UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

UNIVERSITY OF VIRGINIA

DOCKET NO. 50-62

AMENDED FACILITY OPERATING LICENSE

Amendment No. 15
License No. R-66

I. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by the University of Virginia (the licensee) dated March 9, 1977, as supplemented by filings dated December 18, 1978, January 19, 1979, September 18, 1979, July 15, 1980, February 12, 1981, August 19, 1981, March 11, 1982, March 19, 1982, May 18, 1982, June 7, 1982 and August 27, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission;

D. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

E. The licensee is a nonprofit educational institution and will use the facility for the conduct of educational activities, and has satisfied the applicable provisions of 10 CFR 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

F. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public and does not involve a significant hazards consideration;

G. The issuance of this amendment is in accordance with 10 CFR 51 of the Commission's regulations and all applicable requirements have been satisfied; and

H. The receipt, possession and use of the byproduct and special nuclear material as authorized by this license, will be in accordance with the Commission's regulations in 10 CFR 30 and 70, including sections 30.33, 70.23 and 70.31.

II. Facility Operating License No. R-66 is hereby amended in its entirety to read as follows:

A. This license applies to the light water-cooled and -moderated swimming pool nuclear reactor owned by the University of Virginia (the licensee), located on the campus of the University of Virginia at Charlottesville, Albemarle County, Virginia and described in the application for license renewal.

B. Subject to the conditions and requirements incorporated herein, the Commission, hereby, licenses the University of Virginia:

(1) Pursuant to Section 104c of the Act and 10 CFR 50, "Licensing of Production and Utilization Facilities," to possess and operate the reactor as a utilization facility at the designated location near Charlottesville, Virginia, in accordance with the procedures and limitations described in the application and in this license.

(2) Pursuant to the Act and 10 CFR 70, "Special Nuclear Material," to receive, possess and use up to 14.0 kilograms of contained uranium 235 and 16 grams of plutonium in a Pu-Be source for use in connection with operation of the reactor.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material" to receive, possess, store and use in the reactor pool 70,000 curies of cobalt 60; to receive, possess and use 1.0 gram of neptunium 237; and to possess, but not separate, such byproduct materials as may be produced by operation of the reactor.

C. This license shall be deemed to contain, and be subject to, the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now, or hereinafter in effect, and is subject to the additional conditions specified below:

(1) Maximum Power Level

The University of Virginia is authorized to operate the reactor at steady state power levels up to a maximum of 2 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 15, are hereby incorporated in the license. The University of Virginia shall operate the facility in accordance with the Technical Specifications.

(3) Physical Security Plan

The licensee shall maintain and fully implement all provisions of the Commission's approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.790, entitled "University of Virginia Nuclear Reactor Facility Physical Security Plan (July 1980)," submitted by letter dated July 15, 1980, as revised by letters dated February 26, 1981 and July 29, 1981.

This license is effective as of the date of issuance and shall expire at midnight twenty years from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Acting Chief
Standardization & Special
Projects Branch
Division of Licensing

Enclosure:
Appendix A - Technical
Specifications, September 30, 1982

Date of Issuance:

APPENDIX A

FACILITY LICENSE NO. R-66

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF VIRGINIA

DOCKET NO. 50-62

DATE: SEPTEMBER 30, 1982

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

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.

Beam Ports: The beam ports are the two 8-in. neutron beam ports that penetrate the shield on the south side of the pool.

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 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 Co.

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

Large Access Facilities: The large access facilities are the two large openings approximately 5 ft wide by 6 ft high that penetrate the shield on the south side of the pool.

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 min).

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 an average pool temperature of ($<90^\circ$ F) with the effect of xenon poisoning on 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: Reactor operation is 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 incore 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.

Regulating Rod: The regulating rod is a control rod of low reactivity worth fabricated from stainless steel and used to control reactor power. The rod may be controlled by the operator with a manual switch or by an automatic controller.

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 substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment or by forces that can arise as a result of credible malfunctions.

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 a safety rod when the magnet is de-energized.

Survelliance Time Intervals:

Annually (interval not to exceed 15 months)
Semiannually (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 Limits

2.1.1 Safety Limits in Forced Convection Mode of Operation

Applicability: This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the forced convection flow mode of operation. These variables are

P = reactor thermal power
W = reactor coolant flow rate
T_I = reactor coolant inlet temperature
L = height of water above the core

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Specifications: In forced convection flow mode of operation

(1) The pool water level shall not be less than 19 ft above top of the core.

(2) The reactor coolant inlet temperature shall not be greater than 111° F.

(3) The combination of true values of P and W shall be in the unshaded portion as shown in Figure 2.1.

