Appendix A

Technical Specifications

for the

North Carolina State University

PULSTAR Reactor

Facility License No. R-120

Docket No. 50-297

< date – DRAFT Rev 2>

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FIGURES

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1 INTRODUCTION

1.1 Scope

These Technical Specifications provide limits within which operation of the reactor will assure the health and safety of the public, the environment and on-site personnel. Areas addressed are Definitions, Safety Limits (SL), Limiting Safety System Settings (LSSS), Limiting Conditions for Operation (LCO), Surveillance Requirements, Design Features, and Administrative Controls.

1.2 Purpose

These technical specifications represent an agreement between North Carolina State University and the Nuclear Regulatory Commission on administrative controls, equipment availability, and operational parameters for the PULSTAR Nuclear Reactor.

Individual specifications in Sections 2, 3, and 4 shall include the following information in the format shown:

Applicability: This is a statement that indicates which components are involved and when they are involved;

Objective: This is a statement that indicates the purpose of the specification(s);

Specification(s): This statement provides specific data, conditions, or limitations that bound a system or operation. This statement is the most important statement in the technical specifications;

Basis: The basis is a statement that provides the background or reason for the choice of specifications(s), or references a particular portion of the Safety Analysis Report that does.

Although each of the preceding statements provides important information, only the "Specifications(s)" and "applicability" statement govern.

1.3 Definitions

1.3.1 **Channel**: A channel is the combination of sensor, line, amplifier, and output devices which are connected for the purpose of measuring the value of a parameter.

1.3.2 **Channel Calibration**: A channel calibration is an adjustment of the channel, such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include a Channel Test.

1.3.3 **Channel Check**: A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same variable.

1.3.4 **Channel Test**: A channel test is the introduction of a signal into the channel for verification that it is operable.

1.3.5 **Confinement**: Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

1.3.6 **Control Rod**: A control rod is a device fabricated from neutron absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged, i.e. has scram capability.

1.3.7 **Core Configuration:** The core configuration describes a particular arrangement of fuel assemblies, reflectors, and control rods occupying the core grid.

1.3.8 **Damaged Fuel:** The term damaged fuel means that a deterioration of the fuel cladding is present that results in the release of fission products.

1.3.9 **Excess Reactivity**: Excess reactivity is that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff}=1) at reference core conditions.

1.3.10 **Experiment**: Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beam tube or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment. Specific categories of experiments include:

a. **Tried Experiment**: A tried experiment is an experiment that is within the bounds of an experiment review for an approved experiment that has been previously performed at this reactor

b. **Untried Experiment**: An untried experiment is an experiment that is not within the bounds of an experiment review for an approved experiment that has been previously performed at this reactor.

c. **Secured Experiment**: A secured experiment is any experiment, experimental 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 which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

d. **Non-Secured Experiment**: A non-secured experiment is an experiment that does not meet the criteria for being a "secured" experiment.

e. **Movable Experiment**: A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

f. **Fueled Experiment**: A fueled experiment is an experiment which involves any of the following:

i. Neutron irradiation of uranium exceeding 2.0 x 10^6 fissions per second.

ii. Neutron irradiation of any amount of other fissionable material.

iii. A planned release of fission gases or halogens.

Fueled Experiments exclude:

iv. Fissionable material not subjected to neutron fluence.

v. Detectors containing fissionable material used in the operation of the reactor or used in an experiment, sealed sources, and fuel used in operation of the reactor.

Examples of excluded materials include manufactured detectors, sealed sources (i.e., sources encased in a capsule designed to prevent leakage or escape of the material from the intended use of the source or potential minor mishaps) with registration certificates, special form radioactive material as defined in 10 CFR Part 71, and PULSTAR reactor fuel elements in cladding.

1.3.11 Experimental Facilities: Experimental facilities are facilities used to perform experiments. They include beam tubes, thermal columns, void tanks, pneumatic transfer systems, in-core facilities at single-assembly positions, out-of-core irradiation facilities, and the bulk irradiation facility.

1.3.12 Fissionable Material: A nuclide that is capable of undergoing fission after capturing either high-energy (fast) neutrons or low-energy thermal (slow) neutrons. Although formerly used as a synonym for fissile material, fissionable materials also include those (such as uranium-238)

that can be fissioned only with high-energy neutrons. As a result, fissile materials (such as uranium-235) are a subset of fissionable materials.

1.3.13 License: The written authorization by the Nuclear Regulatory Commission for North Carolina State University to carry out the duties and responsibilities associated with the PULSTAR Nuclear Reactor.

1.3.14 Limiting Condition for Operation: Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility (10 CFR Part 50.36).

1.3.15 Limiting Safety System Setting: Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded

(10 CFR Part 50.36).

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

1.3.17 Operable: Operable means a component or system is capable of performing its intended function.

1.3.18 Operating: Operating means a component or system is performing its intended function.

1.3.19 pcm: A unit of reactivity that is the abbreviation for "percent millirho" and is equal to 10^{-5} $\Delta k/k$ reactivity. For example, 1000 pcm is equal to 1.0% $\Delta k/k$.

1.3.20 Protective Action: Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit.

1.3.21 Reactor Building: The Reactor Building includes the Reactor Bay, Control Room and Ventilation Equipment Room, the Mechanical Equipment Room (MER), and the Primary Piping Vault (PPV). The Nuclear Regulatory Commission R-120 license applies to the areas in the Reactor Building and the Waste Tank Vault.

1.3.22 Reactor Operation: Reactor operation is any condition when the reactor is not secured or shutdown.

1.3.23 Reactor Operator: A reactor operator (RO) is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the facility.

1.3.24 Reactor Operator Assistant (ROA): A reactor operator assistant (ROA) is an individual who has successfully completed an in-house training program to assist the licensed reactor operator with non-licensed activities during reactor operation.

1.3.25 Reactor Safety System: Reactor safety systems are those systems,

including their associated input channels, that are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.3.26 Reactor Secured: The reactor is secured when:

a. Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection, **or**

b. The following conditions exist:

i. All scrammable neutron absorbing control rods are fully inserted, **and**

ii. The reactor key switch is in the OFF position and the key is removed from the lock, **and**

iii. 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**

iv. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding one dollar.

1.3.27 Reactor Shutdown: The reactor is shut down if it is subcritical by at least one dollar both in the reference core condition and for all allowed ambient conditions with the reactivity worth of all installed experiments included.

1.3.28 Reference Core Condition: The reference core condition is the reactivity condition of the core when it is 70°F and the reactivity worth of xenon is zero, (i.e., cold, clean, critical).

1.3.29 Reportable Event: A Reportable Event is any of the following:

a. Violation of a Safety Limit.

b. Release of radioactivity from the site above allowed limits.

c. Operation with actual Safety System Settings (SSS) for required systems less conservative than the Limiting Safety System Settings (LSSS) specified in these technical specifications.

d. Operation in violation of Limiting Conditions for Operation (LCO) established in these technical specifications unless prompt remedial action is taken as permitted in Section 3.

e. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is caused during maintenance tests or periods of reactor shutdown. 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 numbers of components or systems specified or required perform their

intended reactor safety function.

f. An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded.

g. Abnormal or significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks).

h. 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.

1.3.30 Safety Limit: 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 (10 CFR Part 50.36).

1.3.31 Scram Time: Scram time, also referred to as drop time, is the time interval measured between the initiation of a scram signal and the instant of the rod seated signal.

1.3.32 Shall, Should, and May: The word “shall” is used to denote a requirement; the word “should” is used to denote a recommendation; and the word “may” is used to denote permission, neither a requirement nor a recommendation.

1.3.33 Shim Rod: A shim rod is a neutron absorbing rod having an in-line drive which is mechanically, rather than magnetically, coupled and does not have a SCRAM capability.

1.3.34 Senior Reactor Operator: A senior reactor operator (SRO) is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the facility and to direct the activities of licensed reactor operators. Such an individual is also a reactor operator.

1.3.35 Shutdown Margin: Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition. It shall be assumed that the most reactive scrammable rod and all non-scrammable rods are in their most reactive position and that the reactor will remain subcritical without further operator action.

1.3.36 Shutdown Reactivity: Shutdown reactivity is the value of the reactivity of the reactor with all control rods in their least reactive positions (e.g., inserted). The value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions

1.3.37 Total Nuclear Peaking Factor: The ratio of the local power density in the fuel pin to the average power density of the core.

1.3.38 True Value: The true value is the actual value of a parameter.

1.3.39 University Management: University Management is the Chancellor, Office of the Chancellor, and other University Administrator(s) having authority designated by the Chancellor or as specified in University policies.

1.3.40 Unscheduled Shutdown: An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation not including shutdowns that occur during testing or check-out operations.

2 SAFETY LIMITS AND LIMITING SAFETYSYSTEM SETTINGS

2.1 Safety Limits (SL)

Applicability

This specification applies to the fuel cladding temperature.

Objective

The objective is to ensure that the integrity of the fuel cladding is maintained.

Specification

Under all conditions, the Safety Limit shall be as follows:

a. The true value of the fuel cladding temperature shall not exceed 2200°F.

