RHODE ISLAND ATOMIC ENERGY COMMISSION

DOCKET NO. 50-193

RHODE ISLAND NUCLEAR SCIENCE CENTER

RENEWED FACILITY OPERATING LICENSE

License No. R-95

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

   A. The application for renewal of Facility Operating License No. R-95 filed by the Rhode Island Atomic Energy Commission ("the licensee"), dated May 3, 2004, as supplemented on January 19, February 4, August 6, August 18, September 3, September 8, November 8, November 26, December 7, and December 14, 2010; January 24, February 24, and July 15, 2011; March 15, September 16, and December 19, 2013; February 24, April 28, and June 30, 2014; August 7 and August 11, 2015; and January 20, February 26, March 1, April 21, July 20, October 6, November 1, November 14, December 1, December 8, December 13, and December 15, 2016 ("the application"), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the Commission's rules and regulations set forth in Title 10, Chapter I, of the *Code of Federal Regulations* (10 CFR);

   B. Construction of the Rhode Island Nuclear Science Center ("the facility") Materials Testing Reactor (MTR) - type nuclear research reactor was completed in substantial conformity with the Construction Permit No. CPRR-73, issued on August 27, 1962, and the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

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

   D. There is reasonable assurance that: (i) the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) 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 license in accordance with the rules and regulations of the Commission

F. The applicable provisions of 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements,” have been satisfied; the issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

G. The issuance of this license is in accordance with 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” of the Commission’s regulations and all applicable requirements have been satisfied; and

H. The receipt, possession and use of byproduct and special nuclear materials as authorized by this facility operating license will be in accordance with the Commission's regulations in 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.”

2. Accordingly, Facility Operating License No. R-95 is hereby renewed in its entirety to read as follows:

A. This license applies to the Rhode Island Nuclear Science Center (herein “the facility”) MTR-type nuclear research reactor owned by the Rhode Island Atomic Energy Commission (herein “the licensee”), located on the Narragansett Bay campus of University of Rhode Island in Narragansett, Rhode Island, and described in the licensee’s application for license renewal, dated May 3, 2004, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Rhode Island Atomic Energy Commission as follows:

1. Pursuant to subsection 104c of the Act and 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” to possess, use, and operate the facility as a utilization facility at the designated location in accordance with the procedures and limitations described in the application and set forth in this license.

2. Pursuant to the Act and 10 CFR Part 70, the following activities are included:

a. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 10 kilograms of contained uranium-235 enriched to less than 20 percent in the form of MTR-type reactor fuel;

b. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 32 grams of plutonium encapsulated in two plutonium-beryllium neutron sources for reactor startup,

c. to receive, possess, and use, but not separate, in connection with the operation of the facility, up to 40 grams total of special nuclear material, of any enrichment, in the form of detectors, fission plates, foils, and solutions; and,

d. to receive, possess, and use, but not separate, in connection with the operation of the facility, such special nuclear material as may be produced by the operation of the facility.

3. Pursuant to the Act and 10 CFR Part 30, the following activities are included:

a. to receive, possess, and use, in connection with the operation of the facility, a sealed antimony-beryllium neutron startup source; and,

b. to receive, possess, and use, in connection with operation of the facility, such byproduct material as may be produced by operation of the reactor, which can not be separated except for byproduct material produced in non-fueled reactor experiments.

C. This license shall be deemed to contain, and is subject to the conditions specified in 10 CFR Parts 20, 30, 40, 50, 51, 55, 70, and 73 of the Commission's regulations; is subject to all provisions of the Act, and to the rules, regulations and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

The licensee is authorized to operate the reactor at a steady-state power level up to a maximum of 2000 kilowatts (thermal) in accordance with the limitations in the Technical Specifications.

2. Technical Specifications

The Technical Specifications contained in Appendix A are hereby incorporated in their entirety in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. <u>Physical Security Plan</u>

The licensee shall maintain and fully implement all provisions of the Commission-approved physical security plan, including changes made pursuant to the authority of 10 CFR 50.54(p). The approved physical security plan, entitled "Rhode Island Nuclear Science Center Reactor, Physical Security Plan for the Protection of Special Nuclear Material, Facility Operating License R-95, Docket Number 50-193, May 25, 2016," consists of documents withheld from public disclosure pursuant to 10 CFR 73.21.

This license is effective as of the date of issuance and shall expire at midnight, 20 years from the date of issuance.

For the Nuclear Regulatory Commission

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A, Technical Specifications

Date of Issuance: January 5, 2017

Appendix A

TO

FACILITY LICENSE NO. R-95
DOCKET NO. 50-193

TECHNICAL SPECIFICATIONS AND BASES

RHODE ISLAND ATOMIC ENERGY COMMISSION (RIAEC)

RHODE ISLAND
NUCLEAR SCIENCE CENTER (RINSC)
REACTOR

December 19, 2016

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1.0 Introduction

Scope

This document constitutes the Rhode Island Nuclear Science Center (RINSC) Technical Specifications for Facility License number R-95 as required by 10 CFR Part 50.36 and supersedes all prior Technical Specification revisions and/or amendments. This document includes the "bases" to support the selection and significance of the specifications. The bases are included for information purposes only, they are not part of the Technical Specifications and do not constitute limitations or requirements to which the licensee must adhere.

Format

These specifications are formatted to NUREG-1537 and ANSI/ANS 15.1/2007.

Definitions

1.1 Channel

A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

1.2 Channel Calibration

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

1.3 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 parameter.

1.4 Channel Test

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

1.5 Confinement

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

1.6 Control Rod

A control rod is a device fabricated from neutron absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.7 Core Configuration

The core configuration includes the number, type, and arrangement of fuel elements, reflector elements, and control rods occupying the core grid.

1.8 Excess Reactivity

Excess reactivity is that amount of reactivity that would exist if all of the control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical when the core is in the reference core condition with the maximum allowed experiment worth installed.

1.9 Experiment

An experiment is any operation that is designed to investigate non-routine reactor characteristics, or any material or device not associated with the core configuration or the reactor safety systems that is intended for irradiation within the pool or an experimental facility. Hardware that is rigidly secured to a core or shield structure so as to be part of its design to carry out experiments is not normally considered to be an experiment.

1.10 Experimental Facility

An experimental facility is any structure or device which is intended to guide, orient, position, manipulate, or otherwise facilitate a multiplicity of experiments of similar character.

1.11 Explosive Material

Explosive material is any material determined to be within the scope of Title 18, United States Code, Chapter 40, "Importation"; Manufacture, Distribution and Storage of Explosive Materials," and any material classified as an explosive by the Department of Transportation in the Hazardous Material regulations (Title 49 CFR, Parts 100-199).

1.12 Fixed Experiment

A fixed experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces shall be substantially greater than other forces to which the experiment might be subjected that are normal to the operating environment of the experiment, or that can arise as a result of a credible malfunction.

1.13 Limiting Conditions for Operation (LCO)

The limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the reactor.

1.14 Limiting Safety System Setting (LSSS)

Limiting Safety System Settings are settings for automatic protective devices related to those variables having significant safety functions, and chosen so that automatic protective action will correct an abnormal situation before a safety limit is exceeded.

1.15 May

The word "may" is used to denote permission, neither a requirement nor a recommendation.

1.16 Mode of Operation

Mode of operation refers to the type of core cooling that is employed while the reactor is operating. The two modes of operation are forced convection cooling mode which supports reactor operation up to 2 MW_t, and natural convection cooling mode which supports reactor operation up to 100 kW_t.

1.17 Moveable Experiment

A moveable 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.

1.18 Operable

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

1.19 Operating

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

1.20 Protective Action

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

1.21 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of:

1.21.1 Insertion or removal from the core,

1.21.2 Intended or anticipated changes in position, or

1.21.3 Credible malfunctions that alter experiment position or configuration.

1.22 Reactor Operating

The reactor is operating whenever it is not secured or shut down.

1.23 Reactor Operator (RO)

A reactor operator is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the RINSC reactor.

1.24 Reactor Operator Trainee

A reactor operator trainee is an individual who is authorized to manipulate the controls of the RINSC reactor under the direct supervision of a licensed operator.

1.25 Reactor Safety Systems

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.

1.26 Reactor Secured

The reactor is secured when under optimal conditions of moderation and reflection either:

1.26.1 There is insufficient moderator available in the reactor to attain criticality, or

1.26.2 There is insufficient fissile material present in the reactor to attain criticality, or

1.26.3 All of the following conditions exist:

1.26.3.1 All four shim safety blades and the regulating rod are fully inserted or other safety devices are in the shutdown position, as required by technical specifications, AND;

1.26.3.2 The master switch is in the off position and the key is removed from the lock, AND;

1.26.3.3 No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, AND;

1.26.3.4 No experiments are being moved or serviced that have a reactivity worth of greater than 0.6%Δk/k when moved, or

1.26.4 For the purpose of centering Shim Safety Blade armatures only, with the key in the master switch and the master switch in the ON position, the reactor is secured if ALL of the following conditions are met:

1.26.4.1 The only task being performed by the RO is centering the Shim Safety Blade armatures, AND;

1.26.4.2 The control room door remains closed and locked while the RO is performing the centering operation, AND;

1.26.4.3 The RO does not leave the pool top level of the confinement building, AND;

1.26.4.4 The RO maintains a visual line of sight to the control room door and the top of the stairwell leading to the pool top level, AND;

1.26.4.5 The RO notifies the individual logged as the second person in the facility (ref TS requirement 6.1.3.1.2.2) when leaving the control room and when returning to the control room, AND;

1.26.4.6 The scram relays are NOT reset, AND;

1.26.4.7 The Shim Safety Blade magnets are de-energized, AND;

1.26.4.8 ALL Shim Safety Blades indicate they are on the bottom.

1.27 Reactor Shutdown

The reactor is shut down if it is subcritical by at least 0.75%Δk/k in the reference core condition with the reactivity of all installed experiments included.

1.28 Readily Available on Call

Readily available on call shall mean that the individual is aware that they are on call, can be contacted within ten minutes, and is within a 30 minute driving time from the reactor building.