Basis: Above 400 gpm in the region of full power operation, the criterion used to establish the safety limit was a burnout ratio of 1.49 including the worst variations in the manufacturer's tolerances and specification, hot channel factors, and other appropriate uncertainties. The analysis is given in Section 9.4 of the Safety Analysis Report (SAR).

In the region below 400 gpm where the flow coasts down to zero, reverses, and natural convection cooling is established, the criterion for selecting a safety limit is taken as a fuel plate temperature. The analysis of a loss of flow transient from 3.45 MW of power and 744 gpm of flow shows that the maximum fuel plate temperature reached is 303° F, which is well below the temperature at which fuel clad damage could occur. The analysis is given in Section 9.7 of the SAR.

2.1.2 Safety Limits in Natural Convection Mode of Operation

Applicability: This specification applies to the interrelated variables associated with core thermal and hydraulic performance in the natural convection flow mode of operation. These variables are

P = Reactor Thermal Power
T_I = Reactor Coolant Inlet Temperature

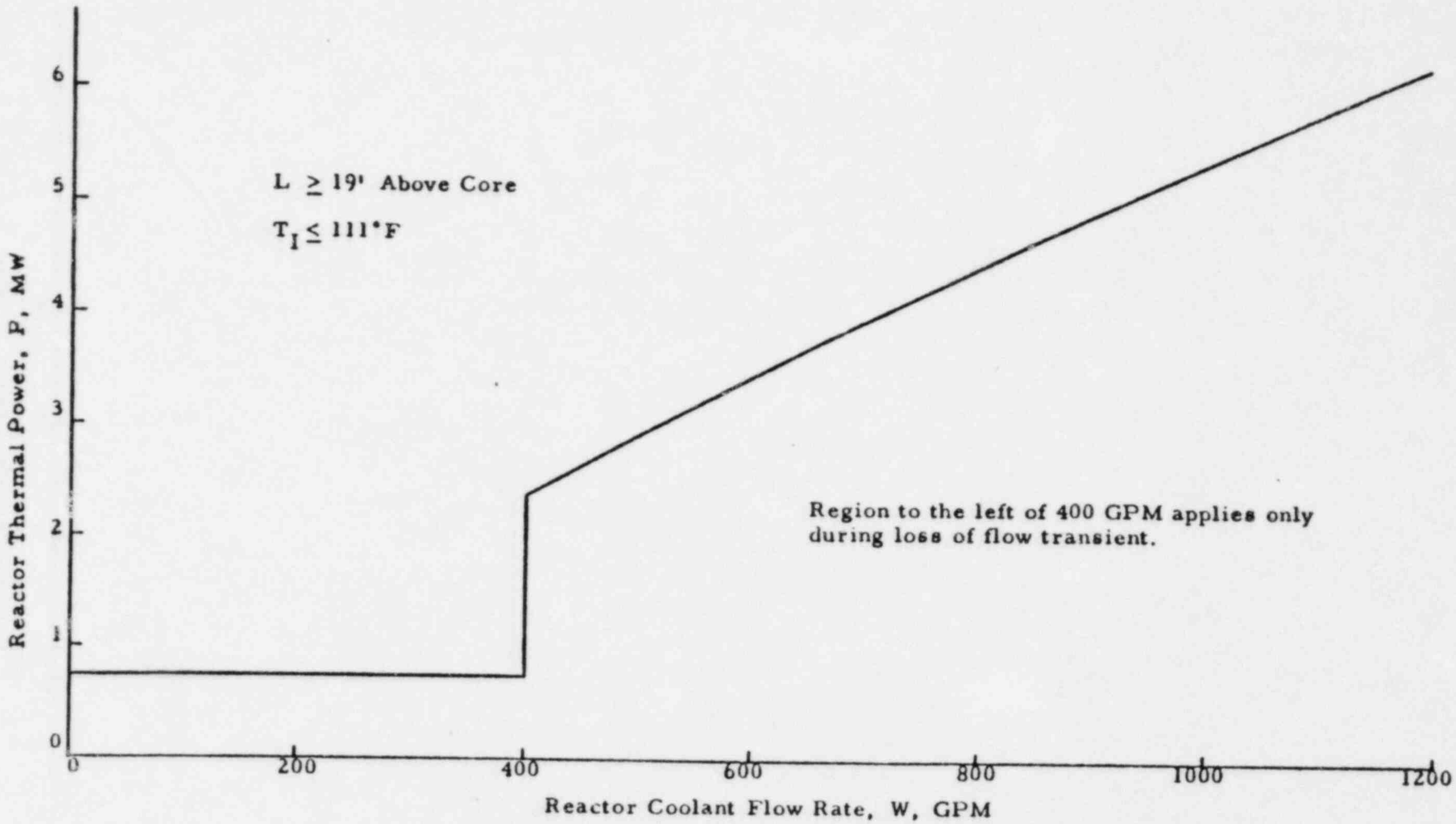

Figure 2.1 Safety limits with forced convection flow

Objective: The objective is to ensure that the integrity of the fuel clad is maintained.

