UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

RENEWAL OF FACILITY LICENSE

DOCKET NO. 50-297

NORTH CAROLINA STATE UNIVERSITY

Amendment No. 11
License No. R-120

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

   A. The application for renewal of Facility License No. R-120 filed by the North Carolina State University (NCSU or the licensee) dated August 19, 1988, as supplemented on January 2, April 17, and December 18, 1989; April 17 and July 18, 1990; January 25, 1991; November 30, 1992; September 15, 1995; and October 4, November 25, and December 30, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. Construction of the facility was completed in substantial conformity with Construction Permit No. CPRR-106, dated October 1, 1968; the provisions of the Act; and the regulations of the Commission;

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

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

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

   F. 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 Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

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

I. 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 Parts 30 and 70, including Sections 30.33, 70.23, and 70.31.

2. Facility License No. R-120 is hereby amended in its entirety to read as follows:

A. The license applies to the PULSTAR nuclear research reactor (the facility) owned by the NCSU. The facility is located on the campus in Raleigh, North Carolina, and is described in the licensee's application for renewal of the license dated August 19, 1988, as supplemented on January 2, April 17, and December 18, 1989; April 17 and July 18, 1990; January 25, 1991; November 30, 1992; September 15, 1995; and October 4, November 25, and December 30, 1996.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the NCSU:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in Raleigh, North Carolina, in accordance with the procedures and limitations set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use in connection with operation of the reactor up to 25 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel; up to 20 grams of contained uranium-235 of any enrichment in the form of fission chambers; up to 2 grams of contained uranium-235 of any enrichment in the form of foils; up to 200 grams of plutonium-239 in the form of plutonium-beryllium neutron sources; and to possess, but not separate, such special nuclear material as may be produced by the operation of the facility.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to possess, use, but not separate, except for byproduct material produced in non-fueled experiments, such byproduct material as may be produced by the operation of the facility.

2. B. (4) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to receive, possess, and use in connection with operation of the facility:

(a) any amount of byproduct material in the form of reactor components or otherwise integral to the reactor or reactor experimental facility;

(b) byproduct material which is to be irradiated in the reactor within 31 days of receipt.

Added per Amdt #16, 8-15-2006

C. This license shall be deemed to contain and is subject to the conditions specified in Parts 20, 30, 50, 51, 55, 70, and 73 of 10 CFR Chapter I, to all applicable provisions of the Act, and to the rules, regulations, and orders of the Commission now or hereafter in effect and to the additional conditions specified below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state power levels not to exceed 1000 kilowatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment 18, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Physical Security Plan

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security plan, including all amendments and revisions made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p), which are part of the license. This plan, which contains information withheld from public disclosure under 10 CFR 2.790, is entitled "NCSU PULSTAR Physical Security Plan," Revision 8, dated January 12, 1996.

D. This license is effective as of the date of issuance and shall expire 20 years from its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Seymour H. Weiss, Director
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical
Specifications

Date of Issuance: April 30, 1997

Appendix A

Technical Specifications for the North Carolina State University PULSTAR Reactor

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Facility License No. R-120

Docket No. 50-297

Amendment No. 17

Date: September 8, 2008

TABLE OF CONTENTS

FIGURES

TABLES

1.0 INTRODUCTION

1.1 Purpose

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.

Included in this document are the "Bases" for the Technical Specifications. The bases provide the technical support for the individual technical specification and are included for information purposes only. The bases are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.2 Definitions

1.2.1 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.

1.2.2 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.

1.2.3 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.

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

1.2.5 Cold Critical: The condition of the reactor when it is critical, with negligible xenon, and the fuel and bulk water are both at an isothermal temperature of 70°F.

1.2.6 Confinement: Confinement means a closure on the overall facility that controls the movement of air into and out of the facility through a controlled path.

1.2.7 Control Rod: A control rod is a neutron absorbing blade having an in-line drive which is magnetically coupled and has SCRAM capability.

1.2.8 Excess Reactivity: Excess reactivity is that amount of reactivity that would exist if all control rods (and Shim Rod) were fully withdrawn from the point where the reactor is exactly critical (k_{eff}=1).

1.2.9 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, and that is not rigidly secured to a core or shield structure so as to be a part of their design. Specific categories of experiments include:

a. **Tried Experiment**: Tried experiments are those experiments that have been previously performed in this reactor. Specifically, a tried experiment has similar size, shape, composition and location of an experiment previously approved and performed in the reactor.

b. **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.

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

d. **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.

e. **Fueled Experiment**: A fueled experiment is an experiment which contains fissionable material.

1.2.10 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.

1.2.11 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 50.36).

1.2.12 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 50.36).

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

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

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

1.2.16 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% $\Delta k/k$.

1.2.17 Reactor Building: The Reactor Building includes the Reactor Bay, Control Room and Ventilation 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.

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

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

1.2.20 Reactor Operator Assistant (ROA): An individual who has been certified by successful completion of an in-house training program to assist the licensed reactor operator during reactor operation.

1.2.21 Reactor Safety System: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

1.2.22 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 (730 pcm).

1.2.23 Reactor Shutdown: That subcritical condition of the reactor where the absolute value of the negative reactivity of the core is equal to or greater than the shutdown margin.

1.2.24 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 specifications.

d. Operation in violation of Limiting Conditions for Operation (LCO) established in these Technical Specifications.

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 discovered during maintenance tests or periods of reactor shutdown. (For components or systems other than those required by these Technical Specifications, the failure of the extra component or systems is not considered reportable provided that the minimum number 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 (730 pcm). 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), which could result in exceeding radiological limits for personnel or environment, or both, as prescribed in the facility Emergency Plan.

h. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence of an unsafe condition with regard to reactor operations.

1.2.25 **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 50.36).

1.2.26 **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.

1.2.27 **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.