1.29 Reference Core Condition

The condition of the core when it is at ambient temperature and the reactivity of xenon is less than 0.2%Δk/k.

1.30 Regulating Rod (RR)

The regulating rod is a low worth control rod used primarily to maintain an intended power level and does not have scram capability. Its position may be varied manually or automatically by servo-controller.

1.31 Reportable Occurrence

A reportable occurrence is any of the following:

1.31.1 A violation of the safety limit,

1.31.2 An uncontrolled or unplanned release of radioactive material which results in concentrations of radioactive materials inside or outside the restricted area in excess of the limits specified in 10 CFR Part 20,

1.31.3 Operation with a safety system setting less conservative than the limiting safety system setting established in the Technical Specifications,

1.31.4 Operation in violation of a limiting condition for operation established in the Technical Specifications,

1.31.5 A reactor safety system component malfunction or other component or system malfunction which could, or threaten to, render the safety system incapable of performing its intended safety functions unless the cause is due to maintenance, 1.31.6 An uncontrolled or unanticipated change in reactivity in excess of 0.75%Δk/k from which the cause is unknown,

1.31.7 Abnormal and significant degradation of the fuel cladding,

1.31.8 Abnormal and significant degradation of the primary coolant boundary, or the confinement boundary,

1.31.9 An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

1.32 Restricted Area

Restricted areas are areas in which access is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

1.33 Safety Channel

A safety channel is a channel in the reactor safety system.

1.34 Safety Limits

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of the principal barrier which guard against the uncontrolled release of radioactivity. The principal barrier is the fuel element cladding.

1.35 Scram Time

Scram time is the elapsed time between the initiation of a scram signal and the time when the blades are fully inserted in the core.

1.36 Senior Reactor Operator (SRO)

A senior reactor operator is an individual who is licensed under 10 CFR Part 55 to manipulate the controls of the RINSC reactor and to direct the licensed activities of reactor operators.

1.37 Shall

The word “shall” is used to denote a requirement.

1.38 Shim Safety Blade (SSB)

A shim safety blade is a control rod of high reactivity worth used primarily to make course adjustments to power level, and to provide a means for very fast reactor shutdown by having scram capability.

1.39 Should

The word “should” is used to denote a recommendation.

1.40 Shutdown Margin

Shutdown Margin is the minimum shutdown reactivity necessary to provide confidence that under reference core conditions, the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition including maximum experiment worth and with the most reactive Shim Safety Blade and the Regulating Rod in their most reactive positions and that the reactor shall remain subcritical without further operator action.

1.41 Site Boundary

That line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

1.42 Surveillance Activities

Surveillance activities are activities that are performed on a periodic basis for the purpose of verifying the integrity and operability of facility infrastructure and equipment which provides confidence that these components will perform their intended functions.

1.43 Surveillance Intervals

Maximum intervals are to provide operational flexibility, not to reduce frequency. Established frequencies shall be maintained over the long term. Allowable surveillance intervals shall not exceed the following:

- 1.43.1 5 years (interval not to exceed 6 years).
- 1.43.2 2 years (interval not to exceed 2 1/2 years).
- 1.43.3 Annual (interval not to exceed 15 months).
- 1.43.4 Semiannual (interval not to exceed 7 1/2 months).
- 1.43.5 Quarterly (interval not to exceed 4 months).
- 1.43.6 Monthly (interval not to exceed 6 weeks).
- 1.43.7 Weekly (interval not to exceed 10 days).
- 1.43.8 Daily (shall be done during the calendar weekday).

1.44 True Value

The true value is the actual value of a parameter.

1.45 Unscheduled Shutdown

An unscheduled shutdown is any unplanned shutdown of the reactor that is not associated with testing or check out operations, which is caused by:

1.45.1 Actuation of the reactor safety system,

1.45.2 Operator error,

1.45.3 Equipment malfunction, or

1.45.4 Manual shutdown in response to conditions that could adversely affect safe operation.

1.46 Water Reactive Material

A material that explodes; violently reacts; produces flammable, toxic or other hazardous gases; or evolves enough heat to cause auto-ignition or ignition of combustibles upon exposure to water or moisture.

2.0 Safety Limits and Limiting Safety System Settings

2.1 Safety Limit

Applicability:

This specification applies to fuel that is loaded in the core.

Objective:

The objective of this specification is to ensure that the integrity of the fuel cladding is not damaged due to overheating.

Specification:

The temperature of the reactor fuel cladding shall be less than or equal to 530^{o} C.

Basis:

NUREG 1313 shows that the integrity of the fuel cladding will not be damaged due to overheating provided that the cladding temperature does not exceed 530^{o} C.

2.2 Limiting Safety System Settings

2.2.1 Limiting Safety System Settings for Natural Convection Mode Operation

Applicability:

These specifications apply to the safety channels that monitor variables that directly impact fuel cladding temperature during natural convection mode operation of the reactor.

Objective:

The objective of these specifications is to ensure that the safety limit for the reactor cannot be exceeded during natural convection mode operation.

Specifications:

2.2.1.1 The limiting safety system setting for reactor thermal power shall be 115 kW.

2.2.1.2 The limiting safety system setting for the height of coolant above the top of the uranium silicide fuel shall be 23 feet 7 inches.

2.2.1.3 The limiting safety system setting for the bulk pool temperature shall be 127^{o} F.

Bases:

This combination of specifications was set to prevent the cladding temperature from approaching the 530^{o} C value at which damage to the fuel cladding could occur, under both steady state, and transient conditions.

The thermal-hydraulic analysis for steady state power operation under natural convection cooling conditions shows that the fuel cladding temperature will remain significantly below the threshold for cladding damage during steady state operation of the reactor if the following combination of limits are in place: The coolant height above the uranium silicide fuel is at least 23 feet 6.5 inches, The bulk pool temperature is no greater than 130^{o} F. Under these conditions for coolant height and bulk pool temperature, peak channel power would have to reach 1.78 kW in order for the onset of nucleate boiling to occur, which corresponds to a fuel cladding temperature that is below the 530^{o} C value at which damage to the fuel cladding could occur. The hottest channel reaches a peak power of 1.78 kW when core power is 369 kW. Consequently, there is a margin of

$$369 \text{ kW} - 115 \text{ kW} = 254 \text{ kW}$$

between the LSSS and the point at which onset of nucleate boiling would occur. The transient analysis for natural convection cooling was performed for the most conservative case in which all of the safety channels are at their respective limiting trip values when the transient is terminated. The analysis shows that the peak fuel cladding temperature will be approximately 78.9^{o} C during a transient in which the following combination of limits are in place:

- The initial power level is no greater than 100 kW,
- The coolant height above the uranium silicide fuel is at least 23 feet 7 inches,

The bulk pool temperature is no greater than 130^o F, and
- The transient is terminated by an over power trip at 125 kW.

Under these conditions for the most conservative case, there is a margin of:

$$530^o\ C - 78.9^o\ C = 451.1^o\ C$$

Measurement uncertainty was based on the nominal operating values of 100 kW and 108^o F for the power and pool temperature respectively, and has been determined to be:

- Power Level ± 10 kW
- Coolant Height 0.5 inches
- Temperature 3^o F

Consequently, the bases for these specifications are:

Specification 2.2.1.1 sets the limiting safety system setting for reactor thermal power to be 115 kW. The analyses show that cladding damage will not occur under any condition if initial power is no greater than 369 kW. Taking into consideration a 10 kW measurement error, if the LSSS is set at 115 kW, then the Limiting Trip Value could be as high as 125 kW, which still leaves a margin of 244 kW between the LSSS and the most conservative true value of the power level used in the analysis.

Specification 2.2.1.2 sets the limiting safety system setting for the height of coolant above the top of the uranium silicide fuel to be 23 feet 7 inches. The analyses show that cladding damage will not occur under any condition if the height is no less than 23 feet 6.5 inches. Taking into consideration a 0.5 inch measurement error, if the LSSS is 23 feet 7 inches, then the Limiting Trip Value could never be less than 23 feet 6.5 inches.

Specification 2.2.1.3 sets the limiting safety system setting for the bulk pool temperature to be 127^o F. The analyses show that cladding damage will not occur under any condition if the pool temperature is no greater than 130^o F. Taking into consideration a 3^o F in measurement error, if the LSSS is 127^o F, then the Limiting Trip Value could never be greater than 130^o F.

2.2.2 Limiting Safety System Settings for Forced Convection Mode of Operation

Applicability:

These specifications apply to the safety channels that monitor variables that directly impact fuel cladding temperature during forced convection mode operation of the reactor.

Objective:

The objective of these specifications is to ensure that the safety limit for the reactor cannot be exceeded during forced convection mode operation.

Specifications:

2.2.2.1 The limiting safety system setting for reactor thermal power shall be 2.3 MW.

2.2.2.2 The limiting safety system setting for the height of coolant above the top of the uranium silicide fuel shall be 23 feet 7 inches.

2.2.2.3 The limiting safety system setting for the primary coolant inlet temperature shall be 122^{o} F.

2.2.2.4 The limiting safety system setting for the primary coolant flow rate shall be 1560 gpm.

Bases:

This combination of specifications was set to prevent the cladding temperature from approaching the 530^{o} C value at which damage to the fuel cladding could occur, under both steady state, and transient conditions.

The thermal-hydraulic analysis for steady state power operation under forced convection cooling conditions shows that the fuel cladding temperature will remain significantly below the threshold for cladding damage during operation of the reactor if the following combination of limits are in place:

- The steady state power level is less than 2.5 MW,
- The coolant height above the uranium silicide fuel is at least 23 feet 6.5 inches,
- The primary coolant inlet temperature is no greater than 125^{o} F, and
- The coolant flow rate through the core is at least 1500 gpm.

The transient analysis for forced convection cooling was performed for the most conservative case in which all of the safety channels are at their respective limiting trip values when the transient is terminated. The analysis shows that the peak fuel cladding temperature will be no greater than 87.9^{o} C during a transient in which the following combination of limits are in place:

- The initial power level is no greater than 2.2 MW,
- The coolant height above the uranium silicide fuel is at least 23 feet 6.5 inches,

- The primary coolant inlet temperature is no greater than 125° F,
- The coolant flow rate through the core is a least 1500 gpm, and
- The transient is terminated by an over power trip at 2.5 MW.

