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

UNIVERSITY OF FLORIDA

AMENDED FACILITY OPERATING LICENSE

DOCKET NO. 50-83

Amendment No. 13
License No. R-56

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

   A. The application for amendment by the University of Florida (the licensee) dated October 21, 1977, as supplemented by filings dated December 8, 1980; December 19, 1980; January 22, 1981; January 26, 1982; April 23, 1982; and May 5, 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 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 operating license renewal 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 materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR 30 and 70, including section 30.33, 70.23 and 70.31.

2. Facility Operating License No. R-56 is hereby amended in its entirety to read as follows:

A. This license applies to the Argonaut-type graphite-moderated and reflected light water cooled nuclear reactor (herein "the reactor") owned by University of Florida (the licensee), located on its campus in Gainesville, Alachua County, Florida and described in the application for license renewal dated October 21, 1977, and supplemental filings dated December 8, 1980; December 19, 1980; January 22, 1981; January 26, 1982; April 23, 1982 and May 5, 1982.

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

(1) Pursuant to Section 104c of the Act and Part 50 of Title 10, Chapter I, "Licensing of Production and Utilization Facilities," to possess, use, and operate the reactor as a utilization facility at the designated location in Gainesville, Florida, in accordance with the procedures and limitations set forth in this license.

(2) Pursuant to the Act and Title 10 CFR, Chapter I, Part 70, "Special Nuclear Materials", to receive, possess, and use (1) up to 4.82 kilograms of contained uranium 235, (2) a 1-curie sealed plutonium-beryllium neutron source, and (3) a 25-curie sealed antimony-beryllium neutron source all in connection with operation of the reactor.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Licensing and Byproduct Material," and Part 70, to possess, but not separate, such byproduct and special nuclear materials as may have been produced and may be produced by the operation of the facility.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state power levels not in excess of 100 kilowatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 13, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications. No changes shall be made in the Technical Specification unless authorized by the Commission as provided in Section 50.59 of 10 CFR Part 50.

(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 "Physical Security Plan for the University of Florida Training Reactor" dated August 4, 1980, submitted by letter dated August 7, 1980, as revised by letter dated April 27, 1981.

D. This amended license is effective as of the date of issuance and shall expire at midnight 20 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

Date of Issuance: AUG 3 0 1982

APPENDIX A

TECHNICAL SPECIFICATIONS AND BASES FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR

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1.0 DEFINITIONS*

Abnormal Occurrences: An abnormal occurrence is any one of the following:

(1) operating the reactor with a safety system setting less conservative than specified in the Limiting Safety System Setting section of the Technical Specifications

(2) operating the reactor in violation of a limiting condition for operation

(3) a malfunction of a safety system component or other component or system malfunction that could, or threatens to, render the system incapable of performing its intended safety function

(4) a release of fission products from the reactor fuel of a magnitude to indicate a failure of the fuel cladding

(5) an uncontrolled or unanticipated change in reactivity greater than one dollar (Reactor trips resulting from a known cause are excluded.)

(6) 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

(7) an uncontrolled or unanticipated release of radioactivity to the environment

Channel Calibration: A channel calibration is an adjustment of the channel components such that its output responds, within specified range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including readouts, alarms, or trips.

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

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

Independent Experiment: An independent experiment is one that is not connected by a mechanical, chemical, or electrical link.

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*The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

Inhibit: An inhibit is a device that prevents the withdrawal of control blades under a potentially unsafe condition.

Measured Value: The measured value of a parameter is the value as it appears at the output of a measuring channel.

Measured Channel: The measured 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 where it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating, or having incore components during operation.

Nonsecured Experiment: A nonsecured experiment, where it is intended that the experiment should not move while the reactor is operating, is held in place with less restraint than a secured experiment.

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

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

Reactor Operating: The reactor is considered to be operating whenever it is not secured or shutdown.

Reactor Safety System: The reactor safety system is that combination of measuring channels and associated circuitry that forms the automatic protective action to be initiated, or provides information which requires the initiation of manual protective action.

Reactor Secured: The reactor is secured when it contains sufficient fissile material or moderator present in the reactor, adjacent experiments or control rods, to attain criticality under optimum available conditions of moderation and reflection,

or

(1) the reactor is shut down, (2) electrical power to the control blade circuits is switched off and the switch key is in proper custody, (3) no work is in progress involving core fuel, core structure, installed control rods or control rod drives unless they are physically decoupled from the control rods, and (4) no experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment or one dollar, whichever is smaller.

Reactor Startup: A reactor startup is a series of operator manipulations of reactor controls (in accordance with approved procedures) intended to bring the reactor to a k_{eff} of 0.99 or greater. It does not include control blade manipulations made for purposes of testing equipment or component operability within a k_{eff} of 0.99 or less.

Reactor Shutdown: The reactor is shut down when all control blades are inserted and the reactor is subcritical by a margin greater than 2% $\Delta k/k$. When calculating the subcritical margin, no credit shall be taken for experiments, temperature effects or xenon poisoning.

Reactor Trip: A reactor trip is considered to occur whenever one of the two following actions take place:

(1) Rod-Drop Trip -- a gravity drop of all control blades into the reactor core as a result of terminating electrical power to the blade drive magnetic clutches.

(2) Full-Trip -- the water is dumped from the reactor core by the safety actuation of the dump valve in addition to the rod-drop trip.

Reportable Occurrence: A reportable occurrence is any of the conditions described in Section 6.5.2 of this specification.

Research Reactor: A research reactor is a device designed to support a self-sustaining neutron chain reaction to supply neutrons or ionizing radiation for research, developmental, educational, training, or experimental purposes, and which may have provisions for the production of nonfissile radioisotopes.

Rod-Drop Time: The rod-drop time is the elapsed time between the instant a limiting safety system set point is reached or a manual scram is initiated and the instant that the rod is fully inserted.

Safety Channel: A safety channel is a measuring channel in the reactor safety system.

Secured Experiment: A secured experiment is a stationary experiment held firmly in place by a mechanical device secured to the reactor structure or by gravity, providing that the weight of the experiment is such that it cannot be moved by a force of less than 60 lb.

Secured Experiment with Movable Parts: A secured experiment with movable parts is one that contains parts that are intended to be moved while the reactor is operating.

Shutdown Margin: The shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the system can be made subcritical by means of the control and safety systems (starting from any permissible operating condition) and that the reactor will remain subcritical without further operator action.

Unscheduled Shutdown: An unscheduled shutdown is any unplanned shutdown of the reactor after startup has been initiated.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. The principal physical barrier shall be the fuel cladding.

Applicability: These specifications apply to the variables that affect thermal, hydraulic, and materials performance of the core.

Objective: To ensure fuel cladding integrity.

Specifications:

(1) The steady-state power level shall not exceed 100 kWt.

(2) The primary coolant flow rate shall be greater than 18 gpm at all power levels greater than 1 watt.

(3) The primary coolant outlet temperature from any fuel box shall not exceed 200°F.

(4) The specific resistivity of the primary coolant water shall not be less than 0.4 megohm-cm for periods of reactor operations over 4 hours.

Bases: Operating experiences and detailed calculations of Argonaut reactors have demonstrated that Specifications (1) and (2) suffice to maintain the maximum fuel temperature below 200°F, which is well below the temperature at which fuel degradation would occur. For the readily available flow rate of up to 65 gpm, it has been shown that the fuel temperature will be well below 200°F for steady-state power operation of up to 500 kWt. No fuel damage is known to occur from transient operation up to 500% full power at the present 40 gpm primary flow rate. Specification (3) is included to prevent boiling of the primary coolant at any fuel box. Specification (4) suffices to maintain adequate water quality conditions to prevent deterioration of the fuel cladding and still allow for expected transient changes in the water resistivity.

2.2 Limiting Safety System Settings

Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions.

Applicability: These specifications are applicable to the reactor safety system set points.

Objective: To ensure that automatic protective action is initiated before exceeding a safety limit or before creating a radioactive hazard that is not considered under safety limits.

Specifications: The limiting safety system settings shall be

(1) Power level at any flow rate shall not exceed 125 kW.

(2) The primary coolant flow rate shall be greater than 30 gpm at all power levels greater than 1 watt.

(3) The average primary coolant outlet temperature shall not exceed 155°F when measured at any fuel box outlet.

(4) The reactor period shall not be faster than 3 sec.

(5) The high voltage applied to Safety Channels 1 and 2 neutron chambers shall be 90% or more of the established normal value.

(6) The primary coolant pump shall be energized during reactor operations.

(7) The primary coolant flow rate shall be monitored at the return line.

(8) The primary coolant core level shall be at least 2 in. above the fuel boxes.