1.2.28 **Shutdown Margin**: Shutdown margin means 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 with the most reactive scrammable rod fully withdrawn, the non-scrammable rod (Shim rod) fully withdrawn, and experiments considered at their most reactive condition, and finally, that the reactor will remain subcritical without further operator action.

1.2.29 **Total Nuclear Peaking Factor**: The factor obtained by multiplying the measured local radial and axial neutron fluence peaking factors.

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

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

1.2.32 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.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits (SL)

2.1.1 Safety Limits for Forced Convection Flow

Applicability

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

P Reactor Thermal Power

W Reactor Coolant Flow Rate

H Height of Water Above the Top of the Core

$\mathbf{T}_{inlet}$ Reactor Coolant Inlet Temperature

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of forced convection flow, the Safety Limit shall be as follows:

a. The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 2.1-1 under any operating conditions. The limits are considered exceeded if the point defined by the true values of P and W is at any time outside the operating envelope shown in Figure 2.1-1.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

Bases

Above 80 percent of the full core flow of 500 gpm in the region of full power operation, the criterion used to establish the Safety Limit was no bulk boiling at the outlet of any coolant channel. This was found to be far more limiting than the criterion of a minimum allowable burnout heat flux ratio of 2.0. The analysis is given in the SAR Appendix 3B.

In the region below 80 percent of full core flow, where, under a loss of flow transient at power the flow coasts down to zero, reverses, and then establishes natural convection, the criterion for selecting a Safety Limit is taken as a fuel cladding temperature. The analysis of a loss of flow transient is presented in Appendix 3B of the SAR. For initial conditions of full flow and an operating power of 1.4 MWt, the maximum clad temperature reached under the conservative assumptions of the analysis was 273°F which is well below the temperature at which fuel clad damage could possibly occur. The Safety Limit shown in Figure 2.1-1 for flow less than 80 percent of full flow is the steady state power corresponding to the maximum fuel clad temperature of 273°F with natural convection flow, namely, 1.4 MWt.

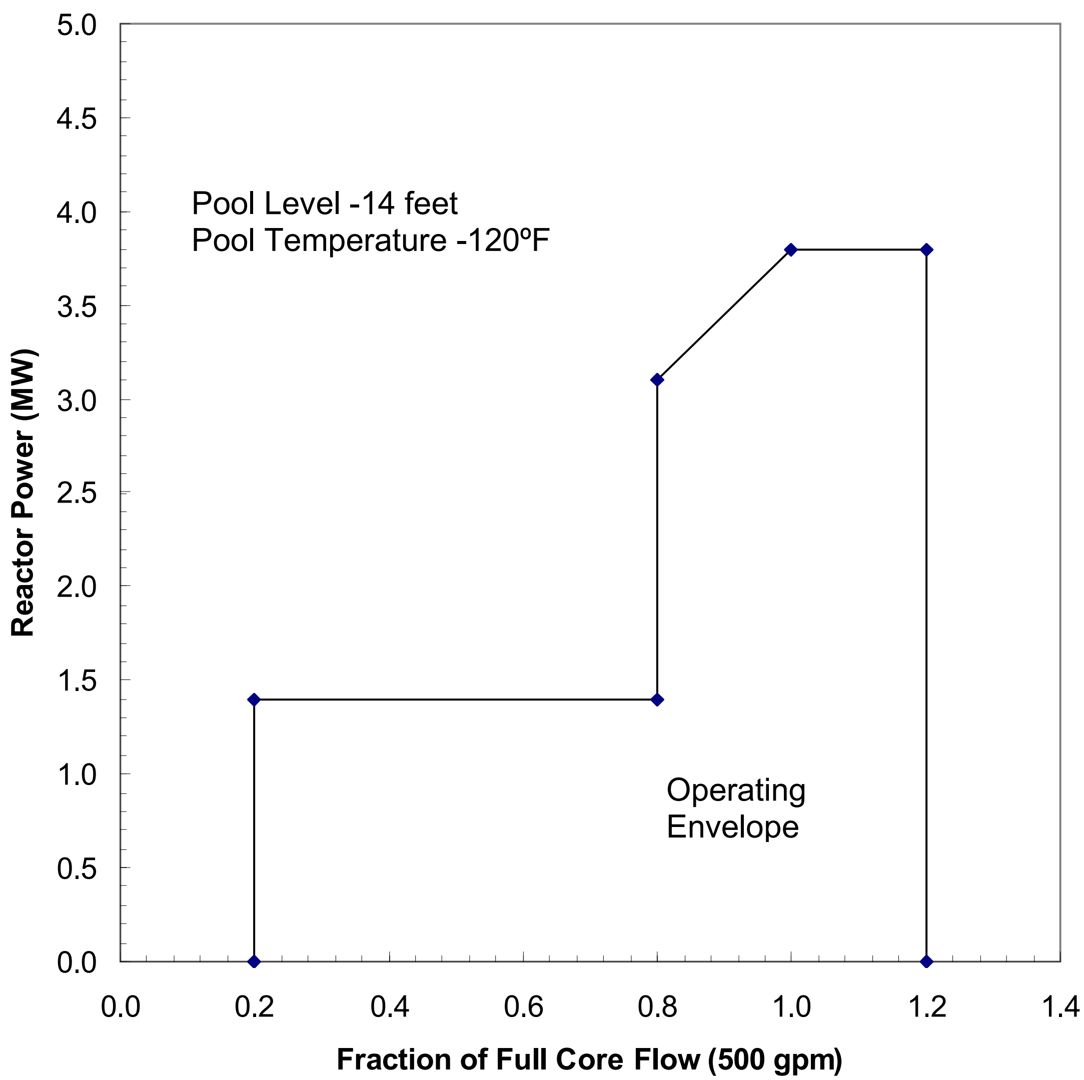


Figure 2.1-1: Power-Flow Safety Limit Curve

2.1.2 Safety Limits for Natural Convection Flow

Applicability

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

P Reactor Thermal Power

H Height of Water Above the Top of the Core

$\mathbf{T}_{inlet}$ Reactor Coolant Inlet Temperature

Objective

The objective is to assure that the integrity of the fuel clad is maintained.