Specification: In the natural convection flow mode of operation, the true value of P and T_I shall not exceed

P 750 kWt
T_I 111° F

Bases: The criterion for establishing a safety limit with natural convection flow is established as a fuel plate temperature. This is consistent with Figure 2.1 for forced convection flow during a transient. The analysis for natural convection flow shows that at 750 kW, the maximum fuel plate temperature is 259° F, which is well below the temperature at which fuel clad damage could occur. The flow rate with natural convection at this power is calculated to be 129 gpm. The analysis is given in Chapter X of Amendment 1 to the SAR (UVAR-18, Part I).

2.2 Limiting Safety System Settings

Applicability: This specification applies to the set points for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), reactor coolant inlet temperature (T_I), and the height of water above the core (L).

Objective: The objective is to ensure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specifications:

(1) For operation in the forced convection mode, the limiting safety system settings shall be

P 3.0 MWt(max)
W 800 gpm (min)
T_I 108° F (max)
L 19 ft 2 in. (min)

(2) For operation in the natural convection mode, the limiting safety system setting shall be

P 300 KWt (max)
T_I 108° F (max)

Bases: The analysis shows that there is sufficient margin between these settings and the safety limits under the most adverse conditions of operation (Section 9.5 of the SAR). With natural convection flow, there is no minimum coolant flow rate and no minimum height of water above the core so long as there is a path for flow (see Section 3.8 of these specifications).

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity

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

Objectives: The objectives are to ensure that the reactor can be shut down at all times and that the safety limit will not be exceeded.

Specifications: The reactor shall not be operated at powers in excess of 1 kW unless the following conditions exist:

(1) The minimum shutdown margin provided by control rods, with secured experiments (see Section 1.0) 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 0.45% $\Delta k/k$ must be a secured experiment.

(3) The total reactivity worth of the two experiments having the highest reactivity worth is less than 1.6% $\Delta k/k$.

(4) The total reactivity worth of all experiments is less than 2.0% $\Delta k/k$.

(5) The maximum excess reactivity with fixed experiments in place and referred to cold, xenon-free condition shall be limited to 5% $\Delta k/k$.

Bases: The shutdown margin required by Specification 3.1(1) is necessary so that the reactor can be shut down from any operating condition and remain shut down after cooldown and xenon decay, even if one control rod should stick in the fully withdrawn position.

The reactivity of 0.45% $\Delta k/k$ in Specification 3.1(2) corresponds to a 3-sec period. An analysis that shows the peak power does not exceed the safety limit, when the reactor power level is increasing on a 3-sec period, as the true value of the limiting safety system setting (LSSS) is reached is given in Section 9.6 of the SAR and in Chapter XI of Amendment 1 to the SAR (UVAR-18, Part I).

The reactivity of 1.6% $\Delta k/k$ in Specification 3.1(3) corresponds to a 6.9-msec period. Reactor Core DU-12/25 of the SPERT-1 series of tests had 12-plate fuel elements containing 168 grams of U-235 substantially similar to the UVAR fuel elements (Reference: Thompson and Beckerly, "Technology of Nuclear Reactor Safety," Volume I, page 683 (1964)). A 6.9-msec period was nondestructive. The simultaneous failure of more than two experiments is considered unlikely.

The total reactivity of 2.0% $\Delta k/k$ in Specification 3.1(4) places a reasonable upper limit on the worth of all experiments.

Operation of the reactor at a power of less than 1 kW is allowed to measure the reactivity worth of untried experiments, in accordance with procedures approved by the Reactor Safety Committee, and to measure the excess reactivity of new core loadings.

The limit of 5% $\Delta k/k$ on excess reactivity is to allow for xenon override and operational flexibility and to ensure that the operational reactor is reasonably similar in configuration to the reactor core analyzed in the SAR. In general the excess reactivity is limited by the shutdown margin requirement.

3.2 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 to ensure that the safety limit is not exceeded during normal operation.

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

Bases: The startup interlock, which requires a neutron count rate of at least 2 counts per second (CPS) before the reactor is operated, ensures that sufficient neutrons are available for proper operation of the startup channel.

The pool-water temperature scram provides protection to ensure that if the limiting safety system setting is exceeded an immediate shutdown will occur to keep the fuel temperature below the safety limit. Power level scrams are provided to ensure that the reactor power is maintained within the licensed limits and to protect against abnormally high fuel temperatures. The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition arises. The period scram is provided to ensure that the power level does not increase on a period less than 3 sec. This ensures that a safety limit will not be exceeded as described in Chapter XI of Amendment 1 to the SAR (UVAR-18, Part I).