(9) The secondary coolant flow shall satisfy the following conditions when the reactor is being operated at power levels equal to or larger than 1 kW:

(a) Power shall be provided to the well pump and the well water flow rate shall be larger than 60 gpm when using the well system for secondary cooling.

or

(b) The water flow rate shall be larger than 8 gpm when using the city water system for secondary cooling.

(10) The reactor shall be shut down when the main alternating current (ac) power is not operating.

(11) The reactor vent system shall be operating during reactor operations.

(12) The water level in the shield tank shall not be reduced 6 in. below the established normal level.

Bases: The University of Florida Training Reactor (UFTR) limiting safety system settings (LSSS) are established from operating experience and safety considerations. The LSSS 2.2.3 (1) through (10) are established for the protection of the fuel, the fuel cladding, and the reactor core integrity. The primary and secondary bulk coolant temperatures, as well as the outlet temperatures for the six fuel boxes, are monitored and recorded in the control room. LSSS 2.2.3 (11) are established for the protection of reactor personnel in relation to accumulation of argon-41 in the reactor cell and for the control of radioactive gaseous effluents from the cell. LSSS 2.2.3 (12) are established to protect reactor personnel from potential external radiation hazards caused by loss of biological shielding.

3.0 LIMITING CONDITIONS FOR OPERATION

Limiting conditions for operation are the lowest functional capabilities or performance levels required of equipment for safe operation of the facility.

3.1 Reactivity Limitations

(1) Shutdown Margin: The minimum shutdown margin, with the most reactive control blade fully withdrawn, shall not be less than 2% $\Delta k/k$.

(2) Excess Reactivity: The core excess reactivity at cold critical, without xenon poisoning, shall not exceed 2.3% $\Delta k/k$.

(3) Coefficients of Reactivity: The primary coolant void and temperature coefficients of reactivity shall be negative.

(4) Maximum Single Blade Reactivity Insertion Rate: The reactivity insertion rate for a single control blade shall not exceed 0.06% $\Delta k/k$/sec, when determined as an average over any 10 sec of blade travel time from the characteristic experimental integral blade reactivity worth curve.

(5) Experimental Limitations: The reactivity limitations associated with experiments are specified in Section 3.5 of this report.

(6) Bases: These specifications are provided to limit the amount of excess reactivity to within limits known to be within the self-protection capabilities of the fuel, to ensure that a reactor shutdown can be established with the most reactive blade out of the core, to ensure a negative overall coefficient of reactivity, and to limit the reactivity insertion rate to levels commensurable with efficient and safe reactor operation.

3.2 Reactor Control and Safety Systems

3.2.1 Reactor Control System

(1) Four cadmium-tipped, semaphore-type blades shall be used for reactor control. The control blades shall be protected by shrouds to ensure freedom of motion.

(2) Only one control blade can be raised by the manual reactor controls at any one time. The safety blades shall not be used to raise reactor power simultaneously with the regulating blade when the reactor control system is in the automatic mode of operation.

(3) The reactor shall not be started unless the reactor control system is operable.

(4) The control-blade-drop time shall not exceed 1 sec from initiation of blade drop to full insertion (rod-drop time), as determined according to surveillance requirements.

(5) The following control blade withdrawal inhibit interlocks shall be operable for reactor operation for the following conditions:

(a) a source (startup) count rate of less than 2 cps (as measured by the wide range drawer operating on extended range)

(b) a reactor period less than 10 sec

(c) safety channels 1 and 2 and wide range drawer calibration switches not in OPERATE condition

(d) attempt to raise any two or more blades simultaneously when the reactor is in manual mode, <u>or</u> two or more safety blades simultaneously when the reactor is in automatic mode

(e) power is raised in the automatic mode at a period faster than 30 sec (The automatic controller action is to inhibit further regulating blade withdrawal or drive the regulating blade down until the period is $\geqq$ 30 sec.)

(6) Following maintenance or modification to the reactor control system, an operability test and calibration of the affected portion of the system, including verification of control blade drive speed, shall be performed before the system is considered operable.

3.2.2 Reactor Safety System

(1) The reactor shall not be started unless the reactor safety system is operable in accordance with Table 3.1.

(2) Tests for operability shall be made in accordance with Table 3.2.

3.2.3 Reactor Control and Safety Systems Measuring Channels

The minimum number and type of measuring channels operable and providing information to the control room operator required for reactor operation are given as follows:

Channel	No. operable
Safety 1 and 2 power channel	2
Linear with auto controller	1
Log N and period channel*	1
Startup channel*	1
Rod position indicator	4
Coolant flow indicator	1
Coolant temperature indicator	
Primary	6
Secondary	1
Core level	1
Ventilation system	
Core vent annunciator	1
Exhaust fan annunciator	1
Exhaust fan rpm	1

*Subsystems of the wide range drawer

Table 3.1 Specifications for reactor safety system trips

Specification	Type of safety system trip
Automatic Trips	
Period less than 3 sec	Full
Power at 125% of full power	Full
Loss of chamber high voltage $\geq$ 10%)	Full
Loss of electrical power to control console	Full
Primary cooling system	Rod-drop
Loss of pump power	
Low-water level in core (< 42.5")	
No outlet flow	
Low inlet water flow (< 30 gpm)	
Secondary cooling system (at power levels above 1 kW 8 gpm)	
Loss of flow (well water < 60 gpm, city water 8 gpm)	Rod-drop
Loss of pump power	
High primary coolant average outlet temperature ($\geq$ 155°F)	Rod-drop
Shield Tank	Rod-drop
Low water level	
Ventilation system	Rod-drop
Loss of power to dilution fan	
Loss of power to core vent system	
Manual Trips	
Manual scram bar	Rod-drop
Console key-switch OFF (two blades off bottom)	Full

Table 3.2 Safety system operability tests

Component or scram function	Frequency
Log-N period channel power level safety channels	Before each reactor startup following a shutdown in excess of 6 hr, <u>and</u> after repair <u>or</u> deenergization caused by a power outage
10% reduction of safety channels high voltage	4/year (4-month maximum interval)
Loss of electrical power to console	4/year (4-month maximum interval)
Loss of primary coolant pump power	4/year (4-month maximum interval)
Loss of primary coolant level	4/year (4-month maximum interval)
Loss of primary coolant flow	With daily checkout
High average primary coolant outlet temperature	With daily checkout
Loss of secondary coolant flow (at power levels above 1 kW)	With daily checkout
Loss of secondary coolant well pump power	4/year (4-month maximum interval)
Loss of shield tank water level	4/year (4-month maximum interval)
Loss of power to vent system and dilution fan	4/year (4-month maximum interval)
Manual scram	With daily checkout

3.2.4 Bases

The reactor control system provides the operator with reactivity control devices to control the reactor within the specified range of reactivity insertion rate and power level. The operator has available digital blade position indicators for the three safety blades and the regulating blade. The three safety blades can only be manipulated by the UP-DOWN blade switches (manual); the regulating blade can be manually controlled or placed under automatic control, which uses the linear channel as the measuring channel, and a percent of power setting control. The two independent reactor safety channels provide redundant protection and information on reactor power in the range 1% - 150% of full power. The linear power channel is the most accurate neutron instrumentation channel, and provides a signal for reactor control in automatic mode. The percent of power information is displayed by the linear channel two-pen recorder. It does not provide a protective function. The log wide range drawer provides a series of information, inhibit, and protection function from extended source range to full power. The safety channel 1 signal and the period protection signal are derived from the wide range drawer. The wide count drawer provides protection during startup through the source count rate interlock (2 cps), 10-sec period inhibit and the 3-sec period trip. The primary and secondary coolant flow rate, temperature and level sensing instrumentation provides information and protection over the entire range of reactor operations and is proven to be conservative from a safety viewpoint. The key switch prevents unauthorized operation of the reactor and is an additional full trip (manual scram) control available to the operator. The core level trip provides redundant protection to the primary flow trip. The core level trip acts as an inhibit during startup until the minimum core water level is reached.

3.3 Reactor Vent System

These specifications apply to the equipment required for controlled release of gaseous radioactive effluent to the environment via the stack or its confinement within the reactor cell.

3.3.1 Specification

(1) The reactor vent system shall be capable of maintaining an air flow rate between 1 and 400 cfm from the reactor cavity whenever the reactor is operating and as specified in these Technical Specifications. The reactor air cavity flow shall be periodically calibrated to minimize argon-41 releases to the environment while maintaining a negative pressure within the reactor cavity to minimize radioactive hazards to reactor personnel.

(2) The diluting fan shall be operated whenever the reactor is in operation and as otherwise specified in these Technical Specifications, at an exhaust flow rate larger than 10,000 cfm.