Under these conditions for the most conservative case, there is a margin of:

$$530^{o}\text{ C} - 87.9^{o}\text{ C} = 442.1^{o}\text{ C}.$$

Measurement uncertainty was based on the nominal operating values of 2 MW, 1950 gpm, and 90° F to 115° F for the power, flow and temperature respectively, and has been determined to be:

- Power Level ± 0.2 MW
- Coolant Height 0.5 inches
- Temperature 3° F
- Flow Rate ± 60 gpm

Consequently, the bases for these specifications are:

Specification 2.2.2.1 sets the limiting safety system setting for reactor thermal power to be 2.3 MW. The analyses show that cladding damage will not occur under any condition if power is no greater than 2.5 MW. Taking into consideration a 0.2 MW measurement error, if the LSSS is 2.3 MW, then the Limiting Trip Value could never be greater than 2.5 MW.

Specification 2.2.2.2 sets the limiting safety system setting for the height of coolant above the top of the uranium silicide fuel to be 23 feet 7 inches. The analyses show that cladding damage will not occur under any condition if the height is no less than 23 feet 6.5 inches. Taking into consideration a 0.5 inch measurement error, if the LSSS is 23 feet 7 inches, then the Limiting Trip Value could never be as less than 23 feet 6.5 inches.

Specification 2.2.2.3 sets the limiting safety system setting for the primary coolant inlet temperature to be 122° F. The analyses show that cladding damage will not occur under any condition if the primary coolant inlet temperature is no greater than 125° F. Taking into consideration a 3° F in measurement error, if the LSSS is 122° F, then the Limiting Trip Value could never be greater than 125° F.

Specification 2.2.2.4 sets the limiting safety system setting for the primary coolant flow rate to be 1560 gpm. The analyses show that cladding damage will not occur under any condition if the primary coolant flow rate is at least 1500 gpm. Taking into consideration a 60 gpm measurement error, if the LSSS is 1560 gpm, then the Limiting Trip Value could never be less than 1500 gpm.

3.0 Limiting Conditions for Operation

3.1 Core Parameters

3.1.1 Reactivity Limits

Applicability:

These specifications apply to all core configurations, including configurations that have experiments installed inside the reactor, the reactor pool, or inside the reactor experimental facilities.

Objective:

The objective of these specifications is to make certain that core reactivity parameters will not exceed the limits used in the safety analysis to ensure that a reactor transient will not result in damage to the fuel.

Specifications:

3.1.1.1 Core

3.1.1.1.1 The core shutdown margin shall be at least 1.0 %Δk/k.

3.1.1.1.2 The core excess reactivity shall not exceed 4.7 %Δk/k.

3.1.1.1.3 The reactor shall be subcritical by at least 3.0 %Δk/k during fuel loading changes.

3.1.1.2 Control Rods

3.1.1.2.1 The reactivity worth of the regulating rod shall not exceed 0.6 %Δk/k.

3.1.1.3 Experiments

3.1.1.3.1 The total absolute reactivity worth of experiments shall not exceed the following limits:

3.1.1.3.1.1	Total Moveable and Fixed	0.6 %Δk/k
3.1.1.3.1.2	Total Moveable	0.08 %Δk/k

3.1.1.3.2 The maximum reactivity worth of any individual experiment shall not exceed the following limits:

3.1.1.3.2.1	Fixed	0.6 %Δk/k
3.1.1.3.2.2	Moveable	0.08 %Δk/k

Bases:

Specification 3.1.1.1.1 provides a limit for the minimum shutdown reactivity margin that must be available for all core configurations. The shutdown margin is necessary to ensure that the reactor can be made subcritical from any operating condition, and to ensure that it will remain subcritical after cool down and xenon decay, even if the most reactive control rod failed in the fully withdrawn position. No credit is taken for the negative reactivity worth of the regulating rod because it would not be available as part of the negative reactivity insertion in the event of a scram.

Specification 3.1.1.1.2 provides a maximum limit for excess reactivity available for all core configurations. Excess reactivity is necessary to overcome the negative reactivity effects of coolant temperature increase, coolant void increase, fuel temperature increase, and xenon build-up that occur during sustained operations. Excess reactivity is also required to be available in order to overcome any negative reactivity effects of experiments that are installed in the core.

Specification 3.1.1.1.3 provides a limit for the minimum core shutdown reactivity during fuel loading changes. This limit takes advantage of the negative reactivity that can be added to the core above and beyond the shutdown margin by the insertion of the highest reactivity worth shim safety blade, and the regulating control rod. This limit assures that the core will remain subcritical during these operations, or in the event that a fuel element is misplaced in the core.

Specification 3.1.1.2.1 provides a limit for the reactivity worth of the regulating rod. The reactivity limit is set to a value less than the delayed neutron fraction so that a failure of the automatic servo system could not result in a prompt critical condition.

Specification 3.1.1.3.1 provides total reactivity limits for all experiments installed in the reactor, the reactor pool, or inside the reactor experimental facilities. The limit on total experiment worth is set to a value less than the delayed neutron fraction so that an experiment failure could not result in a prompt critical condition. The limit on total moveable experiment worth is set to a value that will not produce a stable period of less than 30 seconds, so that the reactivity insertion can be easily compensated for by the action of the control and safety systems. As part of the Safety Analysis, Argonne National Laboratory modeled a reactivity insertion of + 0.08 %Δk/k over a 0.1 second interval, and determined that this reactivity insertion resulted in a stable period of approximately 75 seconds. This specification limits the reactivity worth of experiments to values of reactivity which, if introduced as positive step changes, would preclude violating any Safety Limit. The transient analysis demonstrates that this Limiting Condition for Operation on reactivity for experiments results in no challenge to fuel integrity under credible postulated transients.

Specification 3.1.1.3.2 provides total reactivity limits for any individual experiment installed in the reactor, the reactor pool, or inside the reactor experimental facilities. The reactivity limits for both, individual fixed and moveable experiments are the same as the limits for total fixed and moveable experiments. Consequently, the safety analysis done for Specification 3.1.1.3.1 applies to this specification as well.

3.1.2 Core Configuration Limits

Applicability:

These specifications apply to core configurations during all modes of operation.

Objective:

The objective of these specifications is to ensure that there is sufficient coolant flow to remove heat from the fuel elements when the reactor is in operation.

Specifications:

3.1.2.1 All core grid positions shall contain fuel elements, baskets, reflector elements, or experimental facilities during reactor operations.

3.1.2.2 The pool dam shall be in its storage location during reactor operations.

Bases:

Specification 3.1.2.1 requires that all of the core grid spaces be filled when the reactor is operating. This requirement prevents the degradation of coolant flow through the fuel channels due to flow bypassing the actively fueled region of the core through unoccupied grid plate positions.

Specification 3.1.2.2 requires that the pool dam that is used for separating the sections of the pool, be in its storage location when the reactor is in operation. This requirement ensures that there will be a sufficient heat sink for reactor operations at any power level, and ensures that the full volume of the pool water will be available in the event of a loss of coolant accident.

3.2 Reactor Control and Safety System

Applicability:

These specifications apply to the reactor safety system and instrumentation required for reactor operation.

Objective:

The objective of these specifications is to define the minimum set of safety system and instrumentation channels that must be operable in order for reactor operation.

Specifications:

The reactor shall not be operated unless:

3.2.1 All four shim safety blades and the regulating rod are operable.

3.2.2 All four shim safety blades are capable of being fully inserted into the reactor core within 1 second from the time that a scram condition is initiated.

3.2.3 The total reactivity insertion rate of any one shim safety blade and the regulating rod simultaneously does not exceed 0.02%$\Delta k/k$ per second.

3.2.4 The instrumentation shown in Table 3.1, Required Safety Channels, is operable and capable of performing its intended function:

Table 3.1 Required Safety Channels

Table 3.1.1 Required Safety Channel Scrams					
Line #	**Protection**	**Op Mode**	**Channels Required**	**Function**	**Set Point**
1.	Over Power	Both	2	Scram before power is greater than	115% Power for selected mode
2.	Rate of Change of Power	Both	1	Scram before period is less than	4 seconds
3.	Detector HV Failure for Lines 1 & 2 above	Both	1 per operable channel	Scram on a loss of HV power	50 V below suggested operating voltage
4.	Low Pool Level	Both	1	Scram before pool level is less than	23 feet 7 inches above the top of the fuel
5.	Manual Scram	Both	1	Scram when	Control Room Scram Button Depressed
6.	Control Rod Drive Communication (Watchdog)	Both	1	Scram if loss of communication for greater than	10 seconds
7.	Seismic Disturbance	Both	1	Scram when	Seismic Disturbance Detected
8.	Bridge Movement	Both	1	Scram when	Bridge Movement Detected
9.	Pool Temperature	NC	1	Scram before temperature is greater than	127^{o} F
10.	Primary Coolant Inlet Temperature	FC	1	Scram before temperature is greater than	122^{o} F
11.	Primary Coolant Flow Rate	FC	1	Scram before flow rate is less than	1560 gpm
12.	Coolant Gates Open	FC	1	Scram when	Inlet or outlet gate open
13.	No Flow Thermal Column	FC	1	Scram when	No Flow Detected
14.	Bridge Low Power Position	FC	1	Scram when	Bridge Not Seated at HP End

Table 3.1.2 Required Safety Channel Interlocks					
Line #	**Protection**	**Op Mode**	**Channels Required**	**Function**	**Set Point**
1.1	Servo Control Interlock	Both	1	Regulating rod cannot be placed in automatic servo mode if	Regulating rod not full out
1.2	Servo Control Interlock	Both	1	Regulating rod cannot be placed in automatic servo mode if reactor period is less than	30 seconds
2.	Shim Safety Blade Withdrawal Interlock	Both	1	No shim safety blade withdrawal if start up channel count rate less than	3 counts per second
2.2	Shim Safety Blade Withdrawal Interlock	Both	1	No shim safety blade withdrawal if Neutron Flux Monitor Test / Select switch is	Not in the Off position
2.3	One Shim Safety Blade Withdrawal Interlock	Both	1	Only one SSB can be withdrawn at any one time	Select switch if in manual mode, binary logic must be satisfied if in auto mode

Table 3.1.3 Required Safety Channel Indications					
Line #	**Description**	**Op Mode**	**Channels Required**	**Function**	**Set Point**
1.	Wide Range Linear Power	Both	1	Provide indication of reactor power	N/A
2.1	Log Power	Both	1	Provide indication of reactor power	N/A
2.2	Log Power Start-up Counts	Both	1	Provide indication of start-up channel counts	N/A
2.3	Log Period	Both	1	Provide indication of rate of change in reactor power	N/A
3.	Pool Temperature	NC	1	Provide indication of bulk pool temperature	N/A
4.	Primary Coolant Inlet Temperature	FC	1	Provide indication of primary coolant inlet temperature	N/A
5.	Primary Coolant Outlet Temperature	FC	1	Provide indication of primary coolant outlet temperature	N/A
6.	Primary Coolant Flow Rate	FC	1	Provide indication of primary coolant flow	N/A
7.	Confinement Building Pressure	Both	1	Provide indication of Confinement Building Pressure	N/A

Bases:

Specification 3.2.1 requires that all four shim safety blades and the regulating rod be operable. This ensures that all control rods are being controlled by the reactor control system and the licensed operator.