Bases

The NRC has found that a fuel cladding temperature limit of 2200°F for zircaloy-2 clad PULSTAR type fuel to be acceptable.

2.2 Limiting Safety System Settings

2.2.1 Limiting Safety System Settings (LSSS) for Forced Convection Flow

Applicability

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), height of water above the top of the core (H), and pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specification

Under the condition of forced convection flow, the Limiting Safety System Settings shall be as follows:

P	2.0 MWt (max.)
W	900 gpm (min.)
H	117 feet (min.)
T	117°F

Bases

The Limiting Safety System Settings that are given in the Specification 2.2.1 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded during the most limiting anticipated transient (loss of flow). The safety margin that is provided between the Limiting Safety System Settings and the Safety Limits also allows for the most adverse combination of instrument uncertainties associated with measuring the observable parameters.

The analysis presented in the Safety Analysis Report for all credible accident scenarios indicate that if the interrelated variables were at their LSSS at the initiation of the transient, the Safety Limit specified in 2.1 would not be exceeded.

2.1.1 Limiting Safety System Settings (LSSS) for Natural Convection Flow

Applicability

This specification applies to the setpoints for the safety channel monitoring reactor thermal power (P), the height of water above the core (H), and the pool water temperature (T).

Objective

The objective is to assure that automatic protective action is initiated in order to prevent a Safety Limit from being exceeded.

Specifications

Under the condition of natural convection flow, the Limiting Safety System Settings shall be as follows:

P	250 kWt (max.)
H	17 feet (min.)
T	117°F

Bases

The Limiting Safety System Settings that are given in Specification 2.2.2 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent the Safety Limit from being exceeded.

The specifications given above assure that an adequate safety margin exists between the LSSS and the SL. The limit on reactor thermal power was chosen with the additional consideration related to bulk boiling at the outlet of the hot channel. This criterion is not related to fuel clad damage (for these relatively low power levels) but to minimize ^{16}N dose at the pool surface which might be aided by steam bubble rise during up-flow in natural convection. Analysis of coolant bulk boiling given in the Safety Analysis Report indicates that the large safety margin on reactor thermal power assumed in Specification 2.2.2 above will satisfy this additional criterion of no bulk boiling in any channel.

3 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Fuel and Core Configuration

Applicability

This specification applies to the reactor core configuration during forced convection or natural convection flow operations.

Objective

The objective is to ensure that the reactor will be operated within the bounds of established safety analysis.

Specification

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

a. A maximum of twenty-five fuel assemblies.

b. Any number of reflector assemblies of either graphite or beryllium or a combination of these located on the core periphery.

c. Unoccupied grid plate penetrations plugged.

d. A minimum of four control rod guides are in place with operable control rods.

e. The worth of a fuel assembly while being loaded into the reactor grid plate shall not result in a k_{eff} of greater than 1.01626.

f. The total pin power peaking factor in any fuel assembly shall not exceed 3.0.

g. Operation with damaged fuel is not permitted except as may be necessary to locate such fuel.

Bases

Specifications 3.1.a through 3.1.d require that the core be configured such that there is no bypass cooling flow around the fuel through the grid plate.

Specifications 3.1.d requires control rods are operable to ensure that shutdown margin requirements are satisfied.

Specification 3.1.e provides assurances that a fuel loading accident will not result in fuel cladding failure as discussed in the Safety Analysis Report.

Specification 3.1.f provides assurances that fuel integrity is maintained as discussed in the Safety Analysis Report.

Specification 3.1.g provides assurances that routine reactor operation does not occur with damaged fuel.

3.2 Reactivity

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments.

Objective

The objective is to ensure that the reactor can be shutdown and remain shutdown at any time and that the Safety Limit will not be exceeded.

Specifications

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

a. The shutdown margin, with the highest worth scrammable control rod fully withdrawn, and with experiments at their most reactive condition, relative to the reference core condition, is greater than 400 pcm.

b. The excess reactivity is not greater than 4800 pcm.

c. The scram time of each control rod is not greater than 1.0 second.

d. The rate of reactivity insertion of the control rods is not greater than 200 pcm per second (critical region only).

e. The absolute reactivity worth of experiments or their rate of reactivity change shall not exceed the values indicated in Table 3-1.

f. The sum of the absolute values of the reactivity worths of all experiments shall not be greater than 3000.

Table 3-1 - Reactivity Limits for Experiments

Experiment	Limit
Movable	300 pcm or 200 pcm/sec, whichever is more limiting
Non-secured	1000 pcm
Secured	1600 pcm

Bases

The shutdown margin required by Specification 3.2.a ensures that the reactor can be shut down from any operating condition and will remain shutdown after cool down and xenon decay, even if the highest worth control rod should be in the fully withdrawn position.

The upper limit on excess reactivity by Specification 3.2.b ensures that an adequate shutdown margin is maintained.

Scram time, also referred to as drop time, required by Specification 3.2.c is the time interval measured between the initiation of a scram signal and the instant of the rod seated signal.

The maximum rate of reactivity insertion by the control rods which is allowed by

Specification 3.2.d ensures that the Safety Limit will not be exceeded during a startup accident (linear ramp reactivity insertion) as analyzed in the Safety Analysis Report.

Specification 3.2.e is intended to prevent inadvertent reactivity changes during reactor operation caused by the insertion or removal of an experiment. It further provides assurance that the failure of a single experiment will not result in a reactivity insertion which could result in consequences greater than those analyzed in the Safety Analysis Report.

Specification 3.2.f limits the total reactivity associated with experiments to ensure that an adequate shutdown margin is maintained.

3.3 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to require the minimum number of reactor safety system channels which must be operable during reactor operation.

Specification

The reactor shall not be operated unless all the reactor safety system channels described in Table 3.2 are operable.

Table 3-2 - Required Safety and Safety Related Channels

	Measuring Channel	Function
a.	Startup Power Level(1)	Inhibits Control Rod withdrawal when neutron count is ≤2 cps
b.	Safety Power Level	SCRAM at ≤ 2.0 MW (LSSS) Enable for Flow/Flapper SCRAMs at ≤250 kW (LSSS)
c.	Linear Power Level	SCRAM at ≤ 2.0 MW (LSSS)
d.	Log N Power Level	Enable for Flow/Flapper SCRAMs at ≤250 kW (LSSS)
e.	Flow Monitoring(2)	SCRAM when flapper not closed and Flow/Flapper SCRAMs are enabled
f.	Primary Coolant Flow(2)	SCRAM at ≥ 900 gpm (LSSS) when Flow/Flapper SCRAMs are enabled
g.	Pool Water Temperature Monitoring Switch	ALARM at ≤117°F
h.	Pool Water Temperature Measuring Channel	SCRAM at ≤117°F (LSSS)
i.	Pool Water Level	SCRAM at ≥ 17 feet
j.	Manual SCRAM Button	SCRAM
k.	Reactor Key Switch	SCRAM
l.	Scram Logic Unit Ground Fault Scram	SCRAM
m.	Over-the-Pool Radiation Monitor(3)	Alarm (100 mR/hr)

(1) Required only for reactor startup when power level is less than 4 watts.

(2) Either the Flapper SCRAM or the Flow SCRAM may be bypassed during maintenance testing and/or performance of a startup checklist in order to verify each SCRAM is independently operable. The reactor must be shutdown in order to use these bypasses.

(3) The automatic initiation of the evacuation and confinement system for the Over-the-Pool Monitor may be bypassed for less than five minutes during the return of a pneumatic capsule from the core to the unloading station or during the removal of experiments from the reactor pool.

Bases

The Startup Channel inhibit function ensures that a minimum source multiplication count rate level is being detected to ensure adequate information is available to the operator.

The reactor power level SCRAMs provide the redundant protection channels to ensure that, if a condition should develop which would tend to cause the reactor to operate at an abnormally high power level, an immediate automatic protective action will occur to prevent exceeding the Safety Limit.

The primary coolant flow SCRAMs provide redundant channels to ensure when the reactor is at power levels which require forced flow cooling that, if sufficient flow is not present, an immediate automatic shutdown of the reactor will occur to prevent exceeding the Safety Limit. The Log N Power Channel is included in this section since it is one of the two channels which enables the two flow SCRAMs when the reactor is above 250 kW (LSSS).

The pool water temperature channel provides for shutdown of the reactor and prevents exceeding the Safety Limit due to high pool water temperature.

The pool water level channel together with the Over-the-Pool (Bridge) radiation monitor, provides two diverse channels for shutdown of the reactor and prevents exceeding the Safety Limit due to insufficient pool height.

To prevent unnecessary initiation of the evacuation and confinement systems during the return of the pneumatic capsule from the core to the unloading station or during the removal of experiments from the reactor pool, the automatic initiation of the evacuation and confinement system for the Over-the-Pool monitor may be bypassed for the specified time interval.

The manual SCRAM button and the Reactor Key switch provide two manual SCRAM methods to the reactor operator if unsafe or abnormal conditions should occur.

3.4 Reactor Instrumentation

Applicability

This specification applies to the instrumentation that shall be available to the reactor operator to support the safe operation of the reactor, but are not considered reactor safety systems.