Specification

Under the condition of natural convection flow, the Safety Limit shall be as follows:

a. The true value of reactor thermal power (P) shall not exceed 1.4 MWt.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

Bases

The criterion for establishing a Safety Limit with natural convection flow is established as the fuel clad temperature. This is consistent with Figure 2.1-1 for forced convection flow during a transient. The analysis of natural convection flow given in Appendix 3B and 3C of the SAR shows that at 1.4 MWt the maximum fuel clad temperature is 273°F which is well below the temperature at which fuel clad damage could occur. The flow with natural convection at this power is 98 gpm. This flow is based on data from natural convection tests with fuel assemblies of the same design performed in the prototype PULSTAR Reactor, as referenced in Section 3 of the SAR.

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	1.3 MWt (max.)
W	450 gpm (min.)
H	14 feet, 2 inches (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. These instrument uncertainties include a flow variation of ten percent, a pool level variation of two inches and a power level variation of seven percent.

The analysis presented in Section 3 of the SAR of a loss of flow transient indicates that if the interrelated variables were at their LSSS, as specified in 2.2.1 above, at the initiation of the transient, the Safety Limits specified in 2.1.1 would not be exceeded.

2.2.2 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	14 feet, 2 inches (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 Safety Limits from being exceeded. The specifications given above assure that an adequate safety margin exists between the LSSS and the SL for natural convection. The safety margin 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) which was the criterion used in establishing the Safety Limits (see Specification 2.1.2). It is desirable to minimize to the greatest extent practical, N-16 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 SAR, Section 3, 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.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Configuration

Applicability

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

Objective

The objective is to assure that the reactor will be operated within the bounds of established Safety Limits.

Specification

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

a. A maximum of twenty-five fuel assemblies.

b. A maximum of ten 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.

e. The maximum worth of a single fuel assembly shall not exceed 1590 pcm.

f. The total nuclear peaking factor in any fuel assembly shall not exceed 2.92.

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.

Specification 3.1.e provides assurances that a fuel loading accident will not result in a Safety Limit to be exceeded as discussed in SAR Section 13.2.2.1.

Specification 3.1.f provides assurances that core hot channel power are bounded by the SAR assumptions in Appendix 3-B.

3.2 Reactivity

Applicability

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

Objective

The objective is to assure that the reactor can be shutdown at all times and that the Safety Limits 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, with the shim rod fully withdrawn, and with experiments at their most reactive condition, relative to the cold critical condition, is greater than 400 pcm.

b. The excess reactivity is not greater than 3970 pcm.

c. The drop 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 100 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 3.2-1.

f. The sum of the absolute values of the reactivity worths of all experiments shall not be greater than 2890 pcm.

Table 3.2-1: Reactivity Limits for Experiments

Experiment	Limit
Movable	300 pcm or 100 pcm/sec, whichever is more limiting
Non-secured	1000 pcm
Secured	1590 pcm

Bases

The shutdown margin required by Specification 3.2.a assures 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 scrammable rod should be in the fully withdrawn position. Refer to Section 3.1.2.1.

The upper limit on excess reactivity ensures that an adequate shutdown margin is maintained.

The rod drop time required by Specification 3.2.c assures that the Safety Limit will not be exceeded during the flow reversal which occurs upon loss of forced convection coolant flow. The rise in fuel temperature due to heat storage is partially controlled by the reactivity insertion associated with the SCRAM. The analysis of this transient is based upon this SCRAM reactivity insertion taking the form of a ramp function of two second duration. This analysis is found in SAR Section 3.2.4 and Appendix 3B. The rod drop time is the time interval measured between the instant of a test signal input to the SCRAM Logic Unit 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 assures that the Safety Limit will not be exceeded during a startup accident due to a continuous linear reactivity insertion. Refer to SAR Section 13.

Experiments affecting the reactivity condition of the reactor are commonly categorized by the sign of the reactivity effect produced by insertion of the experiment. An experiment having a large reactivity effect of either sign can also produce an undesirable flux distribution that could affect the peaking factor used in the Safety Limit calculations and the calibration of Safety Channels.

The 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 cause the Safety Limit to be exceeded. Analyses indicate that the inadvertent reactivity insertion of these magnitudes will not result in consequences greater than those analyzed in the SAR Sections 3 and 13.

The total limit on reactivity associated with experiments ensures 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 in order to assure that the Safety Limits are not exceeded.

Specification

The reactor shall not be operated unless the reactor safety system channels described in Table 3.3-1 are operable.

Table 3.3-1: 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 ≤1.3 MW (LSSS) Enable for Flow/Flapper SCRAMs at ≤250 kW (LSSS)
c.	Linear Power Level	SCRAM at ≤1.3 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 ≥450 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 ≥14 feet 2 inches
j.	Manual SCRAM Button	SCRAM
k.	Reactor Key Switch	SCRAM
l.	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) May be bypassed for less than two minutes during the return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

Bases

The Startup Channel inhibit function assures the required startup neutron source is sufficient and in its proper location for the reactor startup, such that a minimum source multiplication count rate level is being detected to assure adequate information is available to the operator.

The reactor power level SCRAMs provide the redundant protection channels to assure 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 assure 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 a 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 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 assure safe operation of the reactor.

Specification

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

a. N-16 Power Measuring Channel when reactor power is greater than 500 kW

b. Control Rod Position Indications for each control rod and the Shim Rod

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

Bases

The N-16 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 stream for both normal and confinement modes.