Specification 3.2.2 requires that all four shim safety blades are capable of being fully inserted into the reactor core within 1 second from the time that a scram condition is initiated. As part of the Safety Analysis, Argonne National Laboratory analyzed a variety of power transients in which it was assumed that the time between the initiation of a scram signal, and full insertion of all of the shim safety rods was one second. The analysis showed that if the reactor is operated within the safety limits, this time delay will not cause an over power excursion to damage the fuel.

Specification 3.2.3 requires that the reactivity insertion rates of individual shim safety and the regulating rod simultaneously do not exceed 0.02 %Δk/k per second. As part of the Safety Analysis, Argonne National Laboratory analyzed ramp insertions of 0.02 %Δk/k reactivity from a variety of initial power levels. The reactivity insertions are stopped by the over power trip. In all cases, peak fuel and cladding temperatures due to the power overshoot are well below the temperatures required to damage the fuel or cladding. Consequently, this limit ensures that an over power condition due to a reactivity insertion from raising a control rod will not damage the fuel or cladding.

Specification 3.2.4 Table 3.1 Instrumentation Required for Reactor Operation identifies the instrumentation that is required to be operable when the reactor is operated.

Two independent power level channels are required for both natural and forced convection cooling modes of operation, each of which must be capable of scramming the reactor by 115% of licensed power. The basis section of Specification 2.2.1.1 shows that this ensures that the power level limiting safety system setting for natural convection cooling will not be exceeded under any analyzed condition. The basis section of Specification 2.2.2.1 shows that this ensures that the power level limiting safety system setting for forced convection cooling will not be exceeded under any analyzed condition. Having two independent power level channels ensures that at least one over power protection will be available in the event of an over power excursion.

One rate of change of power channel is required for both cooling modes of operation. The 4 second period limit serves as an auxiliary protection to assure that the reactor fuel would not be damaged in the event that there was a power transient. As part of the Safety Analysis, Argonne National Laboratory analyzed a power excursion under forced convection cooling operating conditions involving a period of less than 1 second, which was stopped by an over power scram when the true power reached the limiting safety system setting of 2.3 MW. The analysis showed that peak fuel temperatures stayed well below the temperature required to damage the fuel. A 4 second period limit provides an additional layer of protection against this type of transient.

One detector HV failure scram is required for each power channel, and period channel that is considered operable. These channels rely on detectors that require high voltage in order to be operable. These scrams assure that the reactor will not be operated when one of these detectors does not have proper high voltage.

One low pool level channel is required for both forced and natural convection cooling modes of operation. This channel ensures that the reactor will not be in operation if the pool level is below the levels that were used for the steady state and transient analyses. These analyses assumed a minimum pool height of 23 feet 6.5 inches above the top of the uranium silicide fuel. The low pool level channel LSSS is 23 feet 7 inches above the uranium silicide fuel. Taking into consideration a 0.5 inch measurement error, this LSSS ensures that the pool height above the uranium silicide fuel will not be less than the pool level height that was used for the analyses.

One manual scram button that is located in the control room is required to be operational during both modes of operation.

One rod control communication scram is required for both modes of operation. The control rod drive system has a communication link between the digital display in the control room, and the stepper motor controllers out at the pool top. There is a watchdog feature that verifies that this communication link is not broken. In the event that the link is broken, a scram will occur within ten seconds of the break. All of the scram signals are sent independently of this link. The transient analysis performed by Argonne National Laboratory shows that if the control rod drive communication were lost while the reactor were on a period, the over power, and period trips would prevent the power from reaching a level that could damage the fuel cladding.

One seismic disturbance scram is required for both modes of operation. In the event of a seismic disturbance, the shim safety blade magnets would be likely to drop the blades due to the vibration caused by the disturbance. However, this scram ensures that the blades will be dropped in the event of a disturbance.

One bridge movement scram is required for both modes of operation. This scram assures that the reactor will be shut down in the event that the bridge moves during operation.

One pool temperature channel is required for natural convection cooling mode of operation. This channel is capable of scramming the reactor when the temperature reaches 127^{o} F. The basis section of Specification 2.2.1.3 shows that this ensures that the pool temperature will not exceed the 130^{o} F temperature that was used in the safety analysis. This channel provides the over temperature protection when the reactor is operated in the natural convection cooling mode.

One primary inlet temperature channel is required for forced convection cooling mode operation. This channel is capable of scramming the reactor when the temperature reaches 122^{o} F. The basis section of Specification 2.2.2.3 shows that this LSSS will ensure that the coolant inlet temperature will not exceed the 125^{o} F temperature that was used in the thermo-hydraulic analysis to show that fuel cladding could not be damaged under conditions within the bounds of the analyzed safety envelope.

One primary coolant flow rate channel is required for forced convection cooling mode operation. This channel assures that the reactor will not be operated at power levels above 100 kW with a primary coolant flow rate that is less than the 1500 gpm that was used in the thermo-hydraulic analysis to show that fuel cladding could not be damaged under conditions within the bounds of the analyzed safety envelope. The basis section of Specification 2.2.2.4 shows that if this channel is set to scram at a limiting safety system setting of 1560 gpm, the safety limit will not be exceeded under conditions within the bounds of the analyzed safety envelope.

One coolant gate open scram input on each coolant duct, either of which causes a reactor scram if open during forced convection cooling mode operation, is required. This scram ensures that coolant flow through the inlet and outlet ducts are not bypassed during forced convection cooling.

One no flow thermal column scram is required during forced convection cooling mode operation. This scram ensures that there is coolant flow through the thermal column gamma shield during operations above 100 kW.

One bridge low power position scram is required for forced convection cooling mode operation. In order for the forced convection cooling system to work, the reactor must be seated against the high power section pool wall. This scram ensures that the reactor is properly positioned in the pool so that the coolant ducts are properly coupled with the cooling system piping.

One servo control interlock that prevents the regulating rod from being put into automatic servo mode unless the rod is fully withdrawn, is required for both modes of operation. As a result of this interlock, when the regulating rod is transferred to automatic servo control, the blade is unable to insert additional reactivity into the core.

One servo control interlock that prevents the regulating rod from being put into automatic mode if the period is less than 30 seconds, is required for both modes of operation. This interlock limits the power overshoot that occurs when the regulating blade is put into automatic mode.

One shim safety interlock that prevents shim safety blade withdrawal if the startup neutron count rate is less than 3 cps is required for both modes of operation. This interlock ensures that the startup channel, which is the most sensitive indication of subcritical multiplication, is operational during reactor start-ups.

One shim safety interlock that prevents shim safety blade withdrawal if the neutron flux monitor test / select switch is not in the off position is required for both modes of operation. This interlock prevents shim safety withdrawal when this instrument is receiving test signals rather than actual signals from the detector that is part of the neutron flux monitor channel.

One Shim Safety interlock preventing more than one shim safety blade from being withdrawn at a time. This is accomplished using a selector switch when in the manual mode of rod control or a binary logic when in the automatic mode of rod control. This limits the amount of reactivity that can be added to the core during rod withdrawal to the reactivity insertion rates discussed in Specification 3.2.3.

Minimum safety channel indications provide the licensed operator with the necessary information to safely operate the reactor and respond to any abnormal conditions.

3.3 Coolant System

3.3.1 Primary Coolant System

3.3.1.1 Primary Coolant Conductivity

Applicability:

This specification applies to the primary coolant system.

Objective:

The objective of this specification is to maintain the primary coolant in a condition that minimizes corrosion of the fuel cladding, core structural materials, and primary coolant system components, as well as to minimize activation products produced as a result of impurities in the coolant.

Specification:

The reactor shall not be operated unless primary coolant conductivity is ≤ 2 μmhos / centimeter.

Basis:

Specification 3.3.1.1 is based on empirical data from the facility history. Over the lifetime of the facility, primary coolant conductivity has been maintained within the limit specified, and no corrosion on the fuel cladding, core structural materials, or primary coolant system components has been noted.

3.3.1.2 Primary Coolant Activity

Applicability:

This specification applies to the primary coolant.

Objective:

The objective of this specification is to provide a mechanism for detecting a potential fuel cladding leak.

Specification:

The reactor shall not be operated unless Cesium - 137 and Iodine - 131 activity in the primary coolant is indistinguishable from background. An exception can be made if the reactor operation is solely for the purpose of identifying which fuel assembly is damaged.

Basis:

Specification 3.3.1.2 provides a mechanism for detecting a potential fuel cladding leak by requiring that periodic primary coolant analysis be performed to test for the presence of Cesium - 137 or Iodine - 131. These isotopes are prominent fission products. Consequently, if either of these isotopes is detected in the primary coolant, it may be indicative of a fuel cladding defect.