Objective

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

Specification

The reactor shall not be operated unless the following are operable:

a. ^{16}N Power Measuring Channel when reactor power is greater than 500kW.

b. Control Rod Position Indications for each control rod.

c. Differential pressure gauge for "Bay with Respect to Atmosphere".

Bases

The ^{16}N Channel provides the necessary power level information to allow adjustment of Safety and Linear Power Channels.

Control rod position indications give the operator information on rod height necessary to verify shutdown margin.

The differential pressure gauge provides the pressure difference between the Reactor Bay and the outside ambient and confirms air flow in the ventilation system for both normal and confinement modes.

3.5 Radiation Monitoring Equipment

Applicability

This specification applies to the availability of radiation monitoring equipment which shall be operable during reactor operation.

Objective

To ensure that radiation monitoring equipment is available for evaluation of radiation conditions in restricted and unrestricted areas.

Specification

The reactor shall not be operated nor shall irradiated fuel or irradiated fueled experiments be moved within the reactor building unless the radiation monitoring equipment listed below and in Table 3-3 are operable.(1)(2)(3)(7)

a. Three fixed area monitors operating in the Reactor Building with setpoints as listed in Table 3-3.(1)(3)(4)

b. Stack Gas and Stack Particulate building monitors continuously sampling air in the facility exhaust stack with setpoints listed in Table 3-3.(1)(3)(4)

c. Radiation Rack Recorder.(5)

d. Vented fueled experiment exhaust gas monitor continuously monitoring the experiment exhaust gas.(7)(8)

e. Vented fueled experiment flow rate monitor.(7)

Table 3-3 - Required Radiation Area Monitors

Monitor	Alert Setpoint	Alarm Setpoint
Control Room	≤ 2 mR/hr	≤ 5 mR/hr
Over-the-Pool	≤ 5 mR/hr	≤ 100 mR/hr
West Wall	≤ 5 mR/hr	≤ 100 mR/hr
Stack Gas	≤ 4400 Ar-41 EC (6)(9)	≤ 6600 Ar-41 EC (6)(9)
Stack Particulate	≤ 8800 Co-60 EC (6)(9)	≤ 13,200 Co-60 EC (6)(9)

(1) For periods of time, not to exceed ninety days, for maintenance to the radiation monitoring channel, the intent of this specification will be satisfied if one of the installed channels is replaced with a gamma-sensitive instrument which has its own alarm audible or observable in the control room.

(2) The automatic initiation of the evacuation and confinement system for the Over-the-Pool Monitor may be bypassed for less than five minutes during return of a pneumatic capsule from the core to the unloading station or during the removal of experiments from the reactor pool.

(3) Stack Gas and Stack Particulate monitor setpoints are based on the effluent concentration (EC) quantities present in the ventilation flow stream as it exits the stack. Refer to the Safety Analysis Report for setpoint bases for the radiation

monitoring equipment.

(4) The automatic initiation of the evacuation and confinement system for the Stack Gas and Stack Particulate Monitors may be bypassed for less than five minute immediately after starting the pneumatic blower system.

(5) During repair and/or maintenance of the recorder not to exceed 90 days, the specified area and effluent monitor readings shall be recorded manually at a nominal interval of 30 minutes when the reactor is not shutdown.

(6) EC values from 10 CFR Part 20 Appendix B, Table 2.

(7) Monitor for vented fueled experiments are only required to be operable while the experiment is in operation.

(8) Vented fueled experiment exhaust radiation monitor setpoint meets Specification 3.8.d.ii. and isolates experiment exhaust when exceeded.

(9) Monitor setpoints are adjusted for vented fueled experiments while a vented fueled experiment is in operation.

Bases

A continuous evaluation of the radiation levels within the Reactor Building will be made to ensure the safety of personnel. This is accomplished by the area monitoring system of the type described in the Safety Analysis Report.

Evaluation of the continued discharge air to the environment will be made using the information recorded from the stack gas and stack particulate monitors.

When the radiation levels reach the alarm setpoint on any single area or stack exhaust monitor, the building will be automatically placed in confinement as described in the Safety Analysis Report.

To prevent unnecessary initiation of the evacuation confinement system during the return of a pneumatic capsule or during the removal of experiments or radioactive material from the reactor pool, the Over-the-Pool Monitor may be bypassed during the specified time interval.

Stack gas and stack particulate setpoints are based on the Notification of Unusual Event Emergency Action Level (EAL) for Ar-41 and Co-60, respectively, during normal operation with no vented fueled experiments being performed.

Radiation dose for EAL are higher than those for fueled experiments. While vented fueled experiments are performed, the stack gas and stack particulate monitor setpoints are changed to meet the radiation dose criteria given in Specification 3.8.d.ii. Accidental releases from fueled experiments have higher setpoints and therefore those listed in Table 3-3 are conservative.

In addition, the exhaust fission gases and exhaust flow rate from vented fueled experiments are monitored in accordance with Specification 3.8. Upon reaching an alarm setpoint from the vented fueled experiment exhaust radiation monitor, the vented fueled experiment exhaust is automatically isolated. Radiation monitor setpoints are analyzed as described in the Final Safety Analysis Report

3.6 Confinement and Main HVAC Systems

Applicability

This specification applies to the operation of the Reactor Building confinement and main HVAC systems.

Objective

The objective is to ensure that the confinement system is in operation to mitigate the consequences of possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated, nor shall irradiated fuel or irradiated fueled experiments be moved within the reactor building, unless the following conditions are met and equipment is operable:

a. All doors, except the Control Room, and basement corridor entrance are self-latching, self-closing, closed and locked. (1)

b. Control room and basement corridor entrance door are self-latching, self-closing and closed. (2)

c. Reactor Building is under a negative differential pressure of not less than 0.2" H_2O with the normal ventilation system or 0.1" H_2O with one confinement fan operating. (3)

d. Confinement system (4)(5)(7)

e. Evacuation system (6)

(1) Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided audible and/or visual indications are available for the reactor operator to verify door status.

(2) Doors may be opened for periods of less than five minutes for personnel and equipment transport between corridor area and Reactor Building.

(3) During an interval not to exceed 30 minutes after a loss of differential pressure is identified with Main HVAC operating, reactor operation may continue while the loss of differential pressure is investigated and corrected.

(4) Operability is also demonstrated with an auxiliary power source.

(5) One filter train may be out of service for the purpose of maintenance, repair, and/or surveillance for a period of time not to exceed 45 days. During the period of time in which one filter train is out of service, the standby filter train shall be verified to be operable every 24 hours if the reactor is operating with the Reactor Building in normal ventilation.

(6) The public address system can serve temporarily for the Reactor Building evacuation system during short periods of maintenance.

(7) When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, listed in Table 3-3, the building will be automatically placed in

confinement as described in the Safety Analysis Report.

Bases

In the event of a fission product release, the confinement initiation system will secure the normal ventilation fans and close the normal inlet and exhaust dampers. In confinement mode, a confinement system fan will:

- maintain a negative pressure in the Reactor Building and insure in-leakage only
- purge the air from the building at a greatly reduced and controlled flow through charcoal and absolute filters
- control the discharge of all air through a 100-foot stack on site

The allowance for operation under a temporary loss of differential pressure when in normal ventilation is based on the requirement of having the confinement system operable and therefore ready to respond in the unlikely event of an airborne release.

3.7 Limitations of Experiments

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities. Fueled experiments must also meet the requirements of Specification 3.8.

Objective

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

Specification

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

a. All materials to be irradiated shall be either corrosion resistant or encapsulated within a corrosion resistant container to prevent interaction with reactor components or pool water. Corrosive materials, liquids, and gases shall be doubly encapsulated.

b. 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. Pressure buildup inside any container shall be limited to 200 psi.

c. Cooling shall be provided to prevent the surface temperature of an experiment to be irradiated from exceeding the saturation temperature of the reactor pool water.

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

e. Concerning the material content of experiments, the following shall apply:

   i. No experiment shall be performed unless the major constituent of the material to be irradiated is known and a reasonable effort has been made to identify trace elements and impurities whose activation may pose the dominant radiological hazard. When a reasonable effort does not give conclusive information, one or more short irradiations of small quantities of material may be performed in order to identify the trace elements and impurities.

   ii. Attempts should be made to identify and limit the quantities of elements having very large thermal neutron absorption cross sections, in order to quantify reactivity effects.

   iii. Experiments in which the material is considered to be explosive (1), either while contained, or if it leaks from the container, shall be designed to maintain seal integrity even if detonated, to prevent damage to the reactor core or to the control rods or instrumentation and to prevent any change in reactivity.

iv. Each experiment shall be evaluated with respect to radiation induced physical and/or chemical changes in the irradiated material, such as decomposition effects in polymers.

v. Experiments involving cryogenic liquids[(1)] within the biological shield, flammable[(1)], or highly toxic materials[(1)] require specific procedures for handling and shall be limited in quantity and approved as specified in Specification 6.2.3.

f. Credible failure of any experiment shall not result in releases or exposures in excess of the annual limits established in 10 CFR Part 20.

