APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF VIRGINIA REACTOR

FACILITY LICENSE NO. R-66
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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.

Beamports: The beamports are the two 8-in. diameter neutron beamports 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 input values. 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) an 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) an incore operation designed to measure reactor characteristics.

Experimental Facility: An experimental facility is a 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 a 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 an experiment that contains U-235, U-233 or Pu-239 in levels exceeding trace quantities. Reactor fuel elements are not included in this definition.

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 parameter 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 parameter.

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° F) with the effect of xenon poisoning on core reactivity accounted for if greater than or equal to 0.07$. 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.

Reactor Staff: The Reactor Director and all personnel administratively reporting to him.

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 an 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.

Surveillance 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 parameter is its actual value at any instant.

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2.0. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTINGS

2.1. Safety Limit

Applicability: This specification applies to the maximum temperature of the 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 the general public or facility personnel.

Specification: The temperature of the aluminum cladding shall not reach its melting point.

Basis: Major fission product release to the pool, and potentially to the environment, will not occur so long as fuel plate integrity is maintained. The fuel plates are composed mostly of aluminum, the component with the lowest melting temperature. Analysis presented in the LEU SAR provides the basis for setting LSSS such that the burnout limit shown in LEU SAR Figure 9-6 is not reached. If the burnout limit is not reached, fuel integrity is not compromised.

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2.2. Limiting Safety System Settings

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

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

Specifications.

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

Reactor Thermal Power	= 3.0 MWt	(max)
Reactor Coolant Flow Rate	= 900 gpm	(min)
Reactor Coolant Inlet Temperature	= 108°F	(max)
Height of Water above Core	= 19'2"	(min)
Reactor Period	= 3.3 sec	(min)

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

Reactor Power	= 300 kWt	(max)
Reactor Coolant Inlet Temperature	= 108°F	(max)
Reactor Period	= 3.3 sec	(min)

Bases: The analysis in the LEU SAR shows there is sufficient margin between these settings and the safety limit under the most adverse conditions of operation:

(1) For the forced convection mode, the LEU SAR considers accidents with reactor power at 3.45 MW, a period of 3 seconds, pool inlet temperature of 111°F and a coolant flow of 837 gpm. The maximum fuel plate temperature calculated was considerably below the aluminum clad melting point. The LSSS specified above for this mode of operation are more conservative than the parameters used in the LEU SAR analysis.

(2) 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). The LEU SAR shows that the maximum fuel plate temperature under natural convection with initial power of 750 kW and pool inlet temperature of 111°F was well below the aluminum clad melting point. The LSSS specified above for this mode of operation are below the analyzed condition.

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3.0. LIMITING CONDITIONS FOR OPERATION

3.1. Reactivity

Applicability: These specifications apply to the reactiv'ty 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 shim rods, with secured experiments (see Section 1.0) in place and referred to the cold, xenon-free condition with the highest-worth shim rod fully withdrawn, is greater than 0.55$.

(2) An experiment with a reactivity worth greater than 0.60$ must be a secured experiment.

(3) The total reactivity worth of the two experiments having the highest reactivity worth is less than 2.00$.

(4) The total reactivity worth of all experiments is less than 2.70$.

(5) The maximum excess reactivity with fixed experiments in place and referred to cold, xenon-free condition shall be limited to 7.00$.

Bases: 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.

(1) 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 the highest worth control rod should stick in the fully withdrawn position.

(2) The reactivity of 0.60$ in Specification 3.1(2) corresponds to a 3-sec period. LEU SAR analysis shows that this results in the greatest power overshoot starting from a low power level.

(3) The reactivity of 2.00$ in Specification 3.1(3) is less than 2.16$ which corresponds to a 6.9-msec period. Reactor Core DU-12/25 of the SPERT-1 series of tests had MTR plate type 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.

(4) The total reactivity of 2.70$ in Specification 3.1(4) places a reasonable upper limit on the worth of all experiments.

(5) The limit of 7.00$ 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 LEU SAR. In general, the excess reactivity is limited by the shutdown margin requirement.

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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 above that described in the LEU SAR.

Specifications on the pool-water level are included as safety measures in the event of a serious loss of primary water. Reactor operations are terminated if a major leak occurs in the primary system. The analysis in the LEU 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.

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TABLE 3.1 SAFETY SYSTEM CHANNELS

Measuring Channel	Minimum No. Operable	Set Point*	Function	Operating Mode Required
Pool water level monitor	2	19'2" (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 convec. mode
		Application of power	Scram	Natural convection mode
Primary coolant flow	1	900 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)	Scram	Natural Convection mode
Reactor period	1	3.3 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.

3.3. Reactor Instrumentation

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

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.