3.5 Radiation Monitoring Equipment

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless the radiation monitoring equipment listed in Table 3.5-1 is operable.(1)(2)(3)

a. Three fixed area monitors operating in the Reactor Building with their setpoints as listed in Table 3.5-1.(1)(3)(4)

b. Particulate and gas building exhaust monitors continuously sampling air in the facility exhaust stack with their setpoints as listed in Table 3.5-1.(1)(3)(4)

c. The Radiation Rack Recorder.(5)

Table 3.5-1: 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	≤ 1000 Ar-41 AEC(6)	≤ 5,000 Ar-41AEC(6)
Stack Particulate	≤ 1000 Co-60 AEC(6)	≤ 5,000 Co-60 AEC(6)

(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. Refer to SAR Section 5.

(2) The Over-the-Pool Monitor may be bypassed for less than two minutes during return of a pneumatic capsule from the core to the unloading station or five minutes during removal of experiments from the reactor pool. Refer to SAR Section 5.

(3) Stack Gas and Particulate are based on the AEC quantities present in the ventilation flow stream as it exits the stack. Refer to SAR Section 10 for setpoint bases for the radiation monitoring equipment.

(4) May be bypassed for less than one 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. Refer to SAR Section 5.

(6) Airborne Effluent Concentrations (AEC) values from 10 CFR Part 20 Appendix B, Table 2

Bases

A continued evaluation of the radiation levels within the Reactor Building will be made to assure the safety of personnel. This is accomplished by the area monitoring system of the type described in Section 5 of the SAR.

Evaluation of the continued discharge air to the environment will be made using the information recorded from the particulate and gas 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 SAR Section 5.

To prevent unnecessary initiation of the evacuation confinement system during the return of a pneumatic capsule from the core to the unloading station or during removal of experiments from the reactor pool, the Over-the-Pool Monitor may be bypassed during the specified time interval. Refer to SAR Section 5.

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 assure 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 be moved within the pool area, unless the following equipment is operable, and conditions met:

Table 3.6-1: Required Main HVAC and Confinement Conditions

	Equipment/Condition	Function
a.	All doors, except the Control Room, and basement corridor entrance: self-latching, self-closing, closed and locked.	To maintain reactor building negative differential pressure (dp).(1)
b.	Control room and basement corridor entrance door: self-latching, self-closing and closed.	To maintain reactor building negative differential pressure.(2)
c.	Reactor Building 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.	To maintain reactor building negative differential pressure with reference to outside ambient.(3)
d.	Confinement system	Operable(4)(5)(7)
e.	Evacuation system	Operable(6)

(1) Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided audible and visual indications are available for the reactor operator to verify door status. Refer to SAR Section 5.

(2) Doors may be opened for periods of less than five minutes for personnel and equipment transport between corridor area and Reactor Building. Refer to SAR Section 5.

[3] During an interval not to exceed 30 minutes after a loss of dp is identified with Main HVAC operating, reactor operation may continue while the loss of dp is investigated and corrected. Refer to SAR Section 5.

[4] Operability 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 3.6-1, the building will be automatically placed in confinement as described in SAR Section 5.

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; and control the discharge of all air through a 100 foot stack on site. Section 5 of the SAR describes the confinement system sequence of operation.

The allowance for operation under a temporary loss of dp 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 will apply:

    i. No experiment will 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 activated products.

ii. Attempts will 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. Explosive material[1] shall not be allowed in the reactor. Experiments in which the material is considered to be potentially explosive, 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 will 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, 4th Ed., 1995, 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, 1994).

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 -238°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 will ensure the insertion of experiments will not negate the considerations implicit in the Safety Limits and thereby violate 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. The mass, fission rate and power are limited as indicated in Figure 3.8-1 and Table 3.8-1.

b. The reactor shall not be operated with a fueled experiment unless the ventilation system is operated in the confinement mode.

c. Specification 3.2 pertaining to reactivity shall be met.

d. Specification 3.7 pertaining to reactor experiments shall be met.

e. Specification 6.5 pertaining to the review of experiments shall be met.

   Each type of fueled experiment shall be classified as a new (untried) experiment with a documented review. The documented review shall include the following items:

   i. Meeting license requirements for the receipt, use, and storage of fissionable material.

   ii. Limiting the thermal power generated from the fissile material to ensure that the surface temperature of the experiment does not exceed the saturation temperature of the reactor pool water.

   iii. Radiation monitoring for detection of released fission products.

   iv. Design criteria related to meeting conditions given in Specifications 3.2 and 3.7.