[(1)] Defined as follows (reference - *Handbook of Laboratory Safety* - Chemical Rubber Company, 5th Ed., 2000, unless otherwise noted):

Toxic: A substance that has the ability to cause damage to living tissue when inhaled, ingested, injected, or absorbed through the skin (*Safety in Academic Chemistry Laboratories* - The American Chemical Society, 7th Ed., 2003).

Flammable: Having a flash point below 73°F and a boiling point below 100°F. The flash point is defined as the minimum temperature at which a liquid forms a vapor above its surface in sufficient concentrations that it may be ignited as determined by appropriate test procedures and apparatus as specified.

Explosive: Any chemical compound, mixture, or device, where the primary or common purpose of which is to function by explosion with substantially simultaneous release of gas and heat, the resultant pressure being capable of destructive effects. The term includes, but is not limited to, dynamite, black powder, pellet powder, initiating explosives, detonators, safety fuses, squibs, detonating cord, igniter cord, and igniters.

Cryogenic: A cryogenic liquid is considered to be a liquid with a normal boiling point below -243°F (reference - *National Bureau of Standards Handbook 44*).

Bases

Specifications 3.7.a, 3.7.b, 3.7.c, and 3.7.d are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure; and, serve as a guide for the review and approval of new and untried experiments.

Specification 3.7.e ensures that no physical or nuclear interferences compromise the safe operation of the reactor, specifically, an experiment having a large reactivity effect of either sign could produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculation and/or safety channels calibrations.

Review of experiments using the specifications of Section 3 and Section 6 ensure that experiments are consistent with all license conditions.

3.8 Operations with Fueled Experiments

Applicability

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

Objective

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

Specifications

Fueled experiments may be performed in experimental facilities of the reactor with the following conditions and limitations:

a. Specification 3.2 pertaining to experiment reactivity worth shall be met.

b. Specifications 3.5 and 3.6 pertaining to operation of the radiation monitoring system and ventilation system shall be met during reactor operation or if moving or handling an irradiated fueled experiment.

c. Specification 3.7 pertaining to limitations on experiments shall be met, with the exception that containers used for vented fueled experiment shall meet specification 3.8.d.iii.1.

d. Fissionable materials used in fueled experiments shall meet the following:

  i. Fissionable material physical form shall be solid, powder, or liquid.

  ii. Any mixture of fissionable material is permitted provided that the radiation dose to members of the public outside the reactor building is less than three mrem(1).

  iii. Vented fueled experiments shall also meet the following:

    1. Fission gases and halogens may be released.
    2. Filtration of experiment exhaust for particulates and halogens.(2)
    3. Monitoring of the experiment exhaust flow rate.
    4. Monitoring of the experiment exhaust gas for radioactivity with a setpoint meeting Specification d.ii.
    5. Monitoring for halogens in the stack particulate radiation monitoring channel.

e. Specification 5.3 pertaining to criticality control for fueled experiments in storage shall be met.

f. Specifications 6.2.3 and 6.5 pertaining to the review of experiments shall be met.

(1) Total Effective Dose-Equivalent as defined in 10 CFR Part 20 and calculated as described in the Safety Analysis Report.

(2) At flow rates specified for the vented fueled experiment, removal efficiency for the filter train shall be 0.99 or greater. Individual filter removal efficiency

shall be 0.95 or greater. Filter efficiencies shall be maintained for the entire duration that the filter is in use.

Bases

TS 3.8 provides limiting conditions for operation for fueled experiments and establishes an upper limit based on three percent (3%) of the annual radiation dose limit given in 10 CFR Part 20 for members of the public outside of the reactor building as determined from the more restrictive of two release scenarios;

1) A vented experiment in which the fission gases and halogens are continuously filtered, delayed, and then directly exhausted into the ventilation system over the entire duration of the experiment.

or

2) An accidental release from an encapsulation failure which results an instantaneous release of fission gases and halogens into the reactor building and is subsequently ventilated by the reactor building confinement and evacuation system for a period of 24 hours. In this scenario, the fueled experiment irradiation is assumed to end at the initiation of the accidental release due to the activation of the confinement and evacuation systems.

Specification 3.8.a requires all specifications pertaining to experiment reactivity given in TS 3.2 be satisfied thus ensuring that reactivity control of the reactor will be maintained.

Specification 3.8.b requires specifications TS 3.5 and TS 3.6 pertaining to the radiation monitoring and ventilation system be satisfied thus ensuring that a public dose of 0.003 rem will not be exceeded should an accidental release occur during irradiation and/or handling of a fueled experiment.

Specification 3.8.c requires all specifications pertaining to limitations on experiments given in TS 3.7 be satisfied thus ensuring that fueled experiments also meet the requirements for all experiments.

Specification 3.8.d provides limitations for fissionable materials used in fueled experiments.

Specification 3.8.d.i lists the physical forms allowed in fueled experiments.

Specification 3.8.d.ii limits radiation dose from the release of fission products to a Total Effective Dose-Equivalent (TEDE) of 0.003 rem in public areas outside the reactor building. Meeting Specification 3.8.d.ii gives a TEDE less than 1 rem and the Total Organ Dose-Equivalent to the thyroid (TODE) less than 10 rem to occupants inside the reactor building.

Specification 3.8.d.iii provides controls for planned releases from vented experiments needed to ensure that radiation dose does not exceed three percent of the annual radiation dose limits for members of the public given in 10 CFR Part 20. Radiation doses were calculated as described in Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor. A footnote to TS 3.8.d.iii.2 was added to specify the required filter removal efficiency.

Specification 3.8.e requires all specification pertaining to criticality control given in

TS 5.3 be satisfied thus ensuring that fueled experiments are stored in sub-critical configurations.

Specification 3.8.f requires specification TS 6.2.3 and 6.5 pertaining to the review and approval of experiments be satisfied thus ensuring that fueled experiments are reviewed, approved, and documented as required.

3.9 Primary Coolant

Applicability

This specification applies to the water quality of the primary coolant.

Objective

The objective is to ensure that primary coolant quality be maintained to acceptable values in order to reduce the potential for corrosion and limit the buildup of activated contaminants in the primary piping and pool.

Specification

The reactor shall not be operated unless the pool water meets the following limits:

a. The resistivity shall be ≥500 kΩ·cm.

Bases

The limits on resistivity are based on reducing the potential for corrosion in the primary piping or pool liner and to reduce the potential for activated contaminants in these systems. As detailed in the Safety Evaluation on Electrolytic Conductivity (TAC No. ME8511) a specification for pH is not necessary if resistivity is monitored.

4 SURVEILLANCE REQUIREMENTS

All surveillance tests required by these specifications are scheduled as described; however, some system tests may be postponed at the required intervals if that system or a closely associated system is undergoing maintenance. Any pending surveillance tests will be completed prior to reactor startup. Any surveillance item(s) which require reactor operation will be completed immediately after reactor startup. Surveillance requirements scheduled to occur during extended operation which cannot be performed while the reactor is operating may be deferred until the next planned reactor shutdown.

Appropriate surveillance testing on any technical specification required system shall be conducted after replacement, repair, or modification before the system is considered operable and returned to service.

Surveillance specification requirement should prescribe the frequency and scope of surveillance to demonstrate such performance. Maximum allowable intervals listed as follows are to provide operational flexibility only and are not to be used to reduce frequency. Established frequencies shall be maintained over the long term. Allowable surveillance intervals shall not exceed the following:

- 5-year interval not to exceed 6 years;
- biennially interval not to exceed 30 months;
- annually interval not to exceed 15 months;
- semi-annually interval not to exceed 7½ months;
- quarterly interval not to exceed 4 months;
- monthly interval not to exceed 6 weeks;
- bi-weekly interval not to exceed 18 days;
- weekly interval not to exceed 10 days;
- daily must be done during the calendar day;
- within a shift - must be done during a reactor shift;

4.1 Reactor Fuel and Core Configuration

Applicability

This specification applies to the surveillance requirement for the reactor fuel.

Objective

The objective is to monitor the physical condition of the PULSTAR core and fuel.

Specification

a. The following shall be verified prior to the first reactor startup of the day:
   i. A maximum of 25 fuel assemblies are loaded into the core.
   ii. Reflector assemblies are only located on the core periphery.
   iii. All unoccupied grid penetrations are plugged.
   iv. A minimum of four control rod guides are in place with operable control rods.

b. The reactivity worth of a fuel assembly while being loaded into the reactor grid plate shall be determined for all new core configurations.

c. The total pin power peaking factor for each fuel assembly shall be determined for all new core configurations prior to full power operations.

d. All fuel assemblies shall be visually inspected for physical damage biennially.

e. The reactor will be operated at such power levels necessary to determine if there is damaged fuel.

Bases

Specification 4.1.b provides assurances that a fuel loading accident will not result in fuel cladding failure as discussed in the Safety Analysis Report.

Specification 4.1.c provides assurances that fuel integrity is maintained as discussed in the Safety Analysis Report.

Each fuel assembly is visually inspected for physical damage that would include corrosion of the end fitting, end box, zircaloy box, missing fasteners, dents, severe surface scratches, and blocked coolant channels.