Specifications on the pool-water level are included as safety measures in the event of a serious loss of primary system water. Reactor operations are terminated if a major leak occurs in the primary system. The analysis in Section 9.8 of the SAR shows the consequences resulting from loss of coolant.

The bridge radiation monitor gives warning of a high radiation level in the reactor room from failure of an experiment or from a significant drop in pool-water level.

A scram from either loss of primary coolant flow or loss of power to the pump protects the reactor from overheating.

3.3 Reactor Instrumentation

Applicability: This specification applies to the instrumentation that must be operable for safe operation of the reactor.

Table 3.1 Safety system channels

Measuring Channel	Minimum No. Operable	Set Point*	Function	Operating Mode Required
Pool water level monitor	2	19 ft 2 in. (min)	Scram	Forced convection mode
Bridge radiation monitor	1		Scram	All modes
Pool water temperature	1	108° F (max)	Scram	All modes
Power to primary coolant pump	1	Loss of power	Scram	Forced convection mode
		Application of power	Scram	Natural convection mode
Primary coolant flow	1	800 gpm (min)	Scram	Forced convection mode
Startup count rate	1	2 CPS (min)	Prevents withdrawal of any shim rod	Reactor startup
Manual button	1		Scram	All modes
Reactor power level	2	3 MWt(max)	Scram	Forced convection mode
		0.3 MWt(max)		Natural convection mode
Reactor period	1	3 sec (min)	Scram	All modes
Air pressure to header	1		Scram	All modes

*Values listed are limiting set points. For operational convenience, set points may be changed to more conservative values.

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

Specification: The reactor shall not be operated unless the measuring channels described in Section 3.2 "Reactor Safety Systems" and in the following table are operable.

Bases: The neutron detectors provide assurance that measurements of the reactor power level are adequately covered at both low and high power ranges.

Table 3.2 Measuring channels

Measuring Channel	Minimum No. Operable	Operating Mode in Which Required
Linear power	1	All modes
Intermediate (Log N) and period	1	All modes
Core gamma monitor*	1	Forced convection
Reactor room constant* air monitor	1	All modes*
Bridge radiation monitor	1	All modes
Reactor face monitor*	1	All modes*
Pool-water level monitor	2	Forced convection mode
Pool-water temperature	1	All modes
Primary coolant flow	1	Forced convection mode
Startup count rate	1	Reactor startup
Reactor power level	2	All modes

*The reactor room constant air monitor, reactor face monitor, and core gamma monitor may be out of service for a period not to exceed 7 days without requiring reactor shutdown. If the reactor face monitor cannot be repaired within 7 days, it may be replaced by a locally alarming monitor of similar range for up to 30 days without requiring a reactor shutdown.

The radiation monitors provide information to operating personnel of a decrease in pool-water level and of any impending or existing danger from radiation contamination or streaming, allowing ample time to take necessary precautions to initiate safety action.

The reactor room constant air monitor and reactor face monitor provide redundant measures of abnormal high radiation levels. Because the other measuring channels for determining the radiation levels are required for reactor operation, the reactor can be operated safely if the monitors are not functioning for short periods of time.

3.4 Radioactive Effluents

Applicability: This specification applies to the monitoring of radioactive effluents from the reactor facility. Airborne and liquid effluents are discussed separately in the following sections.

3.4.1 Airborne Effluents

Objective: The objective is to ensure that exposure to the public resulting from the release of Ar-41 and other airborne effluents will be well below the limits of 10 CFR 20 for unrestricted areas.

Specification: When either of the neutron beam ports are drained, the centrifugal blower that exhausts that area shall be in operation and the airborne activity in the effluent shall be monitored by an instrument located in the 6-in. exhaust duct.

Basis: The basis for this specification is given by the analysis in Chapter IX of Amendment 1 to the SAR (UVAR-18, Part I).

3.4.2 Liquid Effluents

Objective: The objective is to ensure that exposure to the public resulting from the release of radioactive effluents will be well below the limits of 10 CFR 20 for unrestricted areas.

Specification: The activity of liquids released beyond the site boundary shall not exceed 10 CFR 20 limits.

Basis: The basis for this specification is given in Section 4.8 in the SAR.

3.5 Confinement

Applicability: This specification applies to the capability of isolating the reactor room, when necessary.

Objective: The objective is to prevent the exposure to the public resulting from airborne activity released into the reactor room from exceeding the limits of 10 CFR 20 for unrestricted areas.

Specification: The reactor shall not be operated unless the following equipment is operable:

Equipment	Function
Truck door closed switch	Scram reactor when truck door is not fully closed
Ventilation exhaust duct doors	Close and seal when Bridge Radiation Monitor alarms
Personnel door	Close and seal when Bridge Radiation Monitor Alarms
Emergency exit manhole water level	Water level is high enough to form a water seal at least 6 in. in depth.