(3) The reactor vent system is interlocked to shut off automatically when the air conditioning/ventilation system is shut off. The air conditioning/ventilation system is automatically shut off whenever the reactor building evacuation alarm is automatically or manually actuated.

(4) All doors to the reactor cell shall normally be closed while the reactor is operating. Transit is not prohibited through air lock and control room doors.

3.3.2 Bases

Under normal conditions, to effect controlled release of gaseous activity through the reactor vent system, a negative cell pressure is required so that any building leakage will be inward. Under emergency conditions, the reactor vent system will be shut down and the damper closed, thus minimizing leakage of radioactivity from the reactor cell.

3.4 Radiation Monitoring Systems and Radioactive Effluents

3.4.1 Area Radiation Monitors

The reactor cell shall be monitored by at least three area radiation monitors, two of which shall be capable of audibly warning personnel of high radiation levels. The output of at least two of the monitors shall be indicated and recorded in the control room. The set points for the radiation monitors shall be in accordance with Table 3.3

3.4.2 Argon-41 Discharge

The following operational limits are specified for the discharge of argon-41 to the environment:

(1) The concentration of argon-41 in the gaseous effluent discharge of the UFTR is determined by averaging it over a consecutive 30-day period.

(2) The dilution resulting from the operation of the stack dilution fan (flow rate of 10,000 cfm or more) and atmospheric dilution of the stack plume (a factor of 200) may be taken into account when calculating this concentration.

(3) When calculated as above, discharge concentration of argon-41 shall not exceed MPC (4.0×10^{-8}μc/ml). Operation of the UFTR shall be such that this maximum permissible concentration (averaged over a month) is not exceeded.

Table 3.3 Radiation monitoring system settings

Type	No. of required operable functions	Alarm(s) setting	Purpose
Area radiation monitors	3 detecting 2 audio alarming 2 recording	5 mr/hr low level 25 mr/hr high level	Detect/alarm/record low and high level external radiation
Air particulate monitors	1 detecting 1 audio alarming 1 recording	Range adjusted according to APD* type (according to monitoring requirements)	Detect/alarm/record airborne radioactivity in the reactor cell
Stack radiation monitor	1 detecting 1 audio alarming 1 recording	(1) Fixed alarm at 4000 cps (2) Adjustable alarm as per power level	Detect/alarm/record release of gaseous radioactive effluents in the reactor vent duct to the environs

*Air particulate detector

Notes: For maintenance or repair, the required radiation monitors may be replaced by suitable portable instruments provided the intended function is being accomplished.

Service, calibration, and testing interruptions for brief periods are permissible when the reactor is not in operation.

3.4.3 Reactor Vent System

The reactor vent system shall be operated at all times during reactor operation. In addition, the vent system shall be operated until the stack monitor indicates less than 10 counts per second (cps). Whenever the reactor vent system is operating, air drawn through the reactor vent system shall be continuously monitored for gross concentration of radioactive gases. The output of the monitor shall be indicated and recorded in the control room. The reactor vent system shall be immediately secured upon detection of: a failure in the monitoring system, a failure of the absolute filter, or an unanticipated high stack count rate.

3.4.4 Air Particulate

The reactor cell environment shall be monitored by at least one air particulate monitor, capable of audily warning personnel of radioactive particulate airborne contamination in the cell atmosphere.

3.4.5 Liquid Effluents Discharge

(1) The liquid waste from the radioactive liquid waste holding tanks shall be sampled and the activity measured before release to the sanitary sewage system.

(2) Releases of radioactive liquid waste from the holding tanks/campus sanitary sewage system shall be in compliance with the limits specified in 10 CFR 20, Appendix B, Table 1, Column 2, as specified in 10 CFR 20.303.

3.4.6 Solid Radioactive Waste Disposal

Solid radioactive waste disposal shall be accomplished in compliance with applicable regulations and under the control of the Radiation Control Office of the University of Florida.

3.5 Limitations on Experiments

Applicability: These specifications apply to all experiments or experimental devices installed in the reactor core or its experimental facilities.

Objectives: The objectives are to maintain operational safety and prevent damage to the reactor facility, reactor fuel, reactor core, and associated equipment; to prevent exceeding the reactor safety limits; and to minimize potential hazards from experimental devices.

Specifications:

(1) General

The reactor manager and the radiation control officer (or their duly appointed representative) shall review and approve in writing all proposed experiments prior to their performance. The reactor manager shall refer to the Reactor Safety Review Subcommittee (RSRS) the evaluation of the safety aspects of new experiments and all changes to the facility that may be necessitated by the requirements of the experiments and that may have safety significance. When experiments contain substances that irradiation in the reactor can convert into a material with significant potential hazards, a determination will be made about the acceptable reactor power level and length of irradiation, taking into account such factors as: isotope identity and chemical and physical form and containment, toxicity, potential for contamination of facility or environment, problems in removal or handling after irradiation including containment, transfer, and eventual disposition. Guidance should be obtained from the ANS 15.1 Standard. Experimental apparatus, material, or equipment to be inserted in the reactor shall be reviewed to ensure noninterference with the safe operation of the reactor.

(2) Classification of Experiments

Class I--Routine experiments, such as gold foil irradiation. This class shall be approved by the reactor manager; the radiation control officer may be informed if deemed necessary.

Class II--Relatively routine experiments that need to be documented for each new group of experimenters performing them, or whenever the experiment has not been carried out for one calendar year or more by the original experimenter, and that pose no hazard to the reactor, the personnel, or the public. This class shall be approved by the reactor manager and the radiation control officer.

Class III--Experiments that pose significant questions regarding the safety of the reactor, the personnel, or the public. This class shall be approved by the reactor manager and the radiation control officer, after review and approval by the Reactor Safety Review Subcommittee (RSRS).

Class IV--Experiments that have a significant potential for hazard to the reactor, the personnel, or the public. This class shall be approved by the reactor manager and radiation control officer after review and approval by the RSRS and specific emergency operating instructions shall be established for conducting the experiments.

(3) Reactivity Limitations on Experiments

(a) The absolute reactivity worth of any single movable or nonsecured experiment shall not exceed 0.6% $\Delta k/k$.

(b) The total absolute reactivity worth of all experiments shall not exceed 2.3% $\Delta k/k$.

(c) When determining the absolute reactivity worth of an experiment, no credit shall be taken for temperature effects.

(d) An experiment shall not be inserted or removed unless all the control blades are fully inserted or its absolute reactivity worth is less than that which could cause a positive 20-sec stable period.

(4) Explosive Materials

Explosive materials shall not be irradiated.

(5) Thermal-Hydraulic Effects

The experiment shall be designed so that during normal operation, or failure, the thermal hydraulic parameters of the core do not exceed the safety limits.

(6) Chemical Effects

The experiment shall be designed so that during normal operation, or failure, the physical barrier described in Section 2.1 will not be compromised by either chemical or blast effects from the experiment.

(7) Fueled Experiments

A limit should be established on the inventory of fission products in experiments containing fissile material, according to its potential hazard and as determined by the RSRS.

(8) Radioactive Releases from Experiments

Class III and Class IV experiments shall be evaluated for their potential release of airborne radioactivity and limits shall be established for the permissible concentration of radioisotopes in the experiments, according to the 10 CFR 20 limitations for exposure of individuals in restricted and unrestricted areas.

Bases: The general specifications ensure that an adequate review process is followed to determine the safety, conditions, and procedures for all experiments. The classification of experiments clearly delineates the responsibility for approving experiments according to their potential hazards, to ensure that potentially hazardous experiments are analyzed for their safety implications, and that appropriate procedures are established for their execution. The reactivity limitations on experiments are established to prevent prompt criticality by limiting the worth of movable or nonsecured experiments, to prevent a reactivity insertion larger than the stipulated maximum step reactivity insertion in the accident analysis, and to allow for reactivity control of experiments within the reactor control system capabilities (20-sec positive period limitation). These specifications prohibit the irradiation of explosive materials and limit the amount of fissile materials that can be irradiated in the reactor according to its potential hazard and the reactor system's capability to handle a potential release to the cell environment. Explosive materials are defined as those materials normally used to produce explosive or detonating effects, materials that can chemically combine to produce explosion or detonations, or any materials that can undergo explosive decomposition under influence of neutron, gamma, or heat flux of the reactor or as defined by applicable standards.

3.6 Reactor Building Evacuation Alarm

These specifications apply to the equipment required for the evacuation of the reactor cell and the reactor building (including the reactor annex).

Specification: The reactor cell and the reactor building shall be evacuated when any of the following conditions exist:

(1) The evacuation alarm is actuated automatically when two area radiation monitors alarm high (≥ 25 mrems/hr) in coincidence.