Based on a long history of prototype PULSTAR operation in conjunction with primary coolant analysis, biennial inspections of PULSTAR fuel to ensure fuel assembly integrity have been shown to be adequate for Zircoloy-2 (Zr-2) clad fuel. Any assembly that appears to have leaking fuel pin(s) may be disassembled to confirm and isolate damaged fuel pins. Damaged fuel pins shall be logged as such and permanently removed from service.

4.2 Control Rods

Applicability

This specification applies to the surveillance requirements for the control rods, shim rod, and control rod drive mechanisms (CRDM).

Objective

The objective is to assure the operability of the control rods and to provide current reactivity data for use in verifying adequate shutdown margin.

Specification

a. The reactivity worth of the shim rod and each control rod shall be determined annually.

b. The reactivity worth of all control rods shall be determined for any new core or rod configuration, prior to routine reactor operation.

c. Control rod scram times and control rod drive times shall be determined:

   i. Annually.

   ii. After a control assemblyis moved to a new position in the core or after maintenance or modification is performed on the control rod drive mechanism.

d. The control rods shall be visually inspected biennially.

e. The values of excess reactivity and shutdown margin shall be determined monthly and for new core configurations.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worths of experiments inserted in the core. The measurement of reactivity worths on an annual basis provides a correction for the slight variations expected due to burnup. This frequency of measurement has been found acceptable at similar research reactor facilities, particularly the prototype PULSTAR which has a similar slow change of rod value with burn-up.

Control rod drive and scram time measurements are made to determine whether the rods are functionally operable.

Visual inspections include: detection of wear or corrosion in the rod drive mechanism; identification of deterioration, corrosion, flaking or bowing of the neutron absorber material; and verification of rod travel setpoints. Control rod surveillance procedures should document proper control rod system reassembly after maintenance and recorded post-maintenance data should identify significant trends in rod performance.

4.3 Reactor Safety Systems

Applicability

This specification applies to the surveillance requirements for the Reactor Safety System and other required reactor instruments.

Objective

The objective is to assure that the required Reactor Safety Systems shall remain operable and shall prevent the Limiting Safety System Settings from being exceeded.

Specification

a. A channel check of each measuring channel in the RSS shall be performed daily when the reactor is in operation.

b. A channel test of each channel in the RSS shall be performed prior to the first reactor startup of the day.

c. A channel calibration of the following channels shall be made semi-annually.

   i. Pool Water Temperature

   ii. Primary Cooling and Flow Monitoring (Flapper)

   iii. Pool Water Level

   iv. Primary Heat Exchanger Inlet and Outlet Temperature

   v. Safety and Linear Power Channels

Bases

The daily channel tests and checks will assure the Reactor Safety Systems are operable and will assure operations within the limits of the operating license. The semi-annual calibrations will assure that long term drift of the channels is corrected.

4.4 Reactor Instrumentation

Applicability

This specification applies to the surveillance requirements for the required Reactor Instrumentation.

Objective

The objective is to assure that the required Reactor Instrumentation shall remain operable and shall prevent the Limiting Safety System Settings from being exceeded.

Specification

a. A channel calibration of the ^{16}N Channel shall be made semi-annually.

b. A calorimetric measurement shall be performed semi-annually to determine the ^{16}N detector current associated with full power operation.

c. A channel calibration of the Control Rod Position Indicators for each control rod shall be made semi-annually.

d. The control room differential pressure (dp) gauges shall be calibrated annually.

Bases

The semi-annual calibrations will assure that long term drift of the channels is corrected. The calorimetric calibration of the reactor power level, in conjunction with the ^{16}N Channel, provides a continual reference for adjustment of the Linear, LogN and Safety Channel detector positions.

4.5 Radiation Monitoring Equipment

Applicability

This specification applies to the surveillance requirements for the area and stack effluent radiation monitoring equipment.

Objective

The objective is to assure that the radiation monitoring equipment is operable.

Specification

a. A channel check of each required measuring channel listed in Table 3-3 shall be performed daily when the reactor is in operation.

b. The area and stack monitoring systems shall be calibrated annually.

c. The setpoints shall be verified monthly.

d. Vented fueled experiment exhaust gas radiation and flow monitors shall be calibrated prior to initial operation of the experiment and annually thereafter for as long as the experiment is in operation.

e. Filter replacements for vented fueled experiments shall be biennial and shall have a removal efficiency for iodine adsorption of 0.95 or greater at the specified flow rates used.

Bases

These systems provide continuous radiation monitoring of the Reactor Building with a check of readings performed prior to and during reactor operations.

The weekly verification of the setpoints in conjunction with the annual calibration is adequate to identify long term variations in the system operating characteristics.

4.6 Confinement and Main HVAC System

Applicability

This specification applies to the surveillance requirements for the confinement and main HVAC systems.

Objective

The objective is to assure that the confinement system is operable.

Specification

a. The confinement and evacuation system shall be verified to be operable within seven (7) days prior to reactor operation.

b. Operability of the confinement system on auxiliary power shall be checked monthly. [1]

c. A visual inspection of the door seals and closures, dampers and gaskets of the confinement and ventilation systems shall be performed semi-annually to verify they are operable.

d. The confinement filter train shall be tested biennially and prior to reactor operation following confinement HEPA or carbon adsorber replacement. This testing shall include iodine adsorption, particulate removal efficiency and leak testing of the filter housing.[2]

e. The air flow rate in the confinement stack exhaust duct shall be determined annually.

[1] Operation must be verified following modifications or repairs involving load changes to the auxiliary power source.

[2] Testing shall also be required following major maintenance of the filters or housing.

Bases

Surveillance of this equipment should verify that the confinement of the Reactor Building is maintained as described in Section 5 of the SAR.

4.7 Primary and Secondary Coolant

Applicability

This specification applies to the surveillance requirement for monitoring the radioactivity in the primary and secondary coolant.

Objective

The objective is to monitor the radioactivity in the pool water to verify the integrity of the fuel cladding and other reactor structural components. The secondary water analysis is used to confirm the boundary integrity of the primary heat exchanger.

Specification

a. The primary coolant shall be analyzed bi-weekly. The analysis shall include gross beta/gamma counting of the dried residue of a one (1) liter sample or gamma spectroscopy of a liquid sample, neutron activation analysis (NAA) of an aliquot, and resistivity measurements.

b. The secondary coolant shall be analyzed bi-weekly. This analysis shall include gross beta/gamma counting of the dried residue of a one (1) liter sample or gamma spectroscopy of a liquid sample.

Bases

Radionuclide analysis of the pool water samples should allow detection of fuel clad failure, while neutron activation analysis should give corrosion data associated with primary system components in contact with the coolant. Refer to SAR Section 11. The detection of activation or fission products in the secondary coolant provides evidence of a primary heat exchanger leak. Refer to SAR Section 11.

5 DESIGN FEATURES

5.1. Reactor Fuel

a. The reactor fuel shall be UO_2 with a nominal enrichment of 4% or 6% in U-235, zircaloy clad, with fabrication details as described in the Safety Analysis Report.

b. Total burn-up on the reactor fuel is limited to 20,000 MWD/MTU.

5.2. Reactor Building

a. The reactor shall be housed in the Reactor Building, designed for confinement. The minimum free volume in the Reactor Building shall be 2.4×10^9 cm^3.

b. The Reactor Building ventilation and confinement systems shall be separate from the Burlington Engineering Laboratories building systems and shall be designed to exhaust air or other gases from the building through a stack with discharge at a minimum of 100 feet above ground level.

c. The openings into the Reactor Building are the truck entrance door, personnel entrance doors, and air supply and exhaust ducts.

d. The Reactor Building is located within the Burlington Engineering Laboratory complex on the north campus of North Carolina State University at Raleigh, North Carolina. Restricted Areas as defined in 10 CFR Part 20 include the Reactor Bay, Ventilation Equipment Room, Mechanical Equipment Room, Primary Piping Vault, and Waste Tank Vault. The PULSTAR Control Room is part of the Reactor Building, however it is also a controlled access area and a Controlled Area as defined in 10 CFR Part 20. The facility license applies to the Reactor Building and Waste Tank Vault. Figure 5-1 depicts the licensed area as being within the operations boundary.

5.3. Fuel Storage

Fuel, including fueled experiments, shall be stored in a geometrical configuration where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water and shall have adequate cooling. In cases where a fuel shipping container is used, then the licensed limit for the k_{eff} limit of the container shall apply.

5.4 Reactivity Control

Reactivity control is provided by four neutron absorbing blades. Each control blade is nominally comprised of 80% silver, 15% indium, and 5% cadmium with tin-nickel cladding. These neutron absorbing blades are magnetically coupled and have scram capability. One of the control rods may be used for automatic control of reactor power.

5.5 Primary Coolant System

The primary coolant system consists of the aluminum lined reactor tank, a ^{16}N delay tanks, a pump, and heat exchanger, and associated stainless steel piping. The nominal capacity of the primary system is 16,000 gallons. To allow for isolation, valves are located adjacent to the biological shield and at major components in the primary system.