The radiation monitors provide information to operating personnel of a decrease in pool-water level and of an 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.

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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.

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 a neutron beamport vented to the atmosphere is drained of water during reactor operations, the effluent shall be monitored by an instrument located in the effluent vent and the effluent vent will have sufficient flow to maintain releases within 10 CFR 20 limits.

Basis: The basis for this specification is given by the analysis in the LEU SAR.

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 the LEU SAR.

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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 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 the LEU SAR.

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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 0.15$.

(3) Experiments worth more than 0.15$ 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 0.50$ may be inserted or removed with the reactor 2.70$ or more subcritical.

(5) If an experiment worth more than 0.50$ 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 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 safety operation of the reactor.

(11) The Co-60 rods possessed under the UVAR Operating License shall be used and stored in the UVAR pool at distances greater than 5 feet from the operating UVAR reactor. Gamma irradiation facilities utilizing the Co-60 rods shall be designed to prevent physical damage to the Co-60 rods. UVAR pool water samples shall be subjected to gamma spectroscopy for the presence of Co-60 on a monthly frequency, (interval not to exceed six weeks) to assure that substantial leakage of Co-60 from the rods to reactor pool water does not occur.

<u>Bases</u>: (1-10) 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 0.15$ that can be inserted or removed with the reactor in operation is to accommodate experiments in the hydraulic rabbit.

(11) The Co-60 rods are to be kept a safe distance away from the UVAR reactor when it is operated, to avoid neutron activation and possible failure of the rod cladding, which may result in leakage of Co-60 to the reactor pool water. The Co-60 rods and the gamma irradiation facilities in which they are used will not be used in conjunction with the UVAR.

The monthly reactor pool water sampling frequency, adopted to monitor possible Co-60 leakage from the rods, is the same as that used in the U.S. AEC Safety Evaluation that was performed for these rods by the Division of Reactor Licensing on August 4, 1971. This is a reasonable frequency, for the most likely damage to the rods would be caused by cladding corrosion leading to pin holes. Co-60 leakage under these circumstances would proceed very slowly, into a large pool of water. Therefore, a monthly water sampling and analysis frequency should be adequate to indicate contamination levels before they become significant.

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3.7. Operation with Fueled Experiments

Applicability: This specification applies to the operation of the reactor with a 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 watt (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 calculated total energy produced by the experiment shall not exceed 600 W-years.

(d) The leak rate from the reactor room is not greater than 50% of containment of 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 calculated total energy produced by the experiment shall not exceed 600 W-years.

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 the LEU SAR. 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 the LEU SAR.

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 foot 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.

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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 to 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 limit will not be exceeded in a short period transient. The analysis is given in the LEU SAR.

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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 7 1/2 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.

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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.

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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 semiannually. 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 available 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.

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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 above specifications (1 through 4) do not apply to the following reactor safety 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 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.

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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 7 1/2 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).

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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 objectives 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.

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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.

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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.

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4.7. Airborne Effluents

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

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

Specifications:

(1) When the operation of the airborne effluent monitor is required (TS 3.4.1.), a channel check shall be performed on the monitor prior to reactor operation.

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

Bases: The daily channel check of the monitor will ensure that it is operable. The semiannual calibration with an external source will permit long-term drift to be corrected. It is noted that the use of the airborne effluent monitor is required only if one of the two eight-inch neutron beamports, when in the process of being re-filled with water, is vented past the monitor to the atmosphere (see LEU SAR analysis).

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4.8. Primary Coolant Conditions

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

Objective: The objective is to 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 3.10 of these specifications ensures that the water quality is adequate during reactor operation. Section 4.7 ensures that water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

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5.0. DESIGN FEATURES

5.1. Reactor Fuel

The UVAR reactor fuel shall be LEU fuel approved for use by the Nuclear Regulatory Commission.

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 LEU 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, assuming water 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 experience in the nuclear field. 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.