(2) The evacuation alarm is actuated manually when an air particulate monitor is in a valid alarm condition.

(3) The evacuation alarm is actuated manually when a reactor operator detects a potentially hazardous radiological condition and preventive actions are required to protect the health and safety of operating personnel and the general public.

Bases: To provide early and orderly evacuation of the reactor cell and the reactor building and to minimize radioactive hazards to the operating personnel and reactor building occupants.

3.7 Fuel and Fuel Handling

Applicability: These specifications apply to the arrangement of fuel elements in core and in storage, as well as the handling of fuel elements.

Objectives: The objectives are to establish the maximum core loading for reactivity control purposes, to establish the fuel storage conditions, and to establish fuel performance and fuel-handling specifications with regard to radiological safety considerations.

Specifications:

(1) The maximum fuel loading shall consist of 24 full fuel elements consisting of 11 plates each containing enriched uranium and cladded with high purity aluminum.

(2) Fuel element loading and distribution in the core shall comply with the fuel-handling procedures.

(3) Fuel elements exhibiting release of fission products because of cladding rupture shall, upon positive identification, be removed from the core. Fission product contamination of the primary water shall be treated as evidence of fuel element failure.

(4) The reactor shall not be operated if there is evidence of fuel element failure.

(5) All fuel shall be moved and handled in accordance with approved procedures.

(6) Fuel elements or fueled devices shall be stored and handled out of core in a geometry such that the k_{eff} is less than 0.8 under optimum conditions of moderation and reflection.

(7) Irradiated fuel elements or fueled devices shall be stored so that temperatures do not exceed design values.

Bases: The fuel loading is based on the present fuel configuration. The reactor systems do not have adequate engineering safeguards to continue operating with a detectable release of fission products into the primary coolant. The fuel is to be stored in a safe configuration and shall be handled according to approved written procedures for radiological safety purposes.

3.8 Primary Water Quality

Applicability: These specifications apply to primary cooling system water in contact with fuel elements.

Objective: To minimize corrosion of the aluminum cladding of fuel plates and activation of dissolved materials.

Specifications:

(1) Primary water temperature shall not exceed 155°F.

(2) Primary water shall be demineralized, light water with a specific resistivity of not less than 0.5 megohm-cm after the reactor is operated for more than 6 hr.

(3) Primary water shall be sampled and evaporatively concentrated, and the gross radioactivity of the residue shall be measured with an adequate measuring channel. This specification procedure shall prevail (a) during the weekly checkout, (b) upon the appearance of any unusual radioactivity in the primary water or the primary water demineralizers, and (c) before the release of any primary water from the site.

(4) Primary equipment pit water level sensor shall alarm in the control room whenever a detectable amount of water (1 in. above floor level) exists in the equipment pit.

Bases: Specifications 3.8.3(1) and 3.8.3(2) are designed to protect the fuel element integrity and are based upon operating experience. At the specified quality, the activation products (of trace minerals) do not exceed acceptable limits. Specification 3.8.3(3) is designed to detect and identify fission products resulting from fuel failure and to fulfill reportability requirements pertaining to liquid wastes. Specification 3.8.3(4) is designed to alert the operator to potential loss of primary coolant, to prevent reactor operations with a reduced water inventory, and to minimize the possibility of an uncontrolled release of primary coolant to the environs.

3.9 Radiological Environmental Monitoring Program

3.9.1 General

The UFTR Radiological Environmental Monitoring Program is conducted to ensure that the radiological environmental impact of reactor operations is as low as reasonably achievable (ALARA); it is conducted in addition to the radiation monitoring and effluents control specified under Section 3.8 of these Technical Specifications.

The Radiological Environmental Monitoring Program shall be conducted as specified below and under the supervision of the radiation control officer.

3.9.2 Radiological Environmental Monitoring

(1) Monthly environmental radioactivity surveillance outside the restricted area shall be conducted by measuring the gamma doses at selected fixed locations surrounding the UFTR complex with acceptable personnel monitoring devices. A minimum of six independent locations shall be used. A review of potential causes shall be conducted whenever a measured dose of over 40 mrems/month at two or more locations is determined and a report shall be submitted to the RSRS for review.

(2) Radioactivity surveillance of the restricted area (reactor cell) shall be conducted as follows:

(a) Surface contamination in the restricted area shall be measured by taking random swipes in the reactor cell during the weekly checkout. Measured surface contamination greater than 100 dpm/cm^2 beta-gamma or greater than 50 dpm/100 cm^2 alpha are limiting conditions for operation requiring review and possible radiological safety control actions.

(b) Airborne particulate contamination shall be measured using a high volume air sampler. Measured radioactive airborne contamination 25% above mean normal levels are limiting conditions for operation requiring review and possible radiological safety control actions.

(3) The following radioactivity surveys, using portable radiation monitors are limiting conditions for operation:

(a) Surveys measuring the radiation doses in the restricted area shall be conducted quarterly, at intervals not to exceed 4 months, <u>and</u> at any time a change in the normal radiation levels is noticed or expected. Radiation exposures shall be within 10 CFR 20 limits for radiation workers.

(b) Surveys measuring the radiation levels in the unrestricted areas surrounding the UFTR complex shall be conducted quarterly, at intervals not to exceed 4 months, <u>and</u> at any time a change in the normal radiation levels is noticed or expected. Doses shall be within 10 CFR 20 limits for the general public.

3.9.3 Bases

The bases for establishing the Radiological Environmental Surveillance Program are the established limits for internal and external radiation exposure <u>and</u> requirements that radiation doses be maintained ALARA.

4.0 SURVEILLANCE REQUIREMENTS

Surveillance requirements relate to testing, calibration, or inspection to ensure that the necessary quality of systems and components is maintained; that facility operation will be within safety limits; and that the limiting conditions for operation will be met. Tests not performed within the specified frequency because of physical or administrative limitations shall be performed before resuming normal operations.

4.1 Surveillance Pertaining to Safety Limits and Limiting Safety Settings

(1) Whenever an unscheduled shutdown occurs, an evaluation shall be conducted to determine whether a safety limit was exceeded.

(2) Safety system operability tests shall be performed in accordance with Table 3.2.

4.2 Surveillance Pertaining to Limiting Conditions for Operation

4.2.1 Reactivity Surveillance

(1) The reactivity worth and reactivity insertion rate of each control blade, the shutdown reactivity and excess reactivity shall be measured annually (at intervals less than 14 months) or whenever physical or operational changes create a condition requiring reevaluation of core physics parameters.

(2) The temperature coefficient of reactivity shall be measured annually at intervals not to exceed 14 months.

(3) The void coefficient of reactivity shall be checked biennially to ensure that it is negative, at intervals not to exceed 30 months.

4.2.2 Reactor Control and Safety System Surveillance

(1) The control blades drop time, from the fully withdrawn position, shall be measured semiannually but at intervals not to exceed 8 months. If maintenance is performed on a blade, the drive mechanism, or associated electronics, the rod-drop time shall be measured before the system is considered operable.

(2) The control blade full withdrawal and controlled insertion time shall be measured semiannually, at intervals not to exceed 8 months.

(3) Tests, limits, and frequencies of tests for the control blade withdrawal inhibit interlocks operability tests shall be performed as listed in Table 4.1.

Table 4 1 Control blade withdrawal inhibit interlocks operability tests

Inhibit	Limit	Frequency
Reactor period	$\leq$ 10 sec	Daily checkout
Safety channels and wide range drawer not in OPERATE position	-	Daily checkout
Multiple blade withdrawal	Any 2 or more blades simultaneously in Manual	Daily checkout
	Any 2 safety blades in Automatic	Daily checkout
Source count rate	<2 cps	Verification only when count rate <2 cps during daily checkout

(4) The mechanical integrity of the control blades and drive system shall be inspected during each incore inspection but shall be fully checked at least once every 5 years.

(5) Following maintenance or modification to the control blade system, an operability test and calibration of the affected portion of the system, including verification of control blade drive speed, shall be performed before the system is to be considered operable.

(6) The reactor shall not be started unless (a) the weekly checkout has been satisfactorily completed within 7 days prior to startup, (b) a daily checkout is satisfactorily completed within 8 hr prior to startup, and (c) no known condition exists that would prevent successful completion of a weekly or daily check.

(7) The limitations established under Paragraph 4.2.2(6)(a) and (b) can be deleted if a reactor startup is made within 6 hr of a normal reactor shutdown on any one calendar day.

(8) The following channels shall be calibrated annually, at intervals not to exceed 13 months, and any time a significant change in channel performance is noted:

(a) log N - period channel
(b) power level safety channels (2)
(c) linear power level channel

(d) primary coolant flow measuring system
(e) primary coolant temperature measuring system

(9) Following maintenance or modification to the reactor safety system, a channel test and calibration of the affected channel shall be performed before the reactor safety system is considered operable.