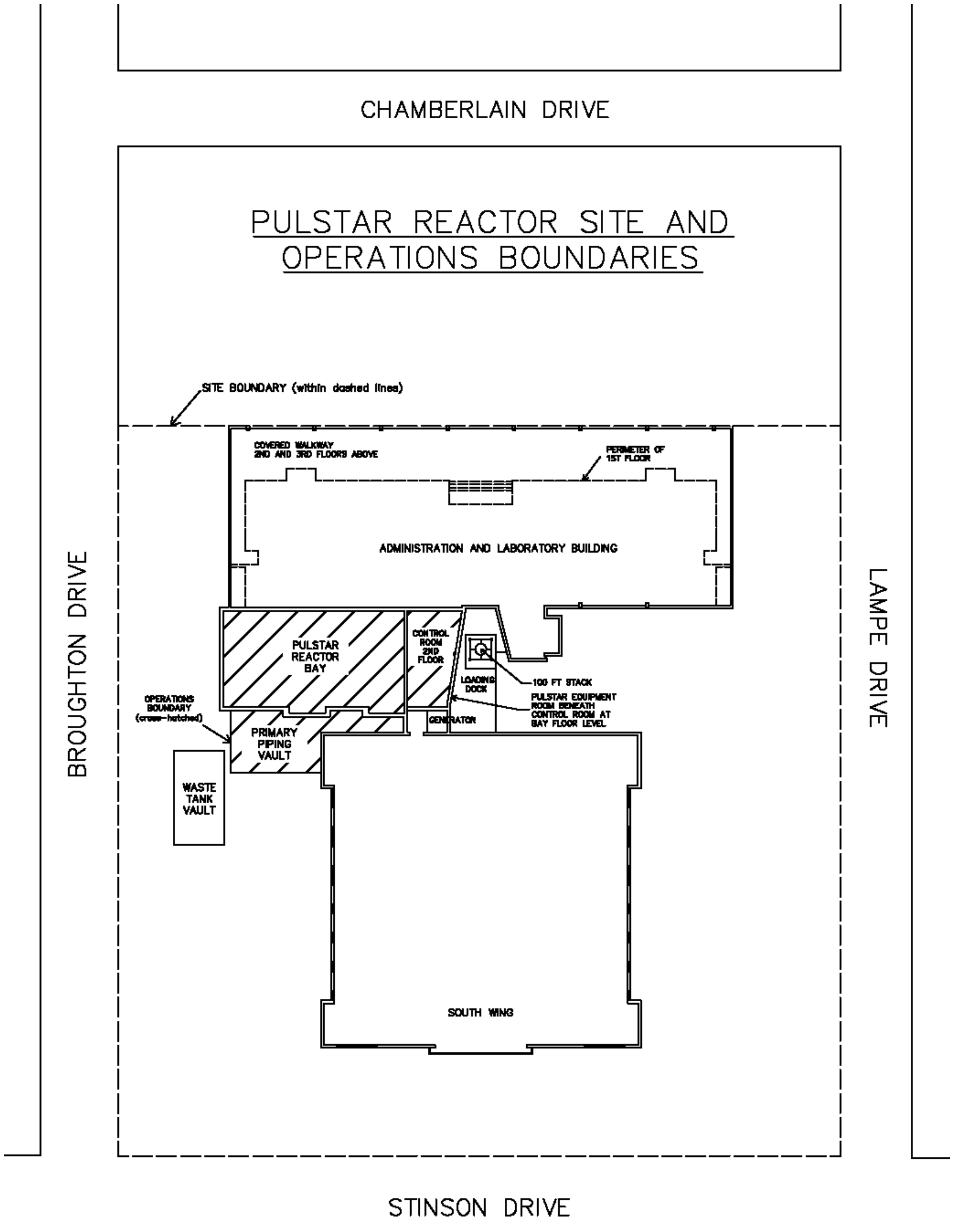


Figure 5-1 – NCSU PULSTAR Reactor Site Map

6 ADMINISTRATIVE CONTROLS

6.1 Organization

The reactor facility shall be an integral part of the Department of Nuclear Engineering of the College of Engineering of North Carolina State University. The reactor shall be related to the University structure as shown in Figure 6-1.

6.1.1 Organizational Structure:

The reporting chain shall be as given in Figure 6.1. The following specific organizational levels (as defined by ANSI/ANS-15.1-2007) and positions shall exist at the PULSTAR Facility:

Level 1 – Administration

This level shall include the Chancellor, the Dean of the College of Engineering, and the Nuclear Engineering Department Head. Within three months of appointment, the Nuclear Engineering Department Head shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the facility. This level has the authority to ensure resources are available to guarantee public health and safety.

Level 2 – Facility Management

This level shall include the Nuclear Reactor Program (NRP) Director. The NRP Director is responsible for the safe and efficient operation of the facility as specified in the facility license and Technical Specifications, general conduct of reactor performance and NRP operations, long range development of the NRP, and NRP personnel matters. The NRP Director evaluates new service and research applications, develops new facilities and support for needed capital investments, and controls NRP budgets.

The NRP Director works through the Manager of Engineering and Operations to monitor daily operations and with the Reactor Health Physicist to monitor radiation safety practices and regulatory compliance. The minimum qualifications for the NRP Director are a Master of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. Within three months of appointment, the NRP Director shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the facility. The NRP Director is a faculty member and reports to the Nuclear Engineering Department Head.

Level 3 – Manager of Engineering and Operations

The Manager of Engineering and Operations (MEO) performs duties as assigned by the NRP Director associated with the safe and efficient operation of the facility as specified in the facility license and Technical Specifications. The MEO is responsible for coordination of operations, experiments, and maintenance at the facility, including reviews and approvals of experiments as defined in Technical Specification 1.3.10 and

6.5, and making changes to procedures as stated in Technical Specification 6.4. The MEO shall receive appropriate facility specific training within three months of appointment and be certified as a Senior Reactor Operator within one year of appointment. The minimum qualifications for the MEO are a Bachelor of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. The MEO reports to the NRP Director.

Level 4 – Operating and Support Staff

This level includes licensed Senior Reactor Operators (SRO), licensed Reactor Operators (RO), and other personnel assigned to perform maintenance and technical support of the facility. Senior Reactor Operators and Reactor Operators are responsible for ensuring that operations are conducted in a safe manner and within the limits prescribed by the facility license and Technical Specifications, applicable Nuclear Regulatory Commission regulations, and the provisions of the Radiation Safety Committee and Reactor Safety and Audit Committee. All Senior Reactor Operators shall have three years of nuclear experience and shall have a high school diploma or successfully completed a General Education Development test. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of nuclear experience as applicable to research reactors for Senior Reactor Operators. Other Level 4 personnel shall have a high school diploma or shall have successfully completed a General Education Development test. All Level 4 personnel report to the Manager of Engineering and Operations.

Reactor Health Physicist

The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP shall have a high school diploma or shall have successfully completed a General Education Development test and have three years of relevant experience in applied radiation safety. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of experience in radiation safety as applicable to research reactors. The RHP reports directly to the Nuclear Engineering Department Head and is independent of the campus Radiation Safety Division as shown in Figure 6-1.

6.1.2 Responsibility

Responsibility for the safe operation of the PULSTAR Reactor shall be within the chain of command established in Figure 6-1.

Individuals at the 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 to all requirements of the operating

license, the Technical Specifications, and federal regulations.

In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon the appropriate qualifications.

6.1.3 Minimum Staffing

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

a. A licensed reactor operator or senior reactor operator shall be present in the Control Room.

b. A Reactor Operator Assistant (ROA), capable of being at the reactor facility within five (5) minutes upon request of the reactor operator on duty.

c. A Designated Senior Reactor Operator (DSRO). This individual shall be readily available on call, meaning:

   i. Has been specifically designated and the designation known to the reactor operator on duty.

   ii. Keeps the reactor operator on duty informed of where he may be rapidly contacted and the telephone number.

   iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15 mile radius).

d. A Reactor Health Physicist or his designated alternate. This individual shall also be on call, under the same limitations as prescribed for the Designated Senior Reactor Operator under Specification 6.1.3.c.

6.1.4 Senior Reactor Operator Duties

The following events shall require the presence of a licensed Senior Reactor Operator within the site boundary as shown in Figure 5-1:

a. Initial startup and approach to power following each time the reactor has been secured.