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

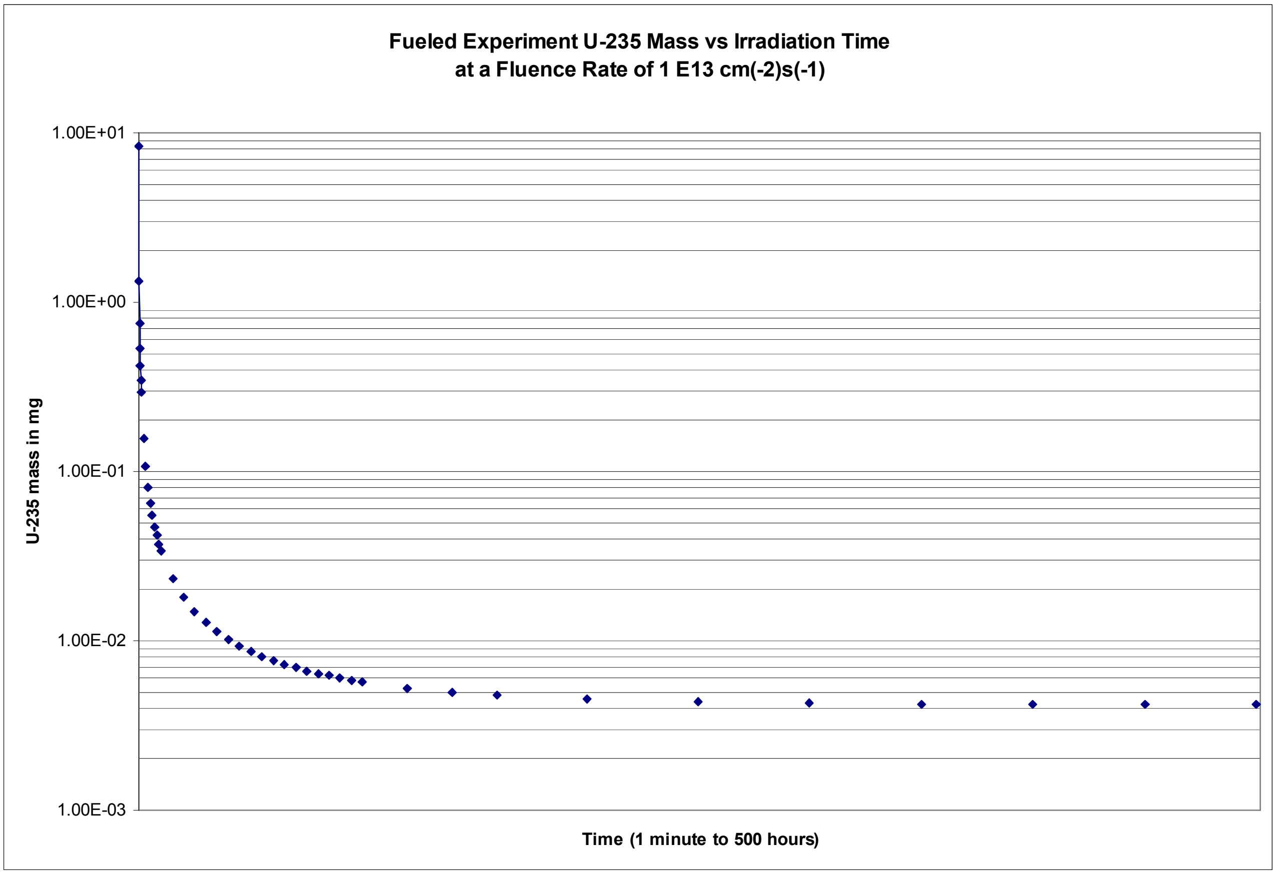


Figure 3.8-1

NOTE: **The mass at 500 hours may be used for periods up to 1 y (8760 hours).**

Table 3.8-1: Data for Fueled Experiments at a Fluence Rate of 1 E13 $cm^{-2}s^{-1}$

Irradiation Time, s	U-235 Mass mg	Mass-Fluence mg cm(-2)	Fission Rate f/s	Power milliwatts
6.00E+01	8.34E+00	5.00E+15	1.25E+11	4.01E+03
1.20E+02	4.75E+00	5.70E+15	7.12E+10	2.28E+03
1.80E+02	3.44E+00	6.19E+15	5.16E+10	1.65E+03
3.00E+02	2.30E+00	6.90E+15	3.45E+10	1.10E+03
6.00E+02	1.33E+00	7.98E+15	1.99E+10	6.39E+02
1.20E+03	7.55E-01	9.06E+15	1.13E+10	3.63E+02
1.80E+03	5.36E-01	9.65E+15	8.04E+09	2.57E+02
2.40E+03	4.18E-01	1.00E+16	6.27E+09	2.01E+02
3.00E+03	3.43E-01	1.03E+16	5.14E+09	1.65E+02
3.60E+03	2.92E-01	1.05E+16	4.38E+09	1.40E+02
7.20E+03	1.57E-01	1.13E+16	2.35E+09	7.54E+01
1.08E+04	1.08E-01	1.17E+16	1.62E+09	5.19E+01
1.44E+04	8.13E-02	1.17E+16	1.22E+09	3.90E+01
1.80E+04	6.55E-02	1.18E+16	9.82E+08	3.15E+01
2.16E+04	5.49E-02	1.19E+16	8.23E+08	2.64E+01
2.52E+04	4.74E-02	1.19E+16	7.11E+08	2.28E+01
2.88E+04	4.18E-02	1.20E+16	6.27E+08	2.01E+01
3.24E+04	3.74E-02	1.21E+16	5.61E+08	1.80E+01
3.60E+04	3.39E-02	1.22E+16	5.08E+08	1.63E+01
7.20E+04	1.81E-02	1.30E+16	2.71E+08	8.69E+00
1.08E+05	1.28E-02	1.38E+16	1.92E+08	6.15E+00
1.44E+05	1.02E-02	1.47E+16	1.53E+08	4.90E+00
1.80E+05	8.67E-03	1.56E+16	1.30E+08	4.16E+00
2.16E+05	7.66E-03	1.65E+16	1.15E+08	3.68E+00
2.52E+05	6.95E-03	1.75E+16	1.04E+08	3.34E+00
2.88E+05	6.42E-03	1.85E+16	9.62E+07	3.08E+00
3.24E+05	6.03E-03	1.95E+16	9.04E+07	2.90E+00
3.60E+05	5.72E-03	2.06E+16	8.57E+07	2.75E+00
3.96E+05	5.27E-03	2.09E+16	7.90E+07	2.53E+00
4.32E+05	4.97E-03	2.15E+16	7.45E+07	2.39E+00
4.68E+05	4.77E-03	2.23E+16	7.15E+07	2.29E+00
7.20E+05	4.51E-03	3.25E+16	6.76E+07	2.17E+00
1.08E+06	4.27E-03	4.61E+16	6.40E+07	2.05E+00
1.44E+06	4.21E-03	6.06E+16	6.31E+07	2.02E+00
1.80E+06	4.19E-03	7.54E+16	6.28E+07	2.01E+00
2.16E+06	4.19E-03	9.05E+16	6.28E+07	2.01E+00
4.32E+06	4.19E-03	1.81E+17	6.28E+07	2.01E+00
4.32E+06	4.19E-03	1.81E+17	6.28E+07	2.01E+00
1.73E+07	4.19E-03	7.24E+17	6.28E+07	2.01E+00
3.15E+07	4.19E-03	1.32E+18	6.28E+07	2.01E+00