4.2.3 Reactor Vent System Surveillance

(1) The reactor vent system flow rates shall be measured annually at intervals not to exceeed 14 months, as follows:

(a) reactor cavity exhaust duct flow (1 cfm < flow rate < 400 cfm)
(b) stack flow rate > 10,000 cfm

(2) The following interlocks shall be tested quarterly at intervals not to exceed 8 months:

(a) core vent system damper closed if diluting fan is not operating
(b) reactor vent system shut off when the air conditioning system is shut off because of actuation of the evacuation alarm

4.2.4 Radiation Monitoring Systems and Radioactive Effluents Surveillance

(1) The area radiation monitor channels, the stack monitor, and the air particulate monitor shall be verified to be operable before each reactor startup, as required by the daily checkout. Calibration of radiation monitoring channels shall be performed quarterly at intervals not to exceed 4 months.

(2) The Ar-41 concentration in the stack effluents shall be measured semiannually at intervals not to exceed 8 months.

(3) Releases of radioactive liquid waste from the holdup tanks shall be monitored before discharging to the sanitary sewage system to ensure compliance with 10 CFR 20 regulations.

(4) The reactor shall be placed in a reactor shutdown condition whenever Specification 4.2.4(1) is not met.

(5) The reactor vent system shall be immediately secured upon detection of failure of the staff monitoring system.

4.2.5 Surveillance of Experimental Limits

(1) Surveillance to ensure that experiments meet the requirements of Section 3.5 shall be conducted before inserting each experiment into the reactor.

(2) The reactivity worth of an experiment shall be determined at approximately 1 W power level or as appropriate within limiting conditions for operation, before continuing reactor operation with said experiment.

4.2.6 Reactor Building Evacuation Alarm Surveillance

(1) The coincidence automatic actuation of the two area monitors and the manual actuation of the evacuation alarm shall be tested as part of the weekly checkout.

(2) The automatic shutoff of the air conditioning system and the reactor vent system shall be tested as part of the weekly checkout.

(3) Evacuation drills for facility personnel shall be conducted quarterly, at intervals not to exceed 4 months, to ensure that facility personnel are familiar with the emergency plan.

4.2.7 Surveillance Pertaining to Fuel

(1) The incore reactor fuel elements shall be inspected biennially at intervals not to exceed 30 months, in a randomly chosen pattern, as deemed necessary. At least two elements will be inspected.

(2) Fuel-handling tools and procedures shall be reviewed for adequacy before fuel loading operations. The assignment of responsibilities and training of the fuel-handling crew shall be performed according to written procedures.

4.2.8 Primary and Secondary Water Quality Surveillance

(1) The primary water resistivity shall be determined as follows:

(a) Primary water resistivity shall be measured during the weekly checkout by a portable Solu Bridge using approved procedures. The measured value shall be larger than 0.4 megohm-cm.

(b) Primary water resistivity shall be measured during the daily checkout at both the inlet and outlet of the demineralizers (DM). The measured value, determined by an online Solu Bridge alarming in the control room, shall be larger than 0.5 megohm-cm at the outlet of the DM.

(2) The primary water radioactivity shall be measured during the weekly checkout for gross β-γ and gross α activity.

(a) The measured α activity shall not exceed 50 dpm above background level.

(b) The measured β-γ activity shall not exceed 25% above mean normal activity level.

(c) The secondary water system shall be tested for radioactive contamination during the weekly checkout according to written procedures.

5.0 DESIGN FEATURES

Design features are specified to ensure that items important to safety are not changed without appropriate review. The items of concern are design features and parameters that were considered as limiting values (or significant for the protection of the reactor personnel and the general public) for the purpose of establishing safety limits, limiting safety system settings, or limiting conditions for operation.

5.1 Site

The UFTR is located on the University of Florida campus, at Gainesville, Florida, in the immediate vicinity of the buildings housing the College of Engineering and the College of Journalism. The Nuclear Science Center, which houses the Department of Nuclear Engineering, is annexed to the reactor building.

5.2 Reactor Cell

The reactor shall be housed in a reinforced concrete cell in the reactor building. The reactor building is a "vault-type" building as defined in 10 CFR 73.2(o). The reactor building is divided into two distinct parts based upon the difference in utilization and their structure. The overall reactor building measures approximately 60 ft by 80 ft inside. The reactor cell area is 30 ft by 60 ft with 29 ft of head room, located at the north end of the building. The rest of the building is used for research laboratories, faculty offices, and graduate study areas.

The reactor cell shall have an independent ventilation and air-conditioning system. The reactor vent effluents shall be discharged through the reactor stack, some 30 ft above ground level.

All gases that may cause a hazard through neutron activation shall be exhausted from the reactor cell, reactor cavity, experiments or experimental facilities installed in or adjacent to the core or surrounding graphite and discharged to the environment through the reactor vent system and appropriately monitored for radioactivity, as specified under Chapter 3 of these Technical Specifications.

The 3-ton bridge crane shall not be used during reactor operation in a manner that could damage the control system and prevent it from performing its intended function. No load above 500 lb shall be lifted over the control blade drive units unless the control blades are fully inserted. The crane shall be operated during reactor operations only by a licensed reactor operator.

The following doors penetrate the reactor cell: (1) an airlock passageway from the cell to the UFTR building lower hallway, (2) a door frm the control room to the UFTR building lower hallway, and (3) a freight door (10 ft x 12 ft) leading to the environs. A panel in the freight door serves as an emergency personnel exit from the reactor cell. The freight door and panel shall be locked to prevent entrance during reactor operation. The freight door and panel shall not be used for general access to or egress from the reactor cell. This is not meant to preclude use of these doors in connection with authorized activities when the reactor is not in operation.

5.3 Reactor Fuel

Fuel elements shall be of the general MTR type, with thin fuel plates clad with aluminum and containing uranium fuel enriched to no more than about 93% U-235. The fuel matrix may be fabricated by alloying high purity aluminum-uranium alloy or by the powder metallurgy method where the starting ingredients (uranium-aluminum) are in fine powder form. The fuel matrix also may be fabricated from uranium oxide-aluminum (U_3O_8-Al) using the powder metallurgy process. There shall be nominally 14.5 g of U-235 per fuel plate.

The UFTR facility license authorizes the receiving, possession, and use of

(1) up to 4.82 kg of contained uranium-235
(2) a 1-Ci sealed plutonium-beryllium neutron source
(3) an up-to-25-Ci antimony-beryllium neutron source

Other neutron and gamma sources may be used if their use does not constitute an unreviewed safety question pursuant to 10 CFR 50.59 and if the sources meet the criteria established by the Technical Specifications.

5.4 Reactor Core

The core shall contain up to 24 fuel assemblies of 11 plates each. Up to six of these assemblies may be replaced with pairs of partial assemblies. Each partial assembly shall be composed of either all dummy or all fueled plates. A full assembly shall be replaced with no fewer than ten plates in a pair of partial assemblies.

Fuel elements shall conform to these nominal specifications:

Item	Specification
overall size (bundle)	2.845 in. x 2.14 in. x 25.625 in.
clad thickness	0.015 in.
plate thickness	0.070 in.
water channel width	0.137 in.
number of plates	standard fuel element - 11 fueled plates partial element - 5 fueled plates
plate attachment	bolted with spacers
fuel content per plate	14.5 g U-235 nominal

The reactor core shall be loaded so that all fuel assembly positions are occupied.

The fuel assemblies are contained in six aluminum boxes arranged in two parallel rows of three boxes each, separated by about 30 cm of graphite. The fuel boxes are surrounded by a 5 ft x 5 ft x 5 ft reactor grade graphite assembly.

The top of the fuel boxes are covered during operations at power above 1 kW, by the use of the shield plugs and/or gasketed alluminum covers secured to the top of the fuel boxes. The devices function to prevent physical damage of the fuel, to minimize evaporation/leakage of water from the top of the fuel boxes, and to minimize entrapment of argon in the coolant water for radiological protection purposes.

5.5 Reactor Control and Safety Systems

Design features of the components of the reactor control and safety systems that are important to safety, as specified under Section 3.2 of these Technical Specifications, are given below.