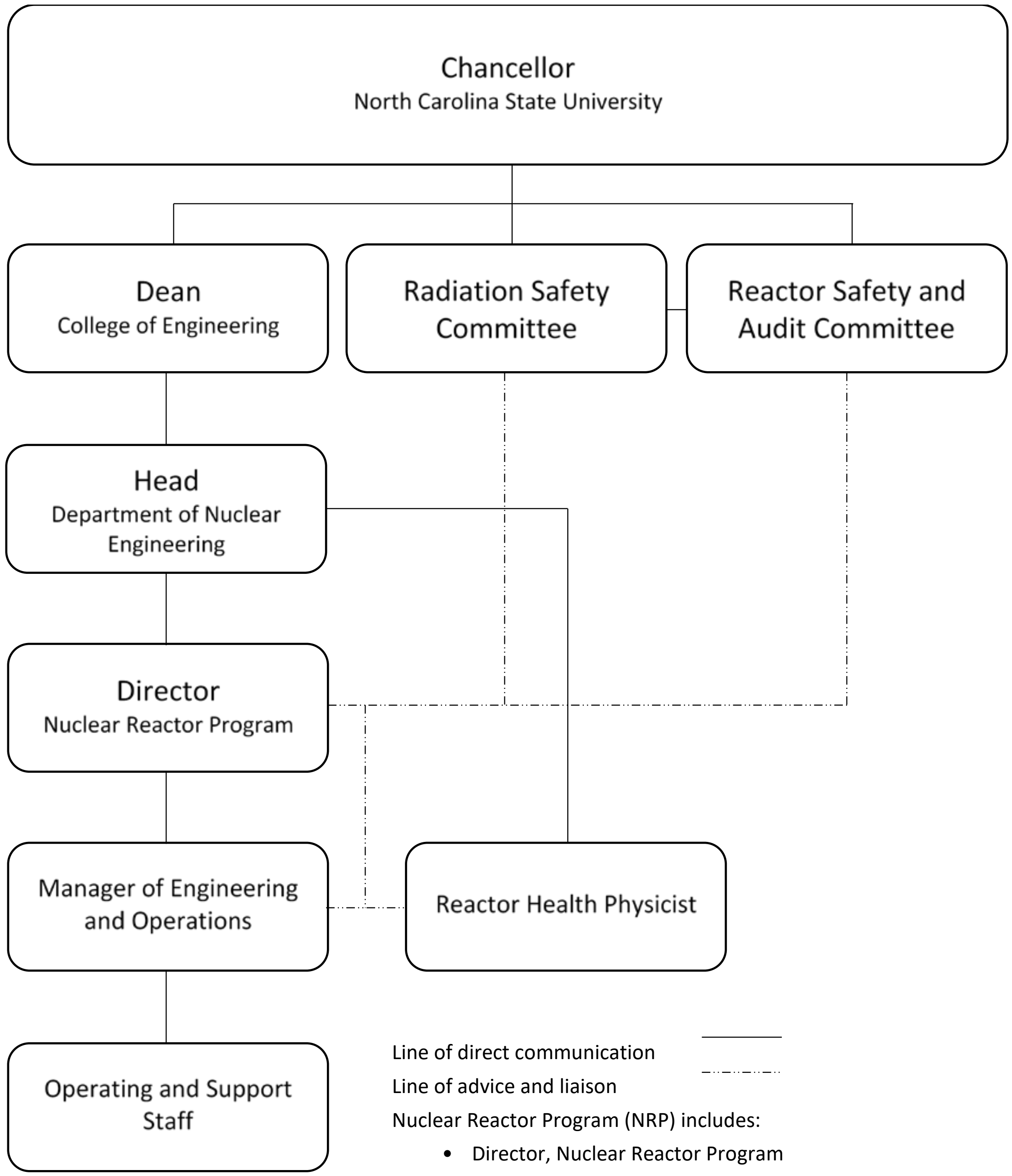
b. All fuel or control rod relocations within the reactor core or pool.

c. Relocation of any in-core experiment with a reactivity worth greater than one dollar.

d. Recovery from unplanned or unscheduled shutdown or an unplanned or unscheduled power reduction of greater than 25% (500 kW) of full licensed power.

6.1.5 Selection and Training

All operators shall undergo a selection, training and licensing program prior to unsupervised operation of the PULSTAR reactor. All licensed operators shall participate in a requalification program, which shall be conducted over a period not to exceed two (2) years. The requalification program shall be followed by successive two (2) year programs.



Line of direct communication

Line of advice and liaison

Nuclear Reactor Program (NRP) includes:

- Director, Nuclear Reactor Program
- Manager, Engineering and Operations
- Operating and Support Staff

Reactor Health Physicist (RHP) reports to the Head, Department of Nuclear Engineering and serves both the NRP and Department of Nuclear Engineering.

Communication on reactor operations, experiments, radiation safety, and regulatory compliance occurs between the NRP, RHP, Reactor Safety and Audit Committee, Radiation Safety Committee, and campus Radiation Safety Division as described in the Technical Specifications and facility procedures.

Figure 6-1 – NCSU PULSTAR Reactor Organizational Chart

6.2 Review and Audit

The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices, including the nuclear reactor, at the University are in compliance with state and federal licenses and all applicable regulations. The RSC reviews and approves all experiments involving the potential release of radioactive material conducted at the University and provides oversight of the University Radiation Protection Program. The RSC is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by RSAC and the Nuclear Reactor Program (NRP).

The primary responsibility of the RSAC is to provide independent review and audit of the operation of the reactor and that it is operated in compliance with the facility license, Technical Specifications, and all applicable regulations. RSAC performs an annual audit of the operations and performance of the NRP.

6.2.1 RSC Composition and Qualifications

RSC shall consist of members from the general faculty who are actively engaged in teaching or research involving radioactive materials or radiation devices. RSC may also include non-faculty members who are knowledgeable in nuclear science or radiation safety. At a minimum, RSC membership shall include the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

6.2.2 RSC Rules

a. RSC committee member appointments are made by University Management for terms of three (3) years.

b. RSC shall meet as required by the broad scope radioactive materials license issued to the University by the State of North Carolina. RSC may also meet upon call of the committee Chair.

c. A quorum of RSC shall consist of a majority of the full committee membership and shall include the committee Chair or a designated alternate for the committee Chair. Members from the line organization shown in Figure 6.1 shall not constitute a majority of the RSC quorum.

6.2.3 RSC Review and Approval Function

a. The following items shall be reviewed and approved by the RSC:

   i. All new experiments or classes of experiments that could result in the release of radioactivity.

   ii. Proposed changes to the facility license or Technical Specifications, excluding safeguards information.

b. The following items shall be reviewed by the RSC:

   i. Violations of the facility license or Technical Specifications

   ii. Violations of internal procedures or instructions having safety

significance.

iii. Operating abnormalities having safety significance.

iv. Reportable Events as defined in Specification 1.3.29.

Distribution of RSC summaries and meeting minutes shall include the RSAC Chair and Director of the Nuclear Reactor Program.

6.2.4 RSAC Composition and Qualifications:

a. RSAC shall consist of at least five individuals who have expertise in one or more of the component areas of nuclear reactor safety. These include Nuclear Engineering, Nuclear Physics, Health Physics, Electrical Engineering, Chemical Engineering, Material Engineering, Mechanical Engineering, Radiochemistry, and Nuclear Regulatory Affairs.

At least three of the RSAC members are appointed from the faculty. The faculty members shall be as follows:

i. NRP Director.

ii. One member from an appropriate discipline within the College of Engineering.

iii. One member from the faculty.

The remaining RSAC members are as follows:

iv. Reactor Health Physicist (RHP).

v. Member from the campus Radiation Safety Division ofthe Environmental Health and Safety Center.

vi. One additional member from an outside nuclear related establishment may be appointed.

At the discretion of RSAC, specialist(s) from other universities and outside establishments may be invited to assist in its appraisals.

The NRP Director, RHP, and a member from the campus Radiation Safety Division of the Environmental Health and Safety Center are permanent members of RSAC.

6.2.5 RSAC Rules

a. RSAC committee member appointments are made by University Management for terms of three (3) years.

b. RSAC shall each meet at least four (4) times per year, with intervals between meetings not to exceed six months. RSAC may also meet upon call of the committee Chair.

c. A quorum of RSAC shall consist of a majority of the full committee membership and shall include the committee Chair or a designated alternate for the committee Chair. Members from the line organization shown in Figure 6.1 shall not constitute a majority of

the RSAC quorum.

6.2.6 RSAC Review and Approval Function

a. The following items shall be reviewed and approved by the RSAC:

   i. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures are allowed without prior authorization by the Nuclear Regulatory Commission pursuant to 10 CFR Part 50.59.

   ii. All new procedures and revisions, and proposed changes in reactor facility equipment, or systems.

   iii. All untried experiments.

   iv. Proposed changes to the facility license or Technical Specifications, including safeguards information.

b. The following items shall be reviewed by the RSAC:

   i. Violations of the facility license or Technical Specifications

   ii. Violations of internal procedures or instructions having safety significance.

   iii. Operating abnormalities having safety significance.

   iv. Reportable Events as defined in Specification 1.3.29.

A summary of RSAC meeting minutes, reports, and audit recommendations approved by RSAC shall be submitted to the Dean of the College of Engineering, the Nuclear Engineering Department Head, the Director of the Nuclear Reactor Program, the RSC Chair, Director of Environmental Health and Safety, RSAC Chair, and the Manager of Engineering and Operations within three (3) months.

6.2.7 RSAC Audit Function

The audit function shall consist of selective, but comprehensive, examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations shall also be used as appropriate. The RSAC shall be responsible for this audit function. In no case shall an individual immediately responsible for the area perform an audit in that area. This audit shall include:

a. Facility operations for conformance to the facility license and Technical Specifications, annually, but at intervals not to exceed fifteen (15) months.

b. The retraining and requalification program for the operating staff, biennially, but at intervals not to exceed thirty (30) months.

c. The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, annually, but at intervals not to exceed fifteen (15) months.

d. The Emergency Plan and Emergency Procedures, biennially, but at intervals not to exceed thirty (30) months.

e. Radiation Protection annually, but at intervals not to exceed fifteen (15) months.

f. Physical Security Plan and Procedures, biennially, but at intervals not to exceed thirty (30) months.