3.3.2 Secondary Coolant System

Applicability:

This specification applies to the secondary coolant.

Objective:

The objective of this specification is to provide a mechanism for detecting a potential primary to secondary system leak.

Specification:

Sodium-24 activity in the secondary coolant shall be maintained at levels that are indistinguishable from background.

Basis:

Specification 3.3.2 provides a mechanism for detecting a potential primary to secondary system leak by requiring that periodic secondary coolant analysis be performed to test for the presence of Sodium-24. This isotope is produced by the activation of the aluminum structural materials in the primary pool, and a small concentration of it is present in the primary coolant during, and immediately following operation of the reactor. If this isotope is found in the secondary coolant, it may indicate a primary to secondary system leak.

3.4 Confinement System

Applicability:

These specifications apply to the confinement building and each of the components of the confinement system as follows:

Confinement Building Normal Personnel Access Door
Confinement Building Truck Bay Door
Confinement Building Roof Hatch
Confinement Building Control Room Emergency Door
Confinement Building Penetrations

Objective:

The objective of these specifications is to assure that confinement envelope is capable of fulfilling its intended function during reactor operations and in an accident scenario. The Confinement System in conjunction with the Confinement Ventilation System addressed in Section 3.5 will minimize the potential for a release of airborne radioactive material to the environment and ensure that any release will be within the limits of 10 CFR Part 20.

Specifications:

Whenever the following operations are in progress:

- The reactor is operating.
- Irradiated fuel handling is in progress.
- Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
- Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
- Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

3.4.1 The confinement system shall be operable.

Bases:

The purpose of the confinement system is to mitigate the consequences of airborne radioactive material release. The Confinement System in conjunction with the Confinement Ventilation System discussed in Section 3.5 maintains a differential pressure of -0.5” WC by keeping all of the doors and the roof hatch closed, except for entry and exit. This ensures that confinement airflow is directed through a defined pathway that is monitored for radiological release.

During operation of the reactor, the production of radioactive gasses or airborne particulates is possible. Though unlikely to occur, fuel cladding failure represents the greatest possible source of airborne radioactivity. The potential causes of fuel cladding damage or failure are:

- Damage during fuel handling operations.
- Fuel cladding damage due to an unanticipated reactivity excursion.

Additionally, fission products could be released due to damage to a sufficiently fueled experiment that has been irradiated long enough to build up a significant fission fragment inventory. In the event that the experiment is not adequately contained, it is conceivable that it could be damaged during handling operations to the extent that there could be fission fragment release.

These specifications ensure that the confinement system components will be operable during conditions for which there is any potential for fuel cladding damage or failure to occur, as well as for experiment failures in which fission products could potentially be released.

3.5 Confinement Ventilation System

Applicability:

These specifications apply to the Confinement Ventilation System including all components listed below and any interconnecting duct work that allows the system to perform its intended function:

Confinement Exhaust Blower
Off-gas Blower
Rabbit Blower
Dilution Blower
Emergency Exhaust Blower
Confinement Ventilation Intake Damper
Confinement Ventilation Exhaust Damper
Emergency Exhaust Air Filter Bank
Confinement Exhaust Stack
Facility Evacuation System

Objective:

The objective of this specification is to assure that the Confinement Ventilation System is capable of performing its intended function.

Specification:

Whenever the following operations are in progress:

- The reactor is operating.
- Irradiated fuel handling is in progress.
- Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
- Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
- Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

3.5.1 The Confinement Ventilation System shall be operable and maintaining a minimum differential pressure of -0.5” WC across the Confinement System boundary.

Basis:

The Confinement Ventilation System maintains a minimum differential pressure of -0.5” WC across the Confinement System discussed in Section 3.4 to ensure that all confinement air exhaust flow is through a controlled pathway that is monitored for radiological release.

Under emergency conditions, when a facility evacuation is initiated, the Confinement Ventilation

System realigns to isolate the Confinement Building and while maintaining the differential pressure of -0.5” WC directs all confinement air through an Emergency Filter Bank containing charcoal filters designed to remove any radioactive iodine that would be expected to be released during a fuel failure. An airflow limit of 1500 cfm through the filter ensures that the flow rate is low enough to allow the charcoal filter to adsorb at least 99% of the iodine that would be expected to be released in the event of a fuel cladding failure. The Emergency Filter Bank also contains absolute filters which prevent charcoal particulates from the charcoal filter from being released to the building exhaust air stream. The Dilution Blower remains running and provides a non-contaminated source of air to mix with the confinement air, so that any airborne radioactivity that is released is diluted prior to release.

3.6 Emergency Power System

Applicability:

This specification applies to the Emergency Electrical Power System that is required in order to ensure that power is available to the Confinement Ventilation System components to make certain that those systems are capable of performing their intended function in the event of an electrical power outage. The Emergency Electrical Power System consists of:

Emergency Generator
Emergency Power distribution components

Objective:

The objective of this specification is to assure that the Emergency Power System is able to perform its intended function when normal electrical power is unavailable.

Specification:

Whenever the following operations are in progress:

- The reactor is operating.
- Irradiated fuel handling is in progress.
- Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
- Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
- Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

3.6.1 The Emergency Power System shall be operable.

Basis:

Operability of the emergency electrical power system ensures that in the event of a facility electrical power outage, power will be available to those components in the Confinement and Confinement Ventilation Systems to allow them to be capable of performing their intended functions.

In the event of a power outage, the reactor will scram due to the loss of magnet current to the shim safety blades. The confinement exhaust blower will shut off due to loss of power. As long as the emergency and dilution blowers continue to be operable, the emergency confinement system will continue to perform its intended function. In the event of a power outage, the emergency power system will supply the emergency and dilution blowers with power so that they will be capable of operating, if needed, and the Confinement Ventilation systems will continue to be functional.

3.7 Radiation Monitoring System and Effluents

3.7.1 Radiation Monitoring Systems

3.7.1.1 Required Radiation Monitoring Systems

Applicability:

These specifications apply to the radiation monitoring systems required for critical operation of the reactor, and fuel handling activities. Refer to Table 3.2, Required Radiation Monitors.

Objective:

The objective of these specifications is to define the minimum set of radiation monitoring systems that must be operable for the reactor to be made critical, or for fuel handling activities.

Specifications:

Whenever the following operations are in progress:

- The reactor is operating.
- Irradiated fuel handling is in progress.
- Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
- Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
- Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

3.7.1.1.1 A minimum of one radiation monitor that is capable of warning personnel of high radiation levels in the confinement gaseous and particulate effluent (Table 3.2, Required Radiation Monitors, lines 1.1 and 1.2) shall be operating.

3.7.1.1.2 If the detector described in specification 3.7.1.1.1 fails during operation, within one hour, place in service a suitable alternative air monitor or begin an hourly grab sample analysis (grab sample analysis applies to particulate only) in lieu of having a functioning monitor.

3.7.1.1.3 A minimum of one gamma sensitive radiation monitor that is capable of warning personnel of high radiation levels shall be on the main floor of the Confinement Building and over the pool.

3.7.1.1.4 If the detector described in specification 3.7.1.1.3 fails, within one hour, place a suitable gamma sensitive alternative meter with alarming capability meeting all of the requirements as the detector originally used to satisfy 3.7.1.1.3 in service.

Bases:

A continuing evaluation of the air within confinement will be made in order to ensure that the airborne radioactivity concentration does not exceed 10 CFR Part 20 limits for personnel working inside confinement, and that the concentration exhausted from confinement does not exceed the limits for the general public.

Specification 3.7.1.1.1 identifies the air radiation monitoring instrumentation that is required to be operable when the reactor is operated, during fuel handling operations and experimentation.

Specification 3.7.1.1.2 allows for the air monitoring instrumentation to be temporarily replaced to allow operations to continue, or for grab samples (particulate only) to be performed in the event that the normal instrument fails.

Continuous monitoring for fission product release is performed at the pool top. In the event of a release, it is anticipated that the first indication would come from the pool top radiation detector which would detect the noble gasses, particularly Krypton and Xenon.

Specification 3.7.1.1.3 identifies the radiation monitoring instrumentation that is required to be operable when the reactor is operated, during fuel handling operations or experimentation.

Specification 3.7.1.1.4 allows for the fission product monitoring instrumentation to be replaced in the event that the normal instrument fails.

3.7.1.2 Radiation Monitoring System Alarm Set Points

Applicability:

These specifications apply to the radiation monitoring systems required for critical operation of the reactor, and fuel handling activities.

Objective:

The objective of these specifications is to ensure that personnel are notified in the event of unusually high radiation levels.

Specifications:

Radiation monitor alarm set points shall be established as follows:

3.7.1.2.1 The stack gaseous monitor shall alarm when radiation levels of the stack gas are 2.5 times normal levels, or greater.

3.7.1.2.2 The stack particulate monitor shall alarm when radiation levels of the stack particulates are 2 times normal levels, or greater.

3.7.1.2.3 The area radiation monitors shall alarm when radiation levels are 2 times normal levels, or greater.

Bases:

All of the radiation monitors in the confinement room have set points that are in terms of “normal” radiation levels. The purpose of defining set points in terms of “normal” radiation levels is to account for the fact that the radiation levels vary in the confinement room, depending on what kinds of experiments are being performed. If “normal” radiation levels are close to or at set point, radiation alarms would be ineffective and new set points could be established to accommodate the higher normal radiation levels for the duration of the experiment causing the higher “normal” levels.