5.5.1 Reactor Control System

Reactivity control of the UFTR is provided by four control blades, three safety blades and one regulating blade. The control blades are of the swing-arm type consisting of four aluminum vanes tipped with aluminum, protected by magnesium shrouds. They operate in a vertical arc within the spaces between the fuel boxes. Blade motion is limited to a removal time of at least 100 sec and the insertion time under trip conditions is stipulated to be less than 1 sec. The reactor blade withdrawal interlock system prevents blade motion which will exceed the reactivity addition rate of 0.06% $\Delta k/k$ per sec, as specified in these Technical Specifications. The control blade drive system consists of a two-phase fractional horsepower motor that operates through a reduction gear train, and an electrically energized magnetic clutch that transmits a motor torque through the control blade shaft, allowing motion of the control blades. The blades are sustained in a raised position by means of this motor, acting through the electromagnetic clutch. Interruption of the magnetic current results in a decoupling of the motor drive from the blade drive shaft, causing the blades to fall back into the core. Position indicators, mechanically geared to the rod drives, transmit rod position information to the operator control console. Reactor shutdown also can be accomplished by voiding the moderator/coolant from the core. Two independent means of voiding the moderator/coolant from the core are provided:

(1) water dump via the primary coolant system dump valve opening under full trip conditions

(2) water dump via the rupture disk breaking under pressure conditions above design value

The integral worths of the individual safety blades vary from about 1.3 to 2.3% $\Delta k/k$ depending on position in the core and individual characteristics. The regulating blade worth is about 1% $\Delta k/k$. The rod worths, drive speeds, and drop-time values are sufficiently conservative to ensure compliance with the specified reactivity limitations. Additional reactivity and power related features are obtained from the control blade withdrawal inhibits. The regulating blade may be engaged by a servo-mechanism controlled by the linear channel for automatic reactor power control.

5.5.2 Reactor Safety System

(1) Power Level Channels

Two independent measuring channels are provided for power level limits; both are required for the reactor to be operable. Each channel covers reliably the range from about 1 to 150% of full power (of 100 kW). One channel (Safety 1) is part of the wide range drawer, and receives its main signal from a fission chamber. The Safety 2 channel uses an uncompensated ion chamber for neutron detection. Each channel drops all control rods <u>and</u> the moderator coolant from the core by actuating bistable trips in the safety system in a one-out-of-one trip logic. Visual indication of the power measured by each chamber, as well as annunciator of channel status is available to the operator in the control room.

(2) Wide Range Logarithmic Power Level and Period Channel

The logarithmic power channel covers the wide range from reactor startup to full power in 10 decades. It uses a fission chamber for this entire range and uses a B^{10} proportional counter only in the startup (source) range. Signals from the fission chamber and the B^{10} counter are amplified by a preamplifier before going to the log channel. The preamplifier also processes test signals from the console controls and deenergizes the B^{10} proportional counter at about 400 cps. Power level information is displayed on a meter and on a two-pen recorder. The channel provides the following blade withdrawal inhibits or blade trips: minimum source count inhibit of 2 cps, fast period inhibit of 10 sec, fast period trip of 3 sec, and inhibit limiting power escalation in the automatic mode to no faster than 30 sec, and a trip at or above 1% power when secondary coolant flow is below the trip setting. Because this is a wide range channel, a separate startup channel is not used. These control or limiting actions prevent startup or operation of the reactor unless it is properly monitored or if operational restrictions are not met. Period is displayed on a meter and is effective for control over the entire range of operation.

(3) Startup (Neutron) Source(s)

A permanent, regenerable, antimony-beryllium source of up to 25 Ci and/or a removable plutonium-beryllium source of 1 Ci may be used for reactor startup to monitor the approach to criticality. The use of a neutron source ensures that behavior of the reactor is being monitored by the reactor instrumentation during subcritical control blade manipulations.

(4) Linear Neutron Channel and Automatic Flux Control System

The linear channel is required to be operable when the reactor is to be operated in the automatic mode. The linear channel uses a compensated ion chamber for neutron detection; its signal is transmitted by a multirange pico-ammeter. The pico-ammeter sends a signal to one channel of the two-pen recorder to display power level from source level to full power. It also sends a signal to the automatic flux controller which, in comparison with a signal from a percent of power setting control acts to establish and/or hold power level at a desired value. The rate of power increase is controlled by the action of a limiter in the linear channel/automatic control system which maintains the reactor period at or slower than 30 sec.

5.6 Cooling Systems

5.6.1 Primary Cooling System

The primary coolant is demineralized light water, which is normally circulated in a closed loop. The flow is from the 200-gal storage (dump) tank to the primary coolant pump; water is then pumped through the primary side of the heat exchanger and to the bottom of the fuel boxes, upward past the fuel plates to overflow pipes located about 6 in. above the fuel, and into a header for return to the storage tank. A purification loop is used to maintain primary water quality. The purification loop pump circulates about 1 gpm of primary water, drawn from the discharge side of the heat exchanger, through mixed-bed ion-exchange resins and a ceramic filter. The purification loop pump automatically shuts off when the primary coolant pump is operating, since flow through the purification system is maintained. Primary coolant may be dumped from the reactor fuel boxes by opening an electrically operated solenoid dump valve, which routes the water to the dump tank. A pressure surge of about 2 lb above normal in the system also will result in a water dump by breaking a graphite rupture disc in the dump line. This drains the water to the primary equipment pit floor actuating an alarm in the control room. The primary coolant system is instrumented as follows:

(1) thermocouples at each fuel box and the main inlet and outlet (eight total), alarming and recording in the control room

(2) a flow sensing device in main inlet line, alarming and displayed in the control room

(3) a flow sensing device (no flow condition) in the outlet line, alarming in the control room

(4) resistivity probes monitoring the inlet and outlet reactor coolant flow, alarming and displayed in the control room

(5) an equipment pit water level monitor, alarming in the control room

The reactor power is calibrated annually by the use of the coolant flow and temperature measuring channels.

5.6.2 Secondary Cooling System

Two secondary cooling systems are normally operable in the UFTR: a well secondary cooling system and a city water secondary cooling system. The well secondary cooling system is the main system used for removal of reactor generated heat to the environment. A deep well furnishes about 200 gpm of cooling water to the shell side of the heat exchanger, removing primary heat and rejecting it to the storm sewer. Weekly samples monitor the activity of this water. Flow indications in the control room are 140 gpm as a warning and 60 gpm to initiate a trip at or above 1 kW after a 10-sec warning. The city water secondary cooling system can be used for backup cooling or for specific operations requiring reactor coolant temperatures hotter than those obtained with the well cooling system. The secondary flow by the city water system is about 40 gpm, with a reactor trip set at 8 gpm (as measured by a flow switch) for power levels at or above 1 kW. A back flow preventer in the city water line ensures compliance with the requirements of the National Plumbing Code to prevent contamination of a potable water supply. The secondary coolant system inlet and outlet temperatures are monitored by thermocouples, with alarm and record functions in the control room.

5.7 Radiological Safety Design Features

5.7.1 Physical Features

The containment structure consists of the reactor cell, with a free air volume of about 1600 m^3. This building houses the reactor, reactor control room, the primary cooling system (including the dump tank heat exchanger and purification loop), secondary coolant piping, and reactor vent system. Access to the reactor cell, which is the designated restricted and security area, is controlled by the specifications established by the Physical Security Plan of the UFTR.* Ventilation is through the independent air conditioning ventilation and reactor vent system. The reactor vent system can be secured to prevent uncontrolled discharge of radioactivity to the environment or releases in excess of permissible levels (per 10 CFR 20). Rough and absolute filters are used to eliminate or minimize radioactive air particulate contamination from the exhaust air. The electrically actuated damper in the core exhaust line is fail-safe and closes upon deenergization.

5.7.2 Monitoring System

Area and stack monitors are used for radioactivity monitoring, as delineated in Sections 3.3, 3.4, and 3.6 of these Technical Specifications. The cell air is monitored by an air particulate detector. Exhaust air drawing from the reactor cavity, reactor cell, or experiments is continously monitored for gross concentrations of radioactive gases.

5.7.3 Evacuation Sequence

The emergency evacuation sequence is initiated either automatically by two area monitors alarming high in coincidence or manually by the console operator. The sequence is that the reactor room air conditioning/ventilation system and the reactor vent system are shut down and the core vent damper is closed.

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*Withheld from public disclosure pursuant to 10 CFR 2.790(d).

5.8 Fuel Storage

5.8.1 New Fuel

Unirradiated new fuel elements are stored in a vault-type room security area equipped with intrusion alarms in accordance with the Security Plan. Elements are stored in a steel, fireproof safe in which a cadmium plant separates each layer of bundles to ensure subcriticality under optimum conditions of moderation and reflection.

5.8.2 Irradiated Fuel

Irradiated fuel is stored upright in dry storage pits within the reactor building in criticality-safe holes.

6.0 ADMINISTRATIVE CONTROLS

6.1 Definitions

Certified Operators: An individual authorized by the Nuclear Regulatory Commission to carry out the duties and responsibilities associated with the position requiring the certification.