Deficiencies uncovered that affect reactor safety shall be reported to the Nuclear Engineering Department Head, Director of the Nuclear Reactor Program, and the RSC.

The annual audit report made by the RSAC, including any recommendations, is provided to the RSC within three (3) months.

6.3 Radiation Safety

The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP reports directly to the Nuclear Engineering Department Head and is independent of the campus Radiation Safety Division as shown in Figure 6-1.

6.4 Operating Procedures

Written procedures shall be prepared, reviewed, approved and followed prior to initiating any of the following:

a. Startup, operation and shutdown of the reactor.

b. Loading, unloading, and movement of fuel within the reactor.

c. Maintenance of major components of systems that may affect reactor safety.

d. Surveillance checks, calibrations and inspections required by the facility license or Technical Specifications or those that may affect reactor safety.

e. Personnel radiation protection, consistent with applicable regulations and that include commitment and/or programs to maintain exposures and releases as low as reasonably achievable (ALARA).

f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that may affect reactor safety or core reactivity.

g. Implementation of the Emergency Plan and Security Plan.

Changes to the above procedures shall be made pursuant to 10 CFR Part 50.59 and implemented only after documented review and approval by the RSAC, the NRP Director, and by the Manager of Engineering and Operations

Minor changes to the above procedures which do not change their original intent shall be made pursuant to 10 CFR Part 50.59 and may be implemented by the Manager of Engineering and Operations, but the change shall be approved by the Director of the Nuclear Reactor Program within fourteen (14) days.

6.5 Review of Experiments

6.5.1 Untried Experiments

All untried experiments, shall be reviewed and approved by the RSC, the RSAC, the Director of the Nuclear Reactor Program, Manager of Engineering and Operations, and the Reactor Health Physicist, prior to initiation of the experiment.

The review of untried experiments shall be pursuant to 10 CFR Part 50.59 based on the limitations prescribed by the facility license and Technical Specifications and other Nuclear Regulatory Commission regulations, as applicable.

6.5.2 Tried Experiments

All proposed experiments are reviewed by the Manager of Engineering and Operations and the Reactor Health Physicist (or their designated alternates). Either of these individuals may deem that the proposed experiment is not adequately covered by the documentation and/or analysis associated with an existing approved experiment and therefore constitutes an untried experiment that will require the approval process detailed under Specification 6.5.1.

If the Manager of Engineering and Operations and the Reactor Health Physicist concur that the experiment is adequately covered by the documentation and analysis associated with an existing approved experiment then it may be deemed a tried experiment, and the request may be approved.

Changes not adequately covered by the documentation and analysis associated with a previously approved experiment will require the approval process detailed under Specification 6.5.1.

6.6 Required Actions

6.6.1 Action to be Taken in Case of Safety Limit Violation

In the event a Safety Limit is violated:

a. The reactor shall be shutdown and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. The Safety Limit violation shall be promptly reported to the Director of the Nuclear Reactor Program, or his designated alternate.

c. The Safety Limit violation shall be reported to the Nuclear Regulatory Commission in accordance with Specification 6.7.1.

d. A Safety Limit violation report shall be prepared that describes the following:

   i. Circumstanced leading to the violation including, when known, the cause and contributing factors.

   ii. Effect of violation upon reactor facility components, systems, or structures and on the health and safety of facility personnel and the public.

   iii. Corrective action(s) to be taken to prevent recurrence.

The report shall be reviewed by the RSC and RSAC and any follow-up report shall be submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation.

6.6.2 Action to be Taken for Reportable Events (other than SL Violation)

In case of a Reportable Event (other than violation of a Safety Limit), as defined by Specification 1.3.29, the following actions shall be taken:

a. Reactor conditions shall be returned to normal or the reactor shall be shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operation shall not be resumed unless authorized by the Director of the Nuclear Reactor Program, or his designated alternate.

b. The occurrence shall be reported to the Director of the Nuclear Reactor Program, and to the Nuclear Regulatory Commission in accordance with Specification 6.7.1.

c. The occurrence shall be reviewed by the RSC and RSAC at their next scheduled meeting.

6.7 Reporting Requirements

6.7.1 Reportable Event

For Reportable Events as defined by Specification 1.3.29, there shall be a report no later than the following work day by telephone to the Nuclear Regulatory Commission Operations Center followed by a written report within fourteen (14) days that describes the circumstances of the event.

6.7.2 Permanent Changes in Facility Organization

Permanent changes in the facility organization involving either Level 1 or 2 personnel (refer to Specification 6.1.1) shall require a written report within thirty (30) days to the Nuclear Regulatory Commission Document Control Desk.

6.7.3 Changes Associated with the Safety Analysis Report

Significant changes in the transient or accident analysis as described in the Safety Analysis Report shall require a written report within thirty (30) days to the Nuclear Regulatory Commission Document Control Desk.

6.7.4 Annual Operating Report

An annual operating report for the previous calendar year is required to be submitted no later than March 31st of the present year to the Nuclear Regulatory Commission Document Control Desk. The annual report shall contain as a minimum, the following information:

a. A brief narrative summary:

   i. Operating experience including a summary of experiments performed.

   ii. Changes in performance characteristics related to reactor safety that occurred during the reporting period.

iii. Results of surveillance, tests, and inspections.

b. Tabulation of the energy output of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality.

c. The number of emergency shutdowns and unscheduled SCRAMs, including reasons and corrective actions.

d. Discussion of the corrective and preventative maintenance performed during the period, including the effect, if any, on the safety of operation of the reactor.

e. A brief description, including a summary of the analyses and conclusions of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR Part 50.59.

f. A summary of the nature and amount of radioactive effluent released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge, including:

Liquid Effluent (summarized by month and year)

i. Radioactive liquid effluent released during the reporting period:

1. Number of batch releases.
2. Total liquid volume released.
3. Total radioactivity released.
4. Tritium radioactivity released.
5. Compliance with 10 CFR Part 20 limits on concentration and radioactivity.

ii. Identification of fission and activation products:

Whenever the undiluted concentration of radioactivity in the waste tank at the time of release exceeds 2×10^{-5} μCi/ml, as determined by gross beta/gamma count of the dried residue of a one liter sample, a subsequent analysis shall also be performed prior to release for principle gamma emitting radionuclides. An estimate of the quantities present shall be reported for each of the identified nuclides.

iii. Disposition of liquid effluent not releasable to the sanitary sewer system:

Any waste tank containing liquid effluent failing to meet the requirements of 10 CFR Part 20, Appendix B, to include the following data:

1. Method of disposal.
2. Total radioactivity in the tank prior to disposal.
3. Total volume of liquid in tank.

4. The dried residue of one liter sample shall be analyzed for the principle gamma-emitting radionuclides. The identified isotopic composition with estimated concentrations shall be reported. The tritium content shall be included.

Airborne Effluent

i. Airborne radioactive effluent resulting from reactor operation during the reporting period for:
   1. Radionuclides, concentrations, and radioactivity.
   2. Halogens and particulates.
   3. Radiation dose to members of the public outside the facility.
   4. Compliance with 10 CFR Part 20 limits on annual radiation dose.

ii. The Effluent Concentration (EC) used and the estimated activity discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis. (EC values are given in 10 CFR Part 20, Appendix B, Table 2.)

Solid Waste

i. The total amount of solid waste packaged.

ii. The total activity involved.

iii. The dates of shipment and disposition (if shipped off-site).

g. A summary of radiation exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed.

h. A summary of the radiation and contamination surveys performed within the facility and significant results.

i. Summarized results of environmental surveys and external radiation dose measurements performed outside the facility.

6.8 Retention of Records

Records and logs of the following items, as a minimum, shall be kept in a manner convenient for review and shall be retained as detailed below. In addition, any additional federal requirement in regards to record retention shall be met.

6.8.1 Records to be retained for a period of at least five (5) years

a. Normal reactor facility operation and maintenance (but not including supporting documents such as checklists, log sheets etc., which shall be maintained for a period of at least 1 year).

b. Principal maintenance activities.

c. Reportable Events.

d. Surveillance activities required by the technical specifications.

e. Reactor facility radiation and contamination surveys other than those used

in support of personnel radiation monitoring.

f. Experiments performed with the reactor.

g. Changes to Operating Procedures.

h. RSC and RSAC meeting minutes and audit summaries.

6.8.2 Records to be retained for at least one (1) certification cycle of six (6) years:

Records of retraining and requalification of certified operating personnel shall be maintained at all times the individual is employed, or until the certification is renewed.

6.8.3 Records to be retained for the life of the facility

a. Gaseous and liquid radioactive waste released to the environs.

b. Results of off-site environmental monitoring surveys.

c. Radiation exposures for monitored personnel and associated radiation and contamination surveys used in support of personnel radiation monitoring.

d. Drawings of the reactor facility.

e. Fuel inventories and transfers.

f. Documentation of significant radioactive leaks and spills.