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

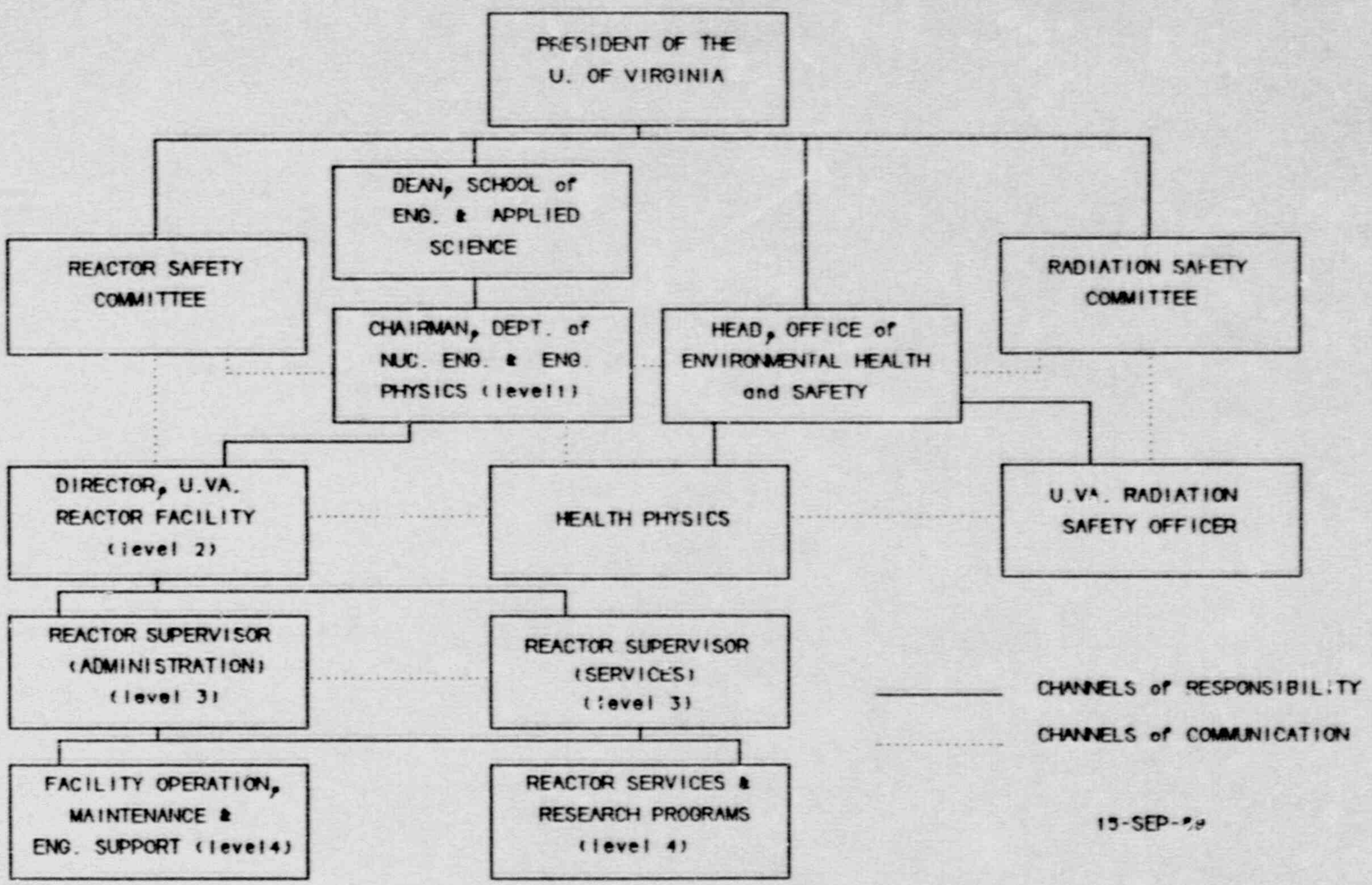

Figure 6.1 Organizational Structure of U.Va. Reactor Facility

(5) One or more health physicists, organizationally independent of the Reactor Staff as shown in Figure 6.1, shall be responsible for radiological safety at the facility.

6.2. Reactor Safety Committee

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 one member will be from the Reactor Staff. 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 meeting 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 and Audit Functions

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 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.

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6.3. Standard 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) preventative 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

(7) radiation control.

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 procedure shall be documented and subsequently reviewed by the Reactor Safety Committee.

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6.4. Required Actions

6.4.1. Actions To Be Taken in the Event the Safety Limit is Exceeded

In the event the 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) 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 0.70$.

(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

(7) major damage to the Co-60 rods resulting in Co-60 concentrations in reactor pool water in excess of 1×10^{-3} uCi/ml.

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.

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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) reactor operation logbook
(2) reactor systems maintenance records
(3) experiments performed using the reactor
(4) reportable occurrences
(5) equipment and component surveillance activity
(6) facility radiation and contamination surveys
(7) transfers of radioactive material to and from the R-66 license
(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 from the Reactor Facility
(2) off-site (radiological) environmental monitoring surveys
(3) fuel inventories and transfers
(4) radiation exposures for all personnel at the Reactor Facility
(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.

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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) accidental off-site release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or exposure.

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

(c) violation of the 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) accidental off-site release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury, or exposure.

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

(c) violation of the 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) substantial variance from performance specifications contained in these specifications or in the LEU SAR

(b) significant change in the transient or accident analyses as described in the LEU 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, and 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, 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 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.