Table 3.2 Required Radiation Monitors

3.2.1 Required Radiation Monitors					
Line #	**Description**	**Maximum Set Point**	**Minimum Required**	**Function**	**Operating Mode**
1.1	Confinement Building Exhaust Stack Gaseous	2.5 times normal	1	Indication and alarm both locally and in control room	As per 3.7.1.1.1
1.2	Confinement Building Exhaust Stack Particulate	2 times normal	1	Indication and alarm both locally and in control room	As per 3.7.1.1.1
2.	Reactor Bridge Area Monitor	2 times normal	1	Indication and alarm both locally and in control room	As per 3.7.1.1.3
3.	Main Floor of Confinement Building (At least one of 3.2.2, lines 3, 6 or 7)	2 times normal	1	Indication and alarm both locally and in control room	As per 3.7.1.1.3

3.2.2 Other Available Radiation Monitors (NO MINIMUM REQUIRED)					
Line #	**Description**	**Maximum Set Point**	**Detector Type**	**Function**	**Operating Mode**
1.	Main Floor Particulate Monitor	2 times normal	Alpha Beta Gamma	Indication and alarm both locally and in control room	N/A, Can be used as temporary alternate for stack particulate monitor
2.	Fuel Safe Area Monitor	2 times normal	Gamma Neutron	Indication and alarm both locally and in control room	N/A
3.	Thermal Column Area Monitor	2 times normal	Gamma Neutron	Indication and alarm both locally and in control room	N/A
4.	Heat Exchanger Area Monitor	2 times normal	Gamma Neutron	Indication and alarm both locally and in control room	N/A
5.	Primary Clean-Up Demineralizer Area Monitor	2 times normal	Gamma Neutron	Indication and alarm both locally and in control room	N/A
6.	Beam Port Area Monitors (4 total)	2 times normal	Gamma Neutron	Indication and alarm both locally and in control room	N/A
7.	Dry Irradiation Facility Area Monitor	2 times normal	Gamma Neutron	Indication and alarm both locally and in control room	N/A
8.	Rabbit room Area Monitor	2 times normal	Gamma Neutron	Indication and alarm both locally and in control room	N/A
9.	Rabbit Room Noble Gas Monitor	2 times normal	Noble Gas	Indication and alarm both locally and in control room	N/A
10.	Pool Level Noble Gas Monitor	2 times normal	Noble Gas	Indication and alarm both locally and in control room	N/A

3.7.2 Effluents

3.7.2.1 Airborne Effluents

Applicability:

This specification applies to the monitoring of airborne effluents from the Rhode Island Nuclear Science Center (RINSC).

Objective:

The objective of this specification is to assure that the release of airborne radioactive material from the RINSC will not cause the public to receive doses that are greater than the limits established in 10 CFR Part 20.

Specification:

The annual total effective dose equivalent to the individual member of the public likely to receive the highest dose from air effluents will be calculated using a generally-accepted computer program and will not exceed 100 mrem per year.

Basis:

10 CFR Part 20, Appendix B limits air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 100 mrem per year from these emissions.

Since the Rhode Island Nuclear Science Center is located on Narragansett Bay, the wind does not blow in the same direction more than about 10% of the time as shown in the following table taken from historical wind rose data.

Table3.3 Historical Wind Rose Data

Wind Blowing From	Frequency	%	Wind Blowing From	Frequency	%
North	6.20 E-02	6.02	South	5.80 E-02	5.80
North/Northeast	5.80 E-02	5.80	South/Southwest	8.40 E-02	8.40
Northeast	4.40 E-02	4.40	Southwest	1.05 E-01	10.50
East/Northeast	1.30 E-02	1.30	West/Southwest	6.40 E-02	6.40
East	1.20 E-02	1.20	West	6.80 E-02	6.80
East/Southeast	1.30 E-02	1.30	West/Northwest	9.50 E-02	9.50
Southeast	5.80 E-02	6.80	Northwest	1.04 E-01	10.40
South/Southeast	4.90 E-02	4.90	North/Northwest	6.80 E-02	6.80

Thus, during routine operations, no individual would be in the pathway of the plume more than about 10% of the time. Calculations of annual dose equivalent due to the primary airborne effluent, Argon-41, using the COMPLY Code shows less than the allowable limitation given in 10 CFR Part 20, Appendix B for the hypothetical maximum exposed individual member of the general public.

3.7.2.2 Liquid Effluents

Applicability:

This specification applies to liquid effluent discharges.

Objective:

The objective of this specification is to assure that liquid discharges are within regulatory limits.

Specification:

All liquid effluent discharges shall be within regulatory limits in accordance with 10 CFR 20, appendix B, table 3.

Basis:

Liquid effluent discharges are made on a periodic basis. This specification ensures that these discharges are within regulatory release limits outlined in 10 CFR 20, appendix B, Table 3.

3.8 Experiments

3.8.1 Experiment Materials

Applicability:

These specifications describe the limitations on the types of materials that may be irradiated or installed inside the reactor, the reactor pool, or inside the reactor experimental facilities.

Objective:

The objective of these specifications is to prevent damage to the reactor, reactor pool, and reactor experimental facilities.

Specifications:

3.8.1.1 Corrosives Materials

Corrosive materials shall be doubly contained in corrosion resistant containers. If failed container is suspected, all fuel assemblies and reactor structural components should be inspected.

3.8.1.2 Highly Water Reactive Materials

Highly water reactive materials shall not be placed inside the reactor, the reactor pool, or inside any reactor experimental facility where exposure to water is possible.

3.8.1.3 Explosive Materials

Explosive materials shall not be placed inside the reactor, the reactor pool, or inside the reactor experimental facilities.

3.8.1.4 Fissionable Materials

3.8.1.4.1 The quantity of fissionable materials used in experiments shall not cause the experiment reactivity worth limits to be exceeded.

3.8.1.4.2 The maximum quantity of fissionable materials used in an experiment shall be no greater than 87.5 milligrams of U-235 equivalent, and the maximum fission rate in a fissionable experiment shall be no greater than 2.1×10^{12} fissions per second.

3.8.1.4.3 Fissionable materials shall be doubly encapsulated.

3.8.1.4.4 Containers for experiments that have fissionable material shall be opened inside confinement.

Bases:

ANSI 15.1 recommends that the kinds of materials used in experiments be taken into consideration in order to limit the possibility of damage to the reactor, reactor pool, or reactor experimental facilities. Specifically, ANSI suggests that:

Damage could arise as a result of corrosive materials reacting with core, or experimental facility materials. Specification 3.8.1.1 reduces the possibility of this by requiring that corrosive materials be doubly contained so that the likelihood of container breach is minimized and if a breach is suspected requires all fuel assemblies and structural core components to be inspected for damage.

Damage could arise as a result of highly water reactive materials reacting with the pool water. Specification 3.8.1.2 makes this scenario impossible by prohibiting the use of highly water reactive materials in experiments.

Damage could arise as a result of explosive materials reacting inside an experimental facility. Specification 3.8.1.3 makes this scenario impossible by prohibiting the use of explosive materials in experiments.

Failure of experiments that contain fissionable materials have the potential to have an impact on reactor criticality, or on radioactive material release. Specification 3.8.1.4.1 ensures that the experiment will not cause a criticality accident that is not bounded by the reactivity limits that have been analyzed.

Specification 3.8.1.4.2 limits the quantity of fissionable material so that the quantity of radioactive material release due to an experiment failure will be within the bounds that were analyzed in the fuel failure analysis. The fissionable experiment malfunction analysis shows that if 96.25 mg of fissionable material is irradiated to saturation levels of iodine and xenon, and the failure occurs without the advantage of taking place under water, 10 CFR Part 20 dose limits will not be exceeded, given the occupancy assumptions that were used in the fuel failure analysis.

Specification 3.8.1.4.3 further reduces the probability of a radioactive material release from a fissionable experiment by requiring that these experiments be double encapsulated.

Specification 3.8.1.4.4 requires that when fissionable experiments are removed from encapsulation, these operations are performed inside confinement so that in the event of a radioactive material release, the advantages of the emergency ventilation system can be utilized.

3.8.2 Experiment Failures or Malfunctions

Applicability:

These specifications apply to experiments that are installed inside the reactor, the reactor pool, or inside the reactor experimental facilities.

Objective:

The objective of these specifications is to ensure that experiments cannot fail in such a way that they contribute to the failure of other experiments, core components, or principle barriers to the release of radioactive material.

Specifications:

3.8.2.1 Experiment shall be designed to ensure that credible failure of any experiment will not result in releases or exposures in excess of limits established in 10 CFR Part 20.

3.8.2.2 Experiment shall be designed to ensure that no reactor transient can cause the experiment to fail in such a way that it contributes to an accident.

3.8.2.3 Experiment shall be designed to ensure that credible failure of any experiment will not contribute to the failure of:

3.8.2.3.1 Other Experiments

3.8.2.3.2 Core Components

3.8.2.3.3 Principle physical barriers to uncontrolled release of radioactivity

Basis:

ANSI 15.1 recommends that experiment design be taken into consideration in order to limit the possibility that an experiment failure or malfunction could result in other failures, accidents, or significant releases of radioactive material.

Experiments are reviewed by the RINSC Nuclear and Radiation Safety Committee prior to being authorized to be installed in the reactor pool, or inside the reactor experimental facilities. These specifications ensure that experimental design is considered as part of the review, in order to minimize the possibility of these types of problems due to experiment failure or malfunction.

3.9 Reactor Core Components

3.9.1 Beryllium Reflectors

Applicability:

This specification applies to neutron flux damage to the standard and plug type beryllium reflectors.

Objective:

The objective of this specification is to prevent physical damage to the beryllium reflectors in the core from accumulated neutron flux exposure.

Specification:

The maximum accumulated neutron fluence shall be 1×10^{22} neutrons/cm^2.

Basis:

This limit is based on an analysis that was done by the University of Missouri Research Reactor (MURR). In their analysis, they note that the HFIR Reactor has noticed the presence of small cracks at fast fluences of 1.8×10^{22} nvt, and suggest that "a value of 1×10^{22} nvt (>1MeV) could be used as a conservative lower limit for determining when replacement of a beryllium reflector should be considered." The RINSC limit of 1×10^{22} nvt is even more conservative than what this analysis considers because it is not limited to fast neutron flux.

3.9.2 Low Enriched Uranium Fuel

Applicability:

This specification applies to the physical condition of the fuel elements used in the core.

Objective:

The objective of this specification is to prevent operation with damaged fuel elements.

Specification:

3.9.2.1 The reactor shall not be operated with known fuel defects unless it is to facilitate the determination of which fuel element is damaged.

Basis:

Fuel elements are initially inspected, and tested for core box fit upon receipt in accordance with written procedures in order to assure that the reactor is operated with fuel elements that are not damaged. Primary coolant is also tested to insure there is no detection of fission products in the coolant. Fuel elements with known defects will be removed from the core.