Bases: The bases for the proper operation of these items of equipment are given in Section 6.1 of the SAR.

3.6 Limitation 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 reactor shall not be operated unless the following conditions exist:

(1) The reactivity worths of all experiments shall be in conformance with specifications in Section 3.1.

(2) Movable experiments 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 shut down 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 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) Cooling shall be provided to prevent the surface temperature of an experiment to be irradiated from exceeding the boiling point of the reactor pool water.

(10) 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 limitations on experiments specified in Items 1-10 are based on the irradiation program authorized by Amendment No. 3 to License No. R-66 dated August 13, 1962. The reactivity of less than 0.1% that can be inserted or removed with the reactor in operation is to accommodate experiments in the hydraulic rabbit.

3.7 Operation with Fueled Experiments

Applicability: This specification applies to the operation of the reactor with any fueled experiment within the reactor building.

Objective: The objective is to ensure that the confinement leak rate and fission product inventory in fueled experiments are within limits used in the safety analysis.

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

(1) For fueled experiments in which the thermal power generated is greater than 1 W

(a) The experiment must be in the reactor pool and under at least 15 ft of water.

(b) The thermal power (or fission rate) generated in the experiment is not greater than 100 W (3.2×10^{12} fissions/sec).

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

(d) The leak rate from the reactor room is not greater than 50% of containment volume in 20 hours as measured within the previous 12 months.

(2) For fueled experiments in which the thermal power generated is less than 1 W (3.2×10^{10} fissions/sec)

(a) The experiment may be located anywhere in the reactor building.

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

Bases: 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 20, using an averaging period of 1 year.

The safety analyses for which results are used here are found in SAR Section 5.4. The analysis supporting Specification 3.7(2) assumes 100% exfiltration of fission products from the reactor building in 2 hours. The analysis supporting Specification 3.7(1) for the fueled experiments within the reactor pool assumes a fission product retention in the reactor room equivalent to 100% fission product exfiltration in 20 hours. The specification provides suitable allowance for degradation between tests. The measurement of the exfiltration value is described in Chapter XII of Amendment 1 to the SAR (UVAR-18, Part I).

3.8 Height of Water Above the Core in Natural Convection Mode of Operation

Applicability: This specification applies to the height of water above the reactor core when the reactor is operating with natural convection cooling.

Objective: The objective is to ensure that there is a continuous path for circulation of water when the reactor is operated in the natural convection mode.

Specification: The reactor shall not be operated in the natural convection mode unless there is at least 1 ft of water above the core.

Bases: One ft of water above the core is sufficient to provide a continuous path for natural convection cooling. For other than zero power operation, the radiation levels may require a greater depth for shielding, in which case, the regulations in 10 CFR 20 will govern.

3.9 Rod Drop Times

Applicability: This specification applies to the time from the initiation of a scram to the time a rod starts to drop (magnet release time) as well as to the time it takes for a rod to drop from the fully withdrawn to the fully inserted position (free-drop time).

Objective: The objective is ensure that the reactor can be shut down within a specified period of time.

Specification: The reactor will not be operated unless (1) the magnet release time for each of the three shim rods is less than 50 msec and (2) the free-drop time for each of the three shim rods is less than 700 msec.

Bases: Rod drop times as specified will ensure that the safety limits will not be exceeded in a short period transient. The analysis is given in Section 9.6 of the SAR and Chapter XI of Amendment 1 (UVAR-18, Part I).

3.10 Emergency Removal of Decay Heat

Applicability: This specification applies to the emergency removal of decay heat.

Objective: The objective is to ensure that the flow rate from this system is sufficient to prevent overheating of the fuel elements subsequent to a total loss of primary water from the core.

Specification: There shall be two separate emergency core spray systems, each capable of maintaining a flow rate of at least 10 gpm over the 64 fuel element positions for the first 30 min, and at least 7½ gpm over the 64 fuel element positions for the next 60 min following a total loss of coolant.

Bases: Either of the two spray systems, as specified, will provide sufficient cooling to maintain the fuel temperature below its melting point as demonstrated by the evaluation in Section 9.9 of the SAR.

3.11 Primary Coolant Condition

Applicability: This specification applies to the quality of the primary coolant in contact with the fuel cladding.

Objectives: The objectives are (1) to minimize the possibility for corrosion of the cladding on the fuel elements and (2) to minimize neutron activation of dissolved materials.

Specifications:

(1) Conductivity of the pool water shall be no higher than 5×10^{-6} mhos/cm.

(2) The pH of the pool water shall be between 5.0 and 7.5.