Class A Reactor Operator: Any individual who is certified to direct the activities of Class B reactor operators; such an individual also is a reactor operator. Such an individual is commonly referred to as a senior reactor operator.

Class B Reactor Operator: Any individual who is certified to manipulate the controls of the reactor. Such an individual is commonly referred to as a reactor operator.

6.2 Organization

6.2.1 Structure

The organization for the management and operation of the reactor facility shall include the structure indicated in Figure 6.1. Job titles are shown for illustration and may vary. Four levels of authority are provided.

Level 1 - individuals responsible for the reactor facility's licenses, charter, and site administration

Level 2 - individual responsible for reactor facility management

Level 3 - individual responsible for reactor operations and supervision of day-to-day facility activities

Level 4 - reactor operating staff (Class A and B reactor operators and trainees)

The Reactor Safety Review Subcommittee is appointed by, and shall report to, the Chairman of the Radiation Control Committee. The Chairman of the Radiation Control Committee reports to the Director of Environmental Health and Safety, who reports to the Vice-President for Administrative Affairs. Radiation safety personnel shall report to Level 2 or higher.

6.2.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Figure 6.1. Individuals at various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering

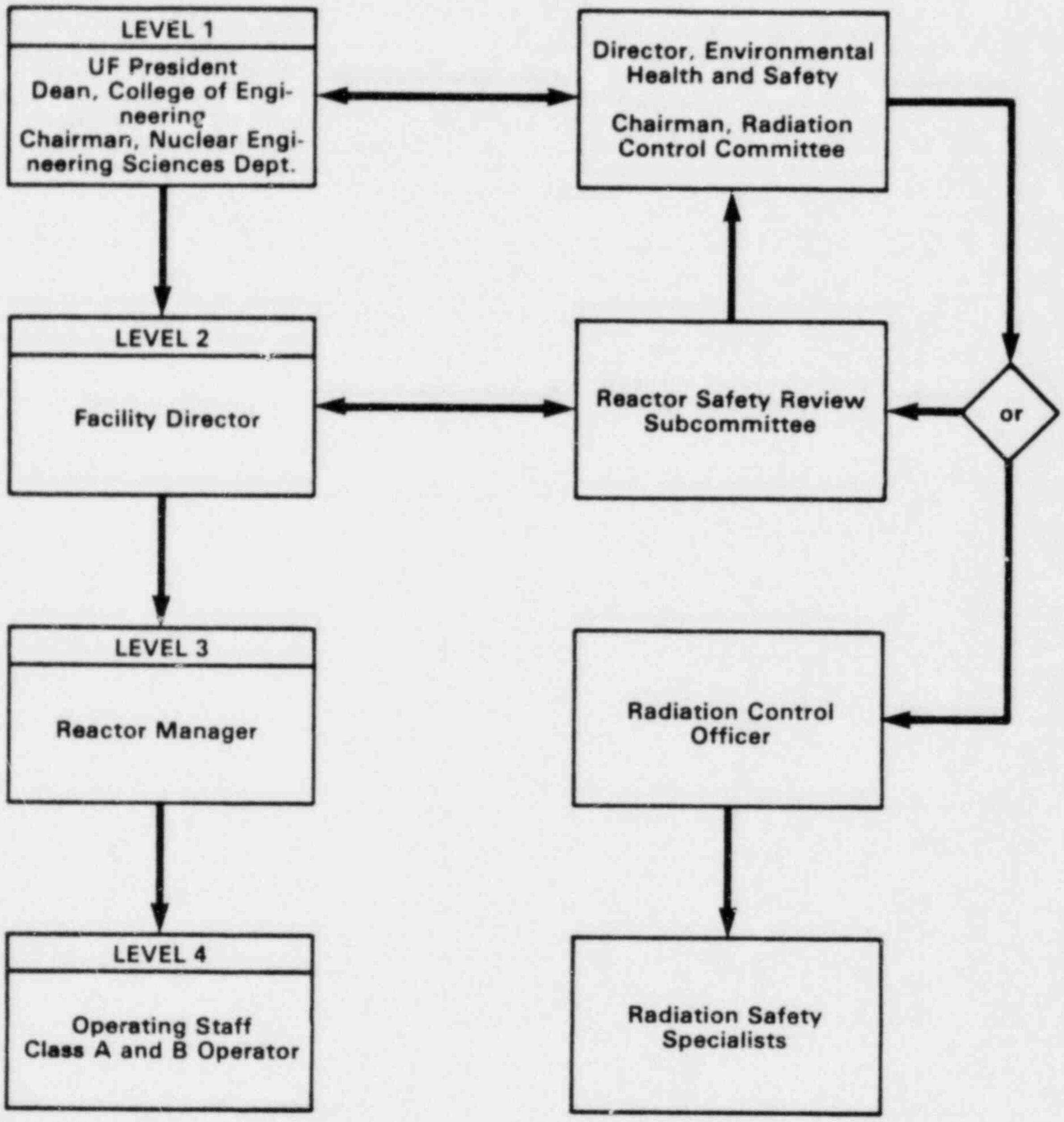

Figure 6.1 UFTR organizational chart

to all requirements of the operating license, charter, and technical specification. In all instances, responsbilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

6.2.3 Staffing

The minimum staffing when the reactor is not secured shall be as follows:

(1) A certified reactor operator shall be in the control room.

(2) A second person shall be present at the facility complex able to carry out prescribed written instructions including instructions to initiate the first stages of the emergency plan, including evacuation and initial notification procedures. Unexpected absence for two hours is acceptable provided immediate action is taken to obtain a replacement.

(3) A designated Class A Reactor Operator shall be readily available on call. "Readily Available on Call" means an individual who (a) has been specifically designated and the designation known to the operator on duty, (b) keeps the operator on duty informed of where he/she may be rapidly contacted and the phone number <u>or</u> other means of communication available, and (c) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 min or within a 15 mi radius).

A list of reactor facility personnel by name and telephone number shall be readily available in the Control Room for use by the operator. The list shall include

(1) management personnel
(2) radiation safety personnel
(3) other operations personnel

Events requiring the direction of Class A reactor operator are

(1) all fuel or control-rod relocations within the reactor core region

(2) relocation of any incore experiment with a reactivity worth greater than one dollar

(3) recovery from unplanned or unscheduled shutdown (in this instance, documented verbal concurrence from Class A operator is required)

6.2.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1977, Sections 4.6.

6.2.5 Review and Audit

A method for the independent review and audit of the safety aspects of reactor facility operations shall be established to advise management. The review and audit functions of the UFTR operations are conducted by the Reactor Safety Review Subcommittee (RSRS).

(1) Composition and Qualifications

The RSRS shall be composed of a minimum of five members, including the reactor manager and radiation control officer (both ex-officio voting members), the Chairman of the Nuclear Engineering Sciences Department and two other members having expertise in reactor technology and/or radiological safety.

(2) Charter and Rules

The review and audit functions shall be conducted in accordance with the following established charter:

<u>Designation</u> - The name of the Subcommittee is Reactor Safety Review Subcommittee (RSRS).

<u>Accountability</u> - The RSRS is a Subcommittee of and reports to the University Radiation Control Committee (URCC). The URCC provides radiological safety recommendations to the Director of Environmental Health and Safety.

<u>Scope</u> - The RSRS shall be responsbile for the review of safety-related issues pertaining to the University of Florida Training Reactor (UFTR).

<u>Purpose</u> - The purpose of the RSRS is to ensure the safe operation of the UFTR through the discharge of the Submcommittee review and audit functions.

<u>Membership</u> -

(a) The RSRS shall consist of at least five members. Membership will include the Chairman of the Nuclear Engineering Sciences Department, University Radiation Control Officer, Reactor Manager and two technical personnel familiar with the operation of reactors and with the design of the UFTR and radiological safety, at least one of whom is from outside the Department of Nuclear Engineering Sciences. The two technical personnel will be recommeded to the Chairman of the URCC by the Chairman of the Department of Nuclear Engineering Sciences. Any member may designate a duly qualified representative from a standing URCC approved list to act in his absence.

(b) An Executive RSRS Committee will consist of the Reactor Manager, University Radiation Control Officer, and Chairman of the RSRS.

(c) The Chairman of the RSRS will be appointed by the Chairman of the URCC. The Chairman of the RSRS is an ex-officio voting member of the URCC and will serve as liaison between the RSRS and the URCC.

(d) Members appointed to the RSRS shall be reviewed, and as appropriate, new appointments made by October 1 of each calendar year.

Meetings -

(a) At least one meeting shall be held quarterly at intervals not to exceed 4 months. Meetings may be held more frequently as circumstances warrant, consistent with the effective monitoring of facility operations as determined by the RSRS Chairman.

