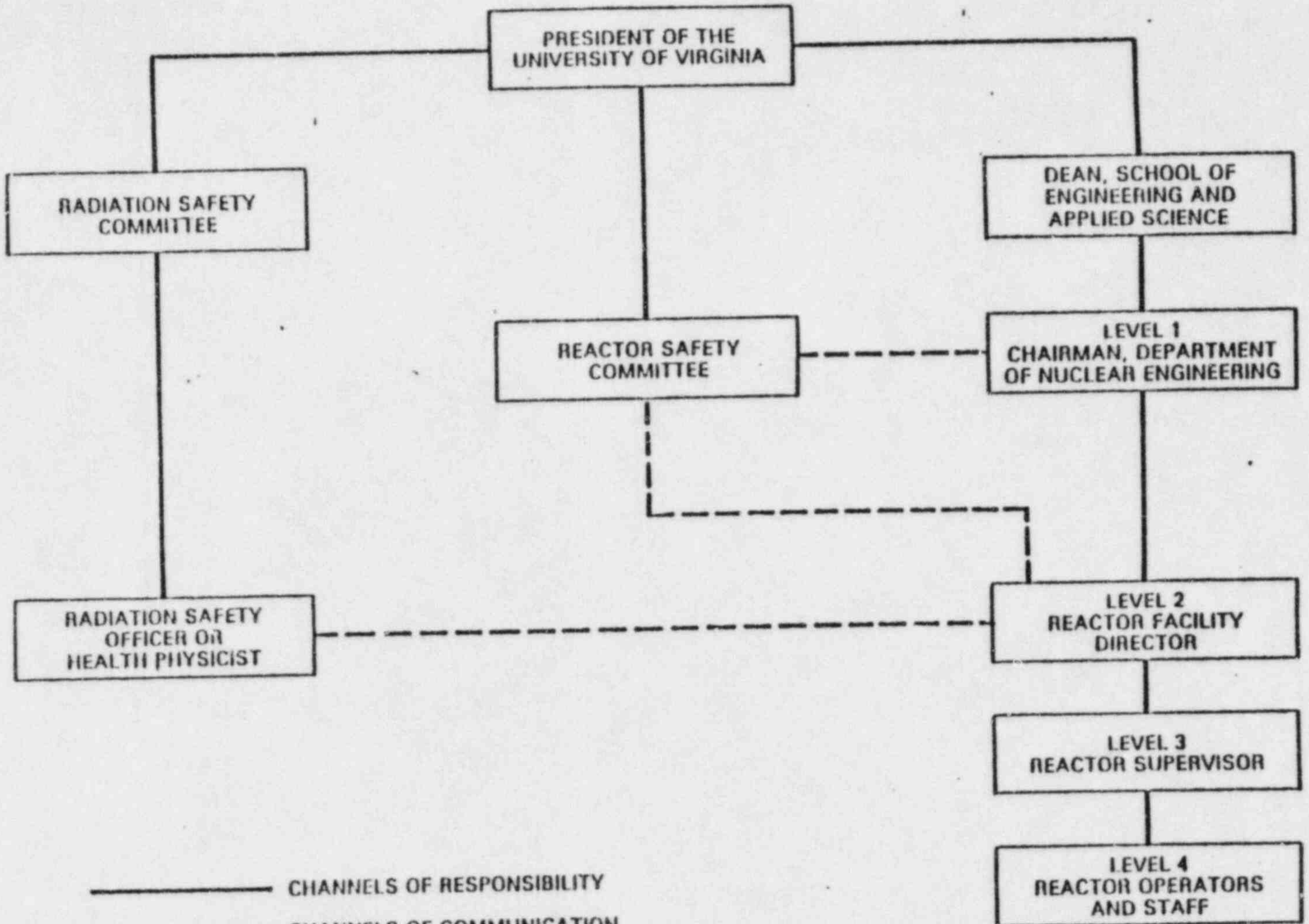


Figure 6.1 Organizational structure of UVA relating to reactor facility