Bases

NUREG 1537 provides guidelines for the format and content of non-power reactor licensing. Guidelines on operating conditions and accident analysis for fueled experiments are given in NUREG 1537. These guidelines include (1) actuation of engineered safety features (ESF) to prevent or mitigate the consequences of damage to fission product barriers caused by overpower or loss of cooling events, (2) use of ESF to control of radioactive material released by accidents, (3) radiation monitoring of fission product effluent and accident releases, (4) accidental analysis for loss of cooling or other experimental malfunction resulting in liquefaction or volatilization of fissile materials, (5) accident analysis for catastrophic failure of the experiment in the reactor pool or air, (6) accident analysis for insertion of excess reactivity leading to fuel melting, and (7) emergency plan activation and classification.

The limitations given in Specification 3.8 ensure that (1) fueled experiments performed in experimental facilities at the reactor prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure, (2) radiation doses to occupational personnel and the public and radioactive material releases are ALARA, (3) adequate radiation monitoring is in place, and (4) in the event of failure of a fueled experiment with the subsequent release of radioactive material, the resulting dose to personnel and the public at any location are well within limits set in 10 CFR Part 20.

Specification 3.8.e ensures that each type of fueled experiment is reviewed, approved, and documented as required by Specification 6.5. This includes (1) meeting applicable limitations on experiments given in Specifications 3.2 and 3.7, (2) limiting the amount of fissile material to ensure that experimental reactivity conditions are met and that radiation doses are well within 10 CFR Part 20 limits following maximum fission product release from a failed experiment, and (3) limiting the thermal power generated from the fissile material to ensure that the surface temperature of the experiment does not exceed the saturation temperature of the reactor pool water.

3.9 Primary Coolant

Applicability

This specification applies to the water quality and flow path 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 $\geq$500 kΩ·cm.

b. The pH shall be within the range of 5.5 to 7.5.

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.

4.0 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.

The intent of the surveillance interval (e.g., annually, but not to exceed fifteen months) is to maintain an average cycle, with occasional extensions as allowed by the interval tolerance. If it is desired to permanently change the scheduled date of surveillance, the particular surveillance item will be performed at an earlier date and the associated interval normalized to this revised earlier date. In no cases will permanent scheduling changes, which yield slippage of the surveillance interval routine scheduled date, be made by using the allowed interval tolerance.

4.1 Fuel

Applicability

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

Objective

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

Specification

a. All fuel assemblies shall be visually inspected for physical damage biennially but at intervals not to exceed thirty (30) months.

b. The reactor will be operated at such power levels necessary to determine if an assembly has had fuel pin cladding failure.

Bases

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) will be disassembled to confirm and isolate damaged fuel pins. Damaged fuel pins will 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 shim rod, 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 but at intervals not to exceed fifteen (15) months for the steady state core in current use. The reactivity worth of all rods shall be determined for any new core or rod configuration, prior to routine operation.

b. Control rod drop times(1) and control rod drive times shall be determined:

   i. Annually but at intervals not to exceed fifteen (15) months.

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

c. The control rods shall be visually inspected biennially but at intervals not to exceed thirty (30) months.

d. The values of excess reactivity and shutdown margin shall be determined monthly, but at intervals not to exceed six (6) weeks, and for new core configurations.

(1) Applies only to magnetically coupled rods.

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 drop time measurements are made to determine whether the rods are functionally operable. These time measurements may also be utilized in reactor transient analysis.

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 will document proper control rod system reassembly after maintenance and recorded post-maintenance data will identify significant trends in rod performance.

4.3 Reactor Instrumentation and 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 instrumentation and Safety Systems will remain operable and will prevent the Safety Limits 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 operation each day, or prior to each operation extending more than one day.

c. A channel calibration of the N-16 Channel shall be made semi-annually, but at intervals not to exceed seven and one-half (7½) months. A calorimetric measurement shall be performed to determine the N-16 detector current associated with full power operation.

d. A channel calibration of the following channels shall be made semi-annually but at intervals not to exceed seven and one-half (7½) months.[(1)]

   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

[(1)] A channel calibration shall also be required after repair of a channel component that has the potential of affecting the calibration of the channel.

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. The calorimetric calibration of the reactor power level, in conjunction with the N-l6 Channel, provides a continual reference for adjustment of the Linear, Log N and Safety Channel detector positions.

4.4 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. The area and stack monitoring systems shall be calibrated annually but at intervals not to exceed fifteen (15) months.

b. The setpoints shall be verified weekly, but at intervals not to exceed ten (10) days.

Bases

These systems provide continuous radiation monitoring of the Reactor Building with a check of readings performed prior to and during reactor operations. Therefore, the weekly verification of the setpoints in conjunction with the annual calibration is adequate to identify long term variations in the system operating characteristics.

4.5 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 will be checked monthly but at intervals not to exceed six (6) weeks.(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 but at intervals not to exceed seven and one-half (7½) months to verify they are operable.

d. The control room differential pressure (dp) gauges shall be calibrated annually but at intervals not to exceed fifteen (15) months.

e. The confinement filter train shall be tested biennially but at intervals not to exceed thirty (30) months 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)

f. The air flow rate in the confinement stack exhaust duct shall be determined annually but at intervals not to exceed fifteen (15) months. The air flow shall be not less than 600 CFM.

(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 will verify that the confinement of the Reactor Building is maintained as described in Section 5 of the SAR.

4.6 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, but at intervals not to exceed eighteen (18) days. 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 pH and resistivity measurements.

b. The secondary coolant shall be analyzed bi-weekly, but at intervals not to exceed eighteen (18) days. 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 will allow detection of fuel clad failure, while neutron activation analysis will give corrosion data associated with primary system components in contact with the coolant. Refer to SAR Section 10. The detection of activation or fission products in the secondary coolant provides evidence of a primary heat exchanger leak. Refer to SAR Section 10.