(b) Review of draft minutes will be completed before subsequent meetings, at which time they will be submitted for approval. Responsibility to ensure that this is done falls upon the RSRS Chairman. The RSRS Chairman is charged with the responsibility to assure that the minutes are submitted for approval in a timely manner.

(c) A quorum shall consist of at least three members and at least three members must agree when voting, regardless of the number present.

(3) Review Function

The following items shall be reviewed:

(a) determination that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question

(b) all new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment or systems having safety significance

(c) all new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity

(d) proposed changes in technical specifications, license, or charter

(e) violations of technical specifications, license, or charter

(f) violations of internal procedures or instructions having safety significance

(g) operating abnormalities having safety significance

(h) reportable occurrences

(i) audit reports and annual facility reports

A written report or minutes of the findings and recommendations of the review group shall be submitted to RSRS members in a timely manner after the review has been completed and to the Chairman of the Radiation Control Committee whenever a finding is deemed to require review by Level 1.

(4) Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Where necessary, discussions with cognizant personnel shall take place. In no case shall the individual immediately reponsible for the area, audit in the area. The following items shall be audited:

(a) facility operations for conformance to the technical specifications and applicable license or charter conditions, at least once per calendar year (interval between audits not to exceed 15 months)

(b) the retraining and requalification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months)

(c) the results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months)

(d) the reactor facility emergency plan, and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months)

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Radiation Control Committee and the Dean of the College of Engineering. A written report of the findings of the audit shall be submitted to the Dean of the College and the review and audit group members within 3 months after the audit has been completed.

6.3 Procedures

The facility shall be operated and maintained in accordance with approved written procedures. All procedures and major revisions thereto shall be reviewed and approved by the Director of Nuclear Facilities before going into effect.

The following types of written procedures shall be maintained:

(1) normal startup, operation and shutdown procedures for the reactor (These procedures shall include applicable checkoff lists and instructions.)

(2) fuel loading, unloading, and movement within the reactor

(3) procedures for handling irradiated and unirradiated fuel elements

(4) routine maintenance of major components of systems that could have an effect on reactor safety

(5) surveillance tests and calibrations required by the technical specifications or those that may have an effect on reactor safety

(6) personnel radiation protection, consistent with applicable regulations

(7) administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity

(8) implementation of the Emergency Plan

(9) procedures that delineate the operator action required in the event of specific malfunctions and emergencies

(10) procedures for flooding conditions in the reactor facility, including guidance as to when the procedure is to be initiated and guidance on reactivity control

Substantive changes to the above procedures shall be made effective only after documented review by the RSRS and approval by the facility director (Level 2) or designated alternatives. Minor modifications to the original procedures which do not change their original intent may be made by the reactor manager (Level 3) or higher, but modifications must be approved by Level 2 or designated alternates within 14 days. Temporary deviations from the procedures may be made by the senior operating individual present, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to Level 2 or designated alternates.

6.4 Experiments Review and Approval

(1) Experiments review and approval shall be conducted as specified under Section 3.5, "Limitations on Experiments," of these Technical Specifications.

(2) The experiments review and approval shall ensure compliance with the requirements of the license, Technical Specifications, and applicable regulations and shall be documented.

(3) Substantive changes to previously approved experiments with safety significance shall be made only after review by the RSRS, approval in writing by Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

(4) Approved experiments shall be carried out in accordance with established approved procedures.

6.5 Required Actions

6.5.1 Action To Be Taken in Case of Safety Limit Violation

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

(2) The safety limit violation shall be promptly reported to Level 2 or designated alternates.

(3) The safety limit violation shall be reported to the Nuclear Regulatory Commission.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

(a) applicable circumstances leading to the violation including, when known, the cause and contributing factors

(b) effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public

(c) corrective action to be taken to prevent recurrence

The report shall be reviewed by the RSRS and any followup report shall be submitted to the Commission when authorization is sought to resume operation of the reactor.

6.5.2 Action To Be Taken in the Event of an Occurrence of the Type Identified in Section 6.6.2

(1) Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by Level 2 or designated alternates.

(2) Occurrence shall be reported to Level 2 or designated alternates and to the Commission as required.

(3) Occurrence shall be reviewed by the review group at their next scheduled meeting.

6.6 Reports

In addition to the requirements of the applicable regulations, reports shall be made to the Commission as follows:

6.6.1 Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted to the Commission within 3 months following the end of each prescribed year. The prescribed year ends August 31 for the UFTR. Each annual operating report shall include the following information:

(1) a narrative summary of reactor operating experience including the energy produced by the reactor and the hours the reactor was critical

(2) the unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence

(3) tabulation of major preventive and corrective maintenance operations having safety significance

(4) tabulation of major changes in the reactor facility and procedures, and tabulation of new tests or experiments, that are significantly different from those performed previously and are not described in the Safety Analysis Report, including conclusions that no unreviewed safety questions were involved

(5) a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the facility operators as determined at or before the point of such release or discharge (The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient.)

(6) a summarized result of environmental surveys performed outside the facility

(7) a summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed

The annual report shall be submitted to the Director, Division of Licensing, U.S. NRC, Washington, DC 20555 and to the Director, NRC Region II, Inspection and Enforcement Office, Atlanta, GA.

6.6.2 Special Reports

There shall be a report not later than the following working day by telephone and confirmed in writing by telegraph or similar conveyance to the Commission, to be followed by a written report that describes the circumstances of the event within 14 days of any of the following:

(1) release of radioactivity from the site above allowed limits

(2) violation of safety limits (see Section 6.5.1)

(3) any of the following:

(a) operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications

(b) operation in violation of limiting conditions for operation established in the Technical Specifications unless prompt remedial action is taken

(c) a reactor safety system component malfunction that renders the reactor safety system incapable of performing its intended safety function, unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns (Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

(d) an unanticipated or uncontrolled change in reactivity greater than one dollar (Reactor trips resulting from a known cause are excluded.)

(e) abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or containment boundary (excluding minor leaks), where applicable, which could result in exceeding prescribed radiation exposure limits of personnel or environment or both

(f) an observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations

(g) a violation of the Technical Specifications or the facility license

6.7 Records

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single, or multiple records, or a combination thereof. Recorder charts showing operating parameters of the reactor (i.e., power level, temperature, etc.) for unscheduled shutdown and significant unplanned transients shall be maintained for a minimum period of 2 years.

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

(1) normal reactor facility operation (Supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least 1 year.)

(2) principal maintenance operations

(3) reportable occurrences

(4) surveillance activities required by the Technical Specifications

(5) reactor facility radiation and contamination surveys where required by applicable regulations

(6) experiments performed with the reactor

(7) fuel inventories, receipts, and shipments

(8) approved changes in operating procedures

(9) records of meetings and audit reports of the RSRS

6.7.2 Records To Be Retained for at Least One Training Cycle

Records of the most recent complete cycle of retraining and requalification of certified operations personnel shall be maintained at all times the individual is employed.

6.7.3 Records To Be Retained for the Lifetime of the Reactor Facility*

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

(2) offsite environmental monitoring surveys required by the Technical Specifications

(3) radiation exposure for all personnel monitored

(4) updated drawings of the reactor facility

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*Applicable annual reports, if they contain all of the required information, may be used as records in this section.

7.0 AS LOW AS IS REASONABLY ACHIEVABLE (ALARA) (10 CFR 50.36a)

The principal routine emission from the UFTR facility complex is argon-41 discharged by the reactor vent system. There is no known biological uptake of argon-41 and exposure limits are based upon external, total body irradiation.

The concentration of argon-41 in the stack effluent is continuously monitored when the reactor is operating, and is normally less than 1×10^{-5} μCi/ml after several hours of full power operation. The annual release is related to the number of equivalent hours of 100 kW operation (kWt per year). Reactor operations are limited by prior agreement, and by these Technical Specifications, to the limit the argon-41 discharges to maximum permissible concentration (MPC) when averaged over a month and using the established atmospheric dilution factor of 200.

The offsite environmental radioactive surveillance program has proven that exposure to the general public from the reactor radioactive effluents approaches consistently the nondetectable level and certainly is always well below the 500 mrems/yr federal limit.

The ALARA program at the UFTR minimizes unnecessary production of radioactive effluents by selectivity of operations. The potential reduction of argon-41 releases is frequently reviewed, and is a major item of consideration during the upcoming reviews to upgrade facility operations to 500 kWt. A reduction of the vent flow as well as the argon dissolving in the primary coolant is being proposed, as well as the possibility of utilizing storage tanks.

Radioactive liquid effluents and personnel radioactive exposure are well within ALARA guidelines.