3.9.3 Experimental Facilities

3.9.3.1 Experimental Facility Configuration during Reactor Operation, Including a 4.5 hour period after shutdown.

Applicability:

These specifications apply to the reactor experimental beam ports and through ports during reactor operation, including a 4.5 hour period after shutdown.

Objective:

The objective of these specifications is to ensure that in the event of a Maximum Credible Accident, the rate at which the pool level would decrease would be low enough to make certain that the fuel cladding would not be damaged due to insufficient cooling.

Specifications:

Prior to reactor operation and for a period 4.5 hours after shutdown, the following experiment facility configurations will be established and maintained:

3.9.3.1.1 Each beam port shall have no more than a 1.25 inch diameter opening to confinement,

3.9.3.1.2 The drain valve from the through port shall be closed when the through port is in use.

3.9.3.1.3 When the through port is in use, gate valves shall be installed on the end(s) of the port that will be used for access.

3.9.3.1.4 When the through port is not physically manned and monitored, the ends of the through port shall be closed.

Bases:

Specification 3.9.3.1.1: The LOCA analysis shows that as long as the pool level does not drain through an area greater than 1.48 square inches (equivalent to a 1.37 inch diameter opening) to confinement, then there will be sufficient time for decay power to drop to a point which will not damage the fuel cladding, provided that the pool level does not drop below the elevation of the bottom of the eight inch beam ports. It also shows that if any single port has a catastrophic failure, the un-damaged ports do not become pool drain pathways. Consequently, limiting the opening on each experimental port that is open to confinement to 1.25 inch diameter is conservative.

Specification 3.9.3.1.2: Shearing of the through port is not considered to be a credible accident. Consequently, a leak in the through port is not anticipated to be catastrophic. The through port has three potential pool leak pathways: the drain/vent lines which join together and have a ½” orifice restriction and both ends, if open. By keeping the drain valve closed during through port use, this potential leak pathway is blocked, and any leakage would have to come out one of the ends. The potential for an unnoticed pool leak though this experimental facility is minimized.

Specification 3.9.3.1.3: The LOCA analysis has shown that the amount of time available for performing mitigating actions in the event of a non-catastrophic pool leak is on the order of hours. Consequently, as long as reactor/experimental personnel will become aware of a pool leak though the through port reasonably quickly, and the gate valves are in place, the consequence of the leak can be mitigated quickly by closing the valves.

Specification 3.9.3.1.4: This specification ensures that if the through port is not being monitored for the event of a pool leak, the ends are sealed so that the through port is not a LOCA pathway.

3.9.3.2 Experimental Facility Configuration Within the 4.5 Hour Period After Shutdown

Applicability:

These specifications apply to the reactor experimental facilities for the 4.5 hour period after reactor shutdown.

Objective:

The objective of these specifications is to ensure that in the event of a Maximum Credible Accident, the rate at which the pool level would decrease would be low enough to make certain that the fuel cladding would not be damaged due to insufficient cooling.

Specifications:

If the experimental facility configuration specified in 3.9.3.1 cannot be maintained for 4.5 hours after the reactor is shutdown, the following actions shall be taken prior to changing the configuration required by 3.9.3.1:

3.9.3.2.1 The reactor shall be moved to the low power section of the pool where it is at the opposite end of the pool from the beam port extensions.

3.9.3.2.2 The pool dam shall be positioned so that the high power section of the pool is isolated in such a way that if a beam port extension were sheared off, the pool level in the low power section would not be affected.

Bases:

Specification 3.9.3.2: In the event that access to a beam port is needed within 4.5 hours after shutdown, a provision is made so that the core can be isolated from the beam port end of the pool. With the core in the low power end of the pool, and the pool dam in place, if a beam port extension were sheared off, and a catastrophic beam port failure were to occur, the coolant level above the core would not be affected.

4.0 Surveillance Requirements

Applicability:

These specifications apply to the surveillance requirements of any system related to reactor safety.

Objective:

The objective is to verify the proper operation of any system related to reactor safety and to ensure compliance with the LCOs in accordance with section 3.0 of these Technical Specifications.

Specification

Surveillance requirements may be deferred during periods when the reactor is shutdown (except as noted in table 4.1, Technical Specification Surveillance Deferral Summary, below); however, they shall be completed prior to reactor start up unless reactor operation is required to perform the surveillance. Such surveillance shall be completed as soon as practical after reactor start up is complete.

Any additions, modifications, or maintenance performed on any of the systems or components addressed by these Technical Specifications shall be made and tested in accordance with the specifications the systems were originally designed and fabricated to or approved by the 10 CFR 50.59 review and approval process. A system shall not be considered operable until it is successfully tested.

Bases

This specification allows for the deferral of surveillances when the reactor is shut down provided they are performed prior to reactor operation or if operation is required to perform the surveillance, they are performed as soon as practical after reactor start up. This ensures that the requirements for limiting conditions of operation in accordance with section 3.0 are met.

Table 4.1 Technical Specification Surveillance Deferral Summary

	Technical Specification SR	Can Be Deferred During Shutdown (Y/N)	Required Prior to Reactor Operation (Y/N)
1	4.1.1.1 Core Reactivity Limit	Y	Y
2	4.1.1.2 Control Rod Reactivity Limit	Y	Y
3	4.1.1.3 Experiment Reactivity Limit	N	N
4	4.1.2 Core Configuration Limit	Y	Y
5	4.2.1 Shim safety drop times	Y	Y
6	4.2.2 Shim safety interlock/reactivity insertion rate	Y	Y
7	4.2.3 Reactor safety and safety related instrumentation	Y	Y
8	4.2.4 Reactor safety and safety related instrumentation for 2 MW mode of operation	Y	Y
9	4.2.5 Reactor safety and safety related instrumentation scrams, and interlocks	Y	Y
10	4.2.6 Reactor safety and safety related instrumentation channel calibration	Y	Y
11	4.3.1.1 Primary Coolant Conductivity	Y	N
12	4.3.1.2 Primary Coolant Activity	Y	N
13	4.3.1.3 Primary Coolant Level Inspection	N	Y
14	4.3.2.1 Secondary Coolant Activity	Y	N
15	4.4.1 Confinement System Operability	N	Y
16	4.4.2 Confinement System Operability	N	Y
17	4.4.3 Confinement System Operability	N	Y
18	4.5.1 Confinement Ventilation System Operability	N	Y
19	4.5.2 Confinement Ventilation System Operability	N	Y
20	4.5.3 Confinement Ventilation System Operability	N	Y
21	4.5.4 Emergency Filter Bank	N	Y
22	4.5.5 Emergency Filter Bank Flow	N	Y
23	4.6.1 Emergency Power System	N	Y
24	4.6.2. Emergency Power System	N	Y
25	4.6.3. Emergency Power System	N	Y
26	4.7.1.1 Radiation monitors	Y	Y
27	4.7.1.2 Radiation monitors	Y	Y
28	4.7.2.1 Airborne Effluents	N	N
29	4.7.2.2 Liquid Effluent Sampling	N	N
30	4.8.1 Experiments	Y	Y
31	4.9.1 Beryllium Reflector Elements	N	Y
32	4.9.2 Fuel Elements	N	Y
33	4.9.3.1 Experimental Facility Configuration	Y	Y
34	4.9.3.2 Accessing an Experimental Facility	N	N

4.1 Core Parameters

4.1.1 Reactivity Limit

Applicability:

These specifications apply to the surveillance requirements for reactivity limits.

Objective:

The objective of these specifications is to ensure that reactivity limits are not exceeded.

Specifications:

4.1.1.1 Core Reactivity Limit

4.1.1.1.1 The core shutdown margin shall be determined:

4.1.1.1.1.1 Annually

4.1.1.1.1.2 Whenever the core reflection is changed

4.1.1.1.1.3 Whenever the core fuel loading is changed

4.1.1.1.1.4 Following control blade changes.

4.1.1.1.2 The core excess reactivity shall be determined:

4.1.1.1.2.1 Annually

4.1.1.1.2.2 Whenever the core reflection is changed

4.1.1.1.2.3 Whenever the core fuel loading is changed

4.1.1.1.2.4 Following control blade changes.

4.1.1.1.3 The core shutdown reactivity shall be determined to remain greater than 3 %ΔK/K prior to and during fuel loading changes.

4.1.1.2 Control Rod Reactivity Limit

4.1.1.2.1 The reactivity worth of the shim safety blades and the regulating rod shall be determined:

4.1.1.2.1.1 Annually

4.1.1.2.1.2 Whenever the core reflection is changed

4.1.1.2.1.3 Whenever the core fuel loading is changed

4.1.1.2.1.4 Whenever maintenance is performed that could have an effect on the reactivity worth of the control rod

4.1.1.3 Experiment Reactivity Limit

4.1.1.3.1 The reactivity worth of new experiments shall be determined prior to the experiments initial use.

4.1.1.3.2 The reactivity worth of any on-going experiments shall be re-determined after the core configuration has been changed to a configuration for which the reactivity worth has not been determined previously.

Bases

Specification 4.1.1.1.1 requires that the core shutdown margin be determined annually, and whenever there is a change in core loading or core reflection. The annual measurement of the shutdown margin provides a snapshot of how the shutdown margin is increasing due to fuel burn-up. Measurements made whenever the core loading or reflection is changed provide assurance that core reactivity limits are not being exceeded due to changes in core configuration.

Specification 4.1.1.1.2 requires that the core excess reactivity be determined annually, and whenever there is a change in core loading or core reflection. The annual measurement of the excess reactivity provides a snapshot of how it is decreasing due to fuel burn-up. Measurements made whenever the core loading or reflection is changed provide assurance that core reactivity limits are not being exceeded due to changes in core configuration.

Specification 4.1.1.1.3 requires that the core shutdown reactivity shall be determined to remain greater than 3 %ΔK/K prior to and during fuel loading changes. This limit on shutdown reactivity while moving fuel insures a margin of safety while changing fuel configuration with no additional negative reactivity insertion available via control and safety systems and accounts for latest fuel burnup levels and new core loading.