5.0. 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 2.25×10^9 cm^3 (refer to SAR Section 13 analysis).

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 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 5.2-1 depicts the licensed area as being within the operations boundary.

5.3. Fuel Storage

Fuel, including fueled experiments and fuel devices not in the reactor, 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 except 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 percent silver, 15 percent indium, and 5 percent cadmium with nickel cladding. Three of these neutron absorbing blades are magnetically coupled and have scramming capability. The remaining neutron absorbing blade is non-scrammable. One of the scrammable rods may be used for automatic servo-control of reactor power. When in use, the servo-control maintains a constant power level as indicated by the Linear Power Channel.

5.5 Primary Coolant System

The primary coolant system consists of the aluminum lined reactor tank, a N-16 delay tank, a pump, and heat exchanger, and associated stainless steel piping. The nominal capacity of the primary system is 15,600 gallons. Valves are located adjacent to the biological shield to allow isolation of the pool, and at major components in the primary system to permit isolation.

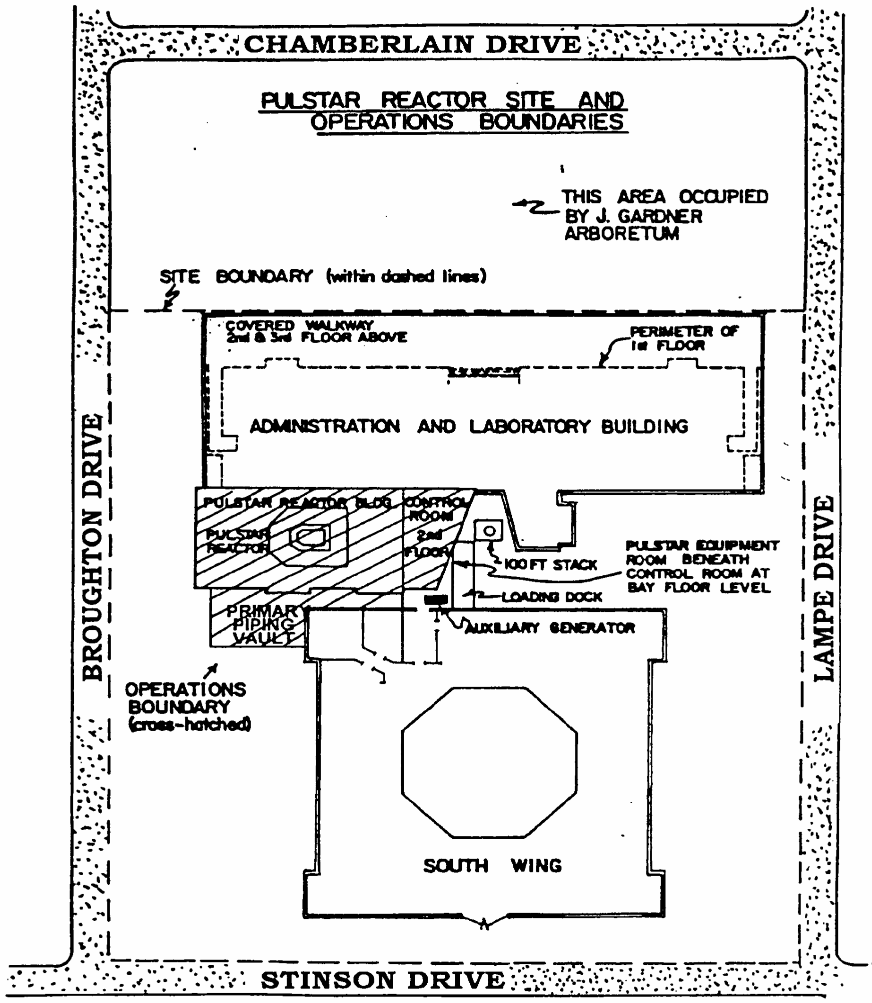


Figure 5.2-1: NCSU PULSTAR Reactor Site Map

6.0 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 6.1-1.

6.1.1 Organizational Structure:

The reporting chain is given in Figure 6.1-1. The following specific organizational levels (as defined by ANSI/ANS-15.1-1990) 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.

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 1.2.9 and 6.5, and making minor 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 assuring 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 6.1-1.

6.1.2 Responsibility

Responsibility for the safe operation of the PULSTAR Reactor shall be with the chain of command established in Figure 6.1-1.

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 Designed 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 Designed 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 at the facility or its administrative offices:

a. Initial startup and approach to power.

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

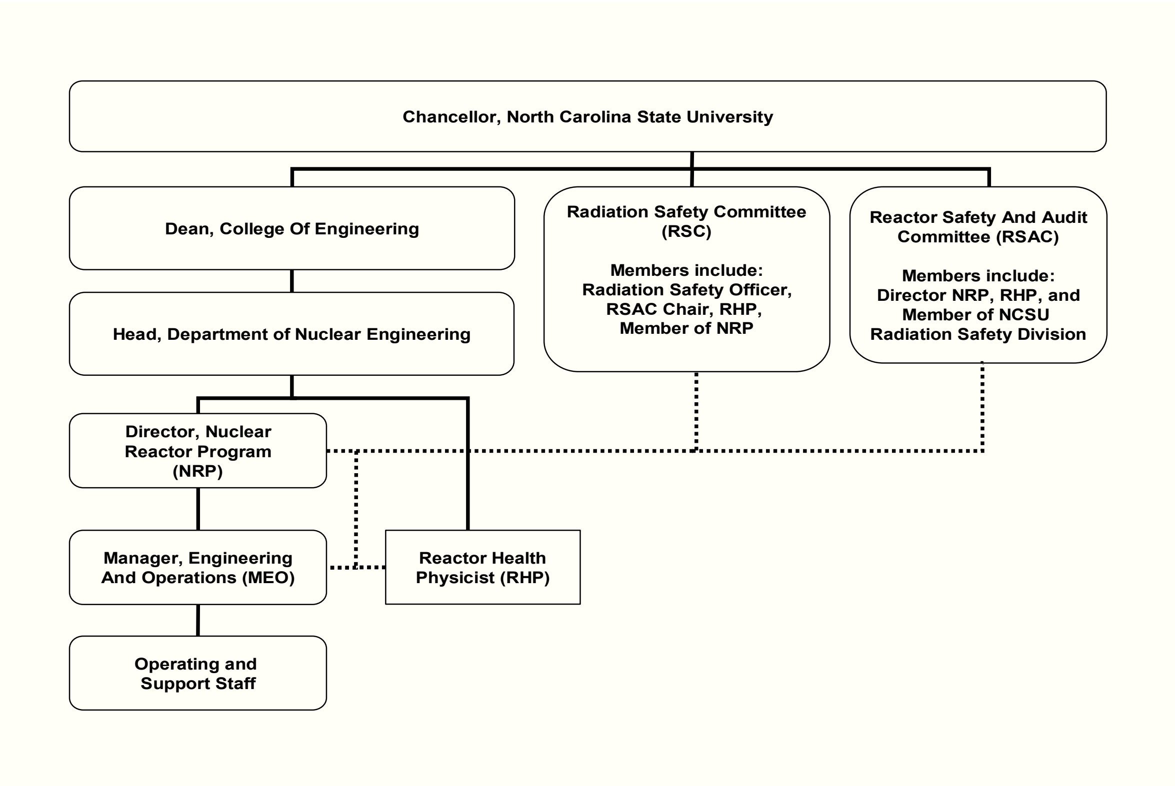
c. Relocation of any in-core experiment with a reactivity worth greater than one dollar (730 pcm).

d. Recovery from unplanned or unscheduled shutdown or significant power reduction (documented verbal concurrence from a licensed Senior Reactor Operator is required).

6.1.5 Selection and Training

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

Figure 6.1-1: NCSU PULSTAR Reactor Organizational Chart



NOTES: Line of direct communication ———

Line of advice and liaison – – –

Nuclear Reactor Program (NRP) includes:

- Director, NRP
- 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 these Technical Specifications and facility procedures.

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).

RSAC has the primary responsibility to ensure that the reactor is operated and used 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 and RSAC Composition and Qualifications

a. 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. RSC membership shall include the University Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

b. 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 general 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 general faculty

The remaining RSAC members are as follows:

iv. Reactor Health Physicist (RHP)

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

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

vii. 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.

6.2.2 RSC and RSAC Rules

a. RSC and RSAC 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. 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.

d. A quorum of RSC or 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 6.1-1 shall not constitute a majority of the RSC or RSAC quorum.

6.2.3 RSC and RSAC 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 and approved by the RSAC:

   i. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures which have safety significance meet facility license and Technical Specification requirements.

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

   iii. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

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

c. The following items shall be reviewed by the RSC and 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 1.2.24.

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

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 prior to the next scheduled RSAC meeting.

6.2.4 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.

Deficiencies uncovered that affect reactor safety shall be immediately 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.

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 6.1-1.

6.4 Operating Procedures

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

a. Startup, operation and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the reactor.

c. Maintenance of major components of systems that could have an affect on reactor safety.

d. Surveillance checks, calibrations and inspections required by the facility license or Technical Specifications or those that may have an affect on the 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 could affect reactor safety or core reactivity.

g. Implementation of the Emergency Plan and Security Plan.

Substantive changes to the above procedures shall be made effective only after documented review and approval by the RSAC and by the Manager of Engineering and Operations.

Minor modifications to the original procedures which do not change their original intent may be made by the Manager of Engineering and Operations, but the modifications shall be approved by the Director of the Nuclear Reactor Program within fourteen (14) days.

Temporary deviations from procedures may be made by Designed Senior Reactor Operator as defined by Specification 6.1.3.c or the Manager of Engineering and Operations, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to the Director of the Nuclear Reactor Program.

6.5 Review of Experiments

6.5.1 New (untried) Experiments

All new experiments or class of experiments, referred to as "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 new experiments shall be 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 a tried experiment, then the request may be approved.

Substantive changes to previously approved experiments 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 1.2.24, 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 1.2.24, there shall be a report not 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 (in megawatt days) 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 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 Waste (summarized by quarter)

i. Radioactivity released during the reporting period:

1. Number of batch releases.
2. Total radioactivity released (in microcuries).
3. Total liquid volume required (in liters).
4. Diluent volume required (in liters).
5. Tritium activity released (in microcuries)
6. Total (yearly) tritium released.
7. Total (yearly) activity released.

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 (in microcuries) prior to disposal.
3. Total volume of liquid in tank (in liters).

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.

Gaseous Waste

i. Radioactivity discharged during the reporting period (in curies) for:

   1. Gases

   2. Particulates, with half lives greater than eight days.

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

Solid Waste

i. The total amount of solid waste packaged (in cubic feet).

ii. The total activity involved (in curies).

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

g. A summary of radiation exposures received by facility personnel and visitors, including pertinent details of significant exposures.

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

i. A description of environmental surveys 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 plant operation and maintenance.

b. Principal maintenance activities.

c. Reportable Events.

d. Equipment and components surveillance activities as detailed in Specification 4.

e. Experiments performed with the reactor.

f. Changes to Operating Procedures.

g. Facility radiation and contamination surveys other than those used in support of personnel radiation monitoring.

h. Audit summaries.

i. RSC and RSAC meeting minutes.

6.8.2 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. Fuel inventories and transfers.

e. Drawings of the reactor facility.

6.8.3 Records to be retained for at least one (1) license period 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.