Bases: A small rate of corrosion continuously occurs in a water-metal system. To limit this rate, and thereby extend the longevity and integrity of the fuel cladding, a water cleanup system is required. Experience with water quality control at many reactor facilities has shown that maintenance within the specified limits provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the as low as is reasonably achievable (ALARA) principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operations.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Shim Rods

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

Objectives: The objectives are to ensure that the shim rods are capable of performing their function and to establish that no significant physical degradation in the rods has occurred.

Specifications:

(1) Shim rod drop times shall be measured semiannualy. 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.

(3) The shim rods shall be visually inspected annually and when rod drop times exceed the limiting conditions for operation (Section 3.9 of these specifications).

Bases: The reactivity worth of the shim rods is measured to assure that the required shutdown margin is avaiable and to provide means for determining the reactivity worth of experiments inserted in the core. The visual inspection of the shim rods and measurement of their drop times are made to determine whether the shim rods are capable of performing properly.

4.2 Reactor Safety System

Applicability: This specification applies to the surveillance requirements for the reactor safety system of the reactor.

Objective: The objective is to ensure that the reactor safety system is operable as required by Specification 3.2.

Specifications:

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

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

(3) A channel calibration of the reactor safety measuring channels shall be performed semiannually.

(4) The power range channels 1 and 2 shall be checked against a primary system heat balance at least once each week the reactor is in operation above 100 kW in the forced convection mode.

(5) The following items, which are listed in Table 3.1, are not considered to be reactor safety measuring channels: power to primary coolant pump, manual button, header air pressure, and pool water level monitor. Operation of these systems will be checked before each day's operation or before each operation extending more than one day.

Bases: The daily channel tests and channel checks will ensure that the safety channels are operable. The semiannual calibration will permit any long-term drift of the channels to be corrected. The weekly calibration of the power measuring channels will correct for drift and ensure operation within the requirements of the license.

4.3 Emergency Core Spray System

Applicability: This specification applies to the emergency core spray system.

Objective: The objective is to ensure that the spray systems are operable and will deliver the specified flow rate of emergency coolant.

Specifications:

(1) Whenever the reactor bridge is moved and replaced into position for forced convection operation, the remote coupler for each spray system shall be air-pressure checked to ensure that there is no leakage.

(2) Measurements will be made annually to verify that each spray system will deliver at least 10 gpm for 30 min.

Bases: The emergency spray system is an engineered safeguard. At the initial installation, each of the two core spray systems was checked to ensure that it delivered the flow as specified in Section 3.10 of these specifications. Because there are no moving parts and no automatic electronic or mechanical mechanisms subject to failure, a verification that the remote couplers are engaged and not leaking will ensure that the two core spray systems are operable. The annual measurement of the flow rate will verify that each of the two core spray systems will deliver the flow as desired. The preoperational test of the core spray system demonstrated that water delivery is at least 10 gpm for 30 min and 7½ gpm for the next 60 min. Subsequent annual tests, which verify the 30 min flow rate, are adequate to verify design performance. The core spray system is described in Section 4.10 and the safety analysis is given in Section 9.9 of the SAR. The annual measurement of the flow rate is described in Chapter IV of Amendment 1 to the SAR (UVAR-18, Part I).

4.4 Area Radiation Monitoring Equipment

Applicability: This specification applies to the area radiation monitoring equipment required by Sections 3.2 and 3.3 of these specifications.

Objectives: The objective are to ensure that the radiation monitoring equipment is operating and to verify appropriate alarm settings.

Specification: The operation of the radiation monitoring equipment and the position of their associated alarm set points shall be verified daily during periods when the reactor is in operation. Calibration of the radiation monitoring equipment shall be performed semiannually.

Bases: Surveillance of the monitoring equipment will provide assurance that sufficient warning of a potential radiation hazard is available.

4.5 Maintenance

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

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

Specification: Following maintenance or modification of a control or safety system or component, it shall be verified that the system is operable before it is returned to service or during its initial operation.

Bases: The intent of the specification is to ensure that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected. Correct operation of some systems, such as power range monitors, cannot be verified unless the reactor is operating. Operation of these systems will be verified during their initial operation following maintenance or modification.

4.6 Confinement

Applicability: This specification applies to the surveillance requirements for confinement of the reactor room.

Objective: The objective is to ensure that the closure equipment to the reactor room is operable.

Specifications

(1) Before each day's operation or before each operation extending more than one day, the water level in the emergency exit manhole shall be verified.

(2) At least once each month, a test shall be made to ensure that the following equipment is operable:

truck door closed switch
ventilation exhaust duct door
personnel door

(3) Semiannually, a visual inspection of the seal and gaskets of the truck door, the personnel door, and the ventilation exhaust duct door shall be made to verify that they are operable.

(4) Before operation with fueled experiments whose power generation is greater than 1 W, leak rate shall be verified when the interval since the last verification is greater than 12 months.