Specification 4.1.1.2.1 requires that the shim safety blade and the regulating rod reactivity be determined annually, and whenever there is a change in core loading or core reflection. These determinations provide assurance that rod worth does not exceed reactivity limits, insertion limits, shutdown margin, and excess reactivity due to fuel burn-up, changes in core configuration, or control rod degradation.

Specification 4.1.1.3.1 requires that the reactivity worth of new experiments be determined prior to initial use. This ensures that reactivity worth limits are not exceeded.

Specification 4.1.1.3.2 requires that the reactivity worth of on-going experiments be re-determined after the core configuration has been changed to a configuration for which the reactivity worth has not been determined previously. This provides assurance that core configuration changes do not cause experiment reactivity worth limits to be exceeded, without requiring that experiment worth be re-determined every time that a recurring core configuration change, such as equilibrium core re-fueling, occurs.

4.1.2 Core Configuration Limit

Applicability:

These specifications apply to core configuration limit surveillances prior to reactor operations.

Objective:

The objective of these specifications is to ensure that core configuration supports reactor operation.

Specifications:

4.1.2.1 Prior to the first reactor start-up of the day, inspect the core to confirm that all grid positions contain fuel elements, baskets, reflector elements, or experimental facilities.

4.1.2.2 Prior to the first reactor start-up of the day, inspect to ensure that the pool dam is in its storage location.

Bases:

Specification 4.1.2.1 requires that all of the core grid spaces be filled when the reactor is operated. This inspection prior to each start-up ensures that the core configuration has not been changed and that all coolant flow will be through the core components as designed and not bypassed through an unoccupied grid location.

Specification 4.1.2.2 requires that the pool dam that is used for separating the sections of the pool be in its storage location when the reactor is in operation. This inspection ensures that the full volume of the pool water is available to support reactor operation.

4.2 Reactor Control and Safety System

Applicability:

These specifications apply to the safety and safety related instrumentation.

Objective:

The objective of these specifications is to ensure that the safety and safety related instrumentation is operable, and calibrated when in use.

Specifications:

4.2.1 Shim safety drop times shall be measured:

- 4.2.1.1 Annually
- 4.2.1.2 Whenever maintenance is performed which could affect the drop time of the blade
- 4.2.1.3 When a new core is configured
- 4.2.1.4 Following control blade changes

4.2.2 Shall measure each shim safety blade and regulating rod reactivity insertion rates:

- 4.2.2.1 Annually
- 4.2.2.2 Whenever maintenance is performed which could affect the reactivity insertion rate of the blade
- 4.2.2.3 When a new core is configured
- 4.2.2.4 Following control blade changes

4.2.3 The following reactor safety and safety related instrumentation shall be verified to be operable by performing a channel test prior to the initial start-up each day that the reactor is started up from the shutdown condition, and after the channel has been repaired:

- 4.2.3.1 Control room manual scram button
- 4.2.3.2 Power level channels
- 4.2.3.3 Period channel
- 4.2.3.4 Rod control communication watchdog scram

4.2.4 The following reactor safety and safety related instrumentation shall be verified to be operable by performing a channel test prior to the initial start-up each day that the reactor is started up from the shutdown condition, and for which reactor power level will be greater than 100 kW, and after the channel has been repaired:

4.2.4.1 All of the reactor safety and safety related instrumentation listed in 4.2.3.

4.2.4.2 Primary coolant flow scram

4.2.5 The following reactor safety and safety related instrumentation scrams, and interlocks shall be channel tested annually:

4.2.5.1 The following detector HV failure scrams:

4.2.5.1.1 Power level channels

4.2.5.1.2 Period channel

4.2.5.2 The following shim safety withdrawal interlocks:

4.2.5.2.1 Start-up count rate

4.2.5.2.2 Test / Select switch position

4.2.5.2.3 Shall verify that only one shim safety blade can be withdrawn at a time

4.2.5.3 The following servo control interlocks:

4.2.5.3.1 Regulating blade not full out

4.2.5.3.2 Period less than 30 seconds

4.2.5.4 The following coolant system channel temperature scrams:

4.2.5.4.1 Primary inlet temperature

4.2.5.4.2 Pool temperature

4.2.5.5 The following coolant system channel flow scrams:

4.2.5.5.1 Primary flow and flow rate

4.2.5.5.2 Coolant gates open

4.2.5.5.3 No flow thermal column

4.2.5.6 Low pool level scram

4.2.5.7 The following bridge scrams:

4.2.5.7.1 Bridge movement

4.2.5.7.2 Bridge low power position

4.2.5.8 Seismic scram

4.2.6 The following reactor safety and safety related instrumentation shall have a channel calibration performed annually:

4.2.6.1 Power level channels

4.2.6.2 Primary flow channel

4.2.6.3 Primary inlet and outlet temperature channels

4.2.6.4 Pool temperature channel

Bases:

Specification 4.2.1 defines the surveillance interval for measuring the shim safety drop times. The annual requirement is consistent with the historical facility frequency, and is within the range recommended by ANSI Standard 15.1. The requirement that this parameter be measured after maintenance is performed which could affect the drop time of the blade assures that the reactor will not be operated with a shim safety blade that does not meet the LCO requirements due to maintenance activities.

Specification 4.2.2 requires that all shim safety blade and regulating rod reactivity insertion rates shall be measured annually. The annual requirement is consistent with the historical facility frequency, and is within the range recommended by ANSI Standard 15.1.

Specification 4.2.3 indicates the reactor safety and safety related instrumentation that must be verified to be operable prior to the initial reactor start-up of each day. This requirement is consistent with the historical facility requirements.

Specification 4.2.4 provides for the fact that if the reactor is operated at power levels less than or equal to 100 kW, the forced cooling system is not required to be operational. However, for operations above 100 kW, this specification requires that the primary coolant flow scram be verified to be operable prior to the initial start-up of the reactor. This requirement is consistent with the historical facility requirements.

Specification 4.2.5 defines the surveillance interval for testing the reactor safety and safety related instrumentation scrams and interlocks that are not tested as part of the requirements of Specifications 4.2.3 and 4.2.4. For all of the scrams listed in these sections except specification 4.2.5.9 watchdog scram, the annual requirement is consistent with the historical facility frequency.

Specification 4.2.6 defines the surveillance interval for calibrating the safety and safety related instrumentation. The annual requirement is consistent with the historical facility frequency, and is within the range recommended by ANSI Standard 15.1.

4.3 Coolant Systems

4.3.1 Primary Coolant System

4.3.1.1 Primary Coolant Conductivity

Applicability:

This specification applies to the surveillance of the primary coolant.

Objective:

The objective of this specification is to provide a periodic verification that the primary coolant conductivity is within prescribed limits.

Specification:

The conductivity of the primary coolant shall be tested monthly.

Basis:

Specification 4.3.1.1 requires that the conductivity of the primary coolant be tested on a monthly basis. ANSI 15.1 recommends that this be performed on a weekly to quarterly schedule. Facility historical data supports this frequency as being sufficient to detect an abnormal trend.

4.3.1.2 Primary Coolant Activity

Applicability:

This specification applies to the surveillance of the primary coolant.

Objective:

The objective of this specification is to provide a periodic verification that the Cesium-137 and Iodine-131 activity in the primary coolant is not significantly above background.

Specification:

Cesium-137 and Iodine-131 activity in the primary coolant shall be measured monthly.

Basis:

Specification 4.3.1.2 requires that the Cesium-137 and Iodine-131 activity in the primary coolant be tested on a monthly basis. This schedule is within the schedule recommended by ANSI 15.1. These isotopes are indicators of a fuel failure.

4.3.1.3 Primary Coolant Level Inspection

Applicability:

This specification applies to the surveillance of the primary coolant.

Objective:

The objective of this specification is to ensure that the coolant level is at an adequate height above the core during reactor operation.

Specification:

The primary coolant level shall be verified to be greater than or equal to the Limiting Safety System Setting value prior to the initial start-up each day that the reactor is started up from the shutdown condition.

Basis:

Specification 4.3.1.3 requires that the primary coolant level be inspected prior to the first reactor start-up of each day. A float switch is used to monitor the pool level 24 hours per day, 7 days per week. This system is tied into the facility alarm system, which is monitored by an offsite alarm company. In the event that the pool reaches a level that is within one inch of the LSSS, the automatic pool fill is started. If the pool level drops to the LSSS, then a scram occurs, the operator receives an alarm, and the alarm company receives an alarm. A daily verification of the pool level prior to starting the reactor up provides adequate assurance that the float switch is working to maintain the pool level.

4.3.2 Secondary Coolant System

4.3.2.1 Secondary Coolant Activity

Applicability:

This specification applies to the surveillance of the secondary coolant.

Objective:

The objective of this specification is to provide a periodic verification that the Sodium-24 activity in the secondary coolant is not significantly above background.

Specification:

Sodium-24 activity in the secondary coolant shall be measured monthly.

Basis:

Specification 4.3.2.1 requires that the Sodium-24 activity in the secondary coolant be tested on a monthly basis. This schedule is within the schedule recommended by ANSI 15.1.

4.4 Confinement System

Applicability:

This specification describes the surveillance requirements for the Confinement System and components.

Objective:

The objective of this specification is to verify that the Confinement System working in conjunction with the Confinement Ventilation System, ref TS 4.5, is capable of maintaining a minimum of negative 0.5”WC differential pressure across the Confinement System boundary prior to being utilized to support operations activities.

Specification:

4.4.1 It shall be verified each day that the Confinement System is operable and working in conjunction with the Confinement Ventilation System, ref TS 4.5, maintaining a minimum of -0.5”WC differential pressure across the Confinement System boundary prior to any of the following conditions:

- 4.4.1.1 Reactor operations.
- 4.4.1.2 Handling of irradiated fuel.
- 4.4.1.3 Experiment handling for an experiment that has a significant fission product, or gaseous effluent activation product inventory.
- 4.4.1.4 Performing any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.
- 4.4.1.5 Performing any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.4.2 It shall be verified that the Confinement System remains operable during an initiation of a facility evacuation.

- 