Bases: Surveillance of this equipment will verify that the confinement of the reactor room is maintained.

4.7 Airborne Effluents

Applicability: This specification applies to the surveillance of the instrument that monitors the airborne effluents from the ground floor experimental area.

Objective: The objective is to ensure that the airborne effluent monitor is operating and properly calibrated.

Specifications:

(1) Before each day's operation or before each operation extending more than one day, when either of the neutron beam ports are drained, the centrifugal blower that exhausts the area shall be in operation and a channel check shall be performed on the airborne effluent monitor.

(2) A calibration of the airborne effluent montior will be performed using a radioactive source semiannually.

Bases: The daily channel check of the monitor will ensure that it is operable. The semiannual calibration with an external source will permit any long-term drift to be corrected. The analysis is given in Chapter IX of Amendment 1 to the SAR (UVAR-18, Part I).

4.8 Reactor Fuel Dose Measurements

Applicability: This specification applies to reactor fuel possessed under the reactor facility licenses.

Objectives: The objective of this specification is to ensure that the maximum quantity of special nuclear material does not exceed the limits specified in the facility licenses.

Specification:

(1) The amount of special nuclear material (SNM) possessed at the reactor facility will be determined, as necessary, to ensure that limits specified by the facility licenses are not exceeded. As a minimum, an evaluation will be completed and documented every 6 months.

(2) Fuel elements will be irradiated as a part of the core or shipped away from the reactor facility as necessary to ensure that the quantity of nonexempt SNM (as defined in 10 CFR 73) does not exceed that allowed by the facility licenses. If the amount of nonexempt SNM exceeds 5.0 kg the action specified in the Physical Security Plan will be implemented.

(3) If fuel elements have not been irradiated as a part of the core for at least one month, dose rate measurements of representative fuel elements will be made as necessary to determine which elements have dose rates higher than specified by 10 CFR 73.67(b).

Basis: The specification will provide a high degree of assurance that the amount of SNM and nonexempt SNM does not exceed the license limits. The amount of nonexempt SNM will normally be maintained at less than 5.0 kg. In the event that this quantity is exceeded, the Reactor Safety Committee will be informed and actions necessary to reduce the amount or other appropriate actions as defined in the Physical Security Plan will be taken.

4.9 Primary Coolant Conditions

Applicability: This specification applies to the surveillance of primary water quality.

Objective: The objective is ensure that water quality does not deteriorate over extended periods of time if the reactor is not operated.

Specification: The conductivity and pH of the primary coolant water shall be measured at least once every 2 weeks and shall be

Conductivity $\leq 5 \times 10^{-6}$ mhos/cm
pH between 5.0 and 7.5

Bases: Section 3.11 of these specifications ensures that the water quality is adequate during reactor operation. Section 4.9 ensures that water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

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 165 ($\pm$3%) g of U-235, or 18 fuel plates containing 195 ($\pm$3%) g of U-235, in the standard fuel elements. There shall be six fuel plates containing 82.5 ($\pm$3%) g of U-235, or nine fuel plates containing 98 ($\pm$3%) g of U-235, in the control rod fuel elements. Partially loaded fuel elements in which some of the fuel plates do not contain uranium may be used. The mass of U-235 listed above refers to the initial (zero burnup) loading.

Various core configurations 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 3.1 of these specifications are not exceeded.

5.2 Reactor Building

Applicability: This specification applies to the room containing the reactor pool and the control room.

Specifications:

(1) The reactor shall be housed in a room designed to restrict leakage, as stated in Section 3.7(1)(d) of these specifications.

(2) The reactor room shall be equipped with a ventilation system designed to exhaust air or other gases from the reactor room through a stack at a minimum of 37 ft above ground level.

(3) The minimum free volume of the reactor room shall be 60,000 ft^3.

Bases: The parameters specified were used in the safety and/or environmental impact analyses in the final SAR.

5.3 Fuel Storage

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.

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

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 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 of 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 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.

(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.

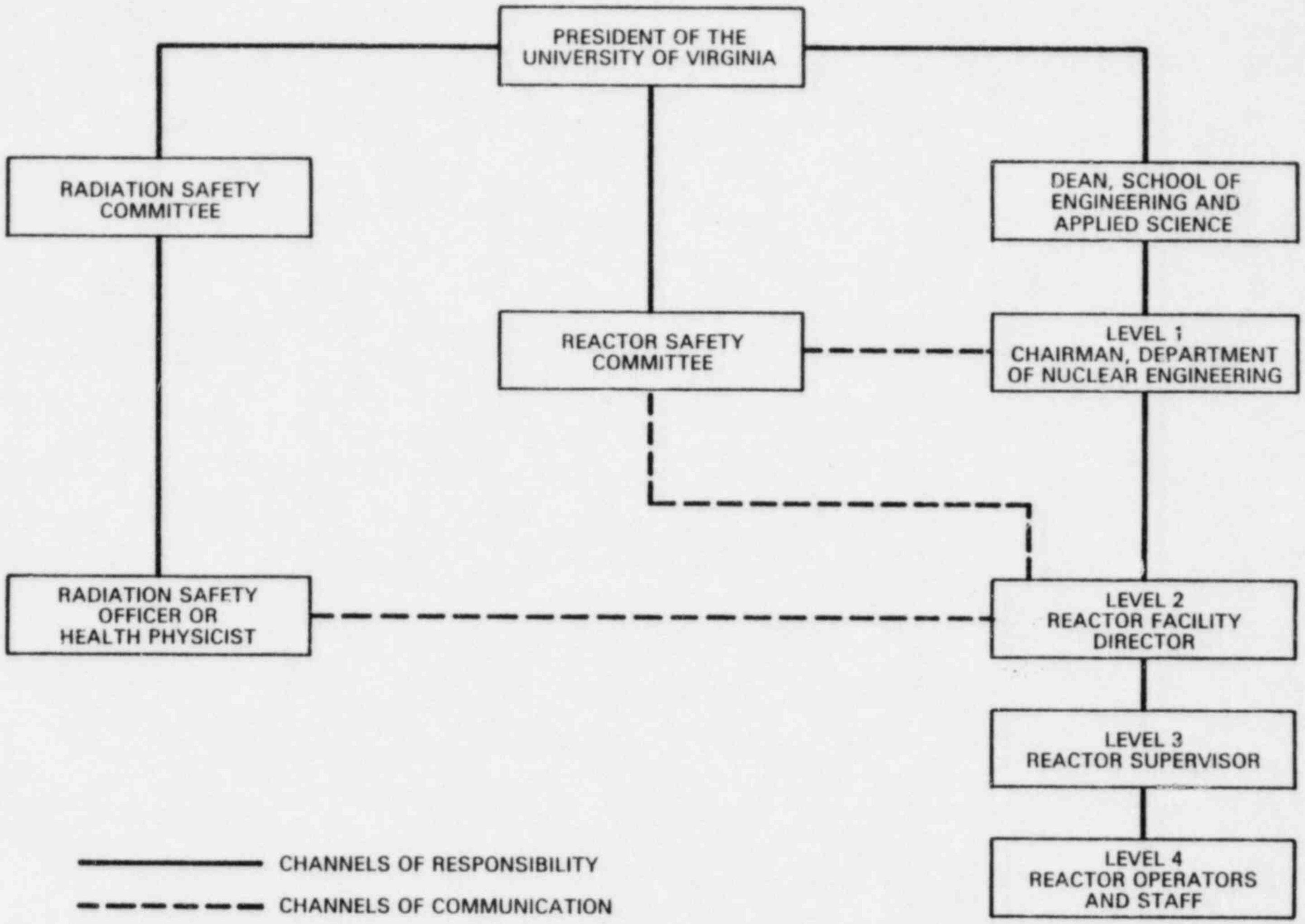

Figure 6.1 Organizational structure of UVA relating to reactor facility

(4) All 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 group, 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 miniumum 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 features 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 (These audits shall be performed at least once each calendar year.)

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 primary coolant system leaks, 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 6.7 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 malfunctons 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 0.005 $\Delta k/k$

(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 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 6.7 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 enviromental 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, corrected, and as-built drawings of the facility
(7) minutes of Reactor Safety Committee meetings

6.6 Reporting Requirements

In addition to the requirements of applicable regulations, 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 NRC Region II, Office of Inspection and Enforcement of

    (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, Division of Reactor Licensing, US NRC, Washington, D.C. 20555 with a copy to the NRC Region II, Office of Inspection and Enforcement of

    (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 a safety limit.

(3) A report within 30 days in writing to the Director, Division of Reactor Licensing, US NRC, Washington, D.C. 20555, with a copy to the Commission Region II, Office of Inspection and Enforcement of

(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, Division of Reactor Licensing, US NRC, Washington, D.C. 20555 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) rectivity 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, Division of Reactor Licensing, US NRC, Washington, D.C. 20555, with a copy to the Commission Region II Office of Inspection and Enforcement, providing the following information:

(1) A narrative summary will be prepared 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 will be prepared showing the energy generated by the reactor (in megawatt hours) and the number of hours the reactor was critical each quarter during the year.

(3) A report will be made 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 shall be prepared of the number of emergency shutdowns and inadvertent scrams, including their reasons and the corrective actions taken.

(5) A summary will be prepared 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 will be prepared of the nature and amount of radioactive gaseous, liquid and solid effluents released or discharged 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 report will be prepared with a description of any environmental surveys performed outside the facility.

(8) A summary will be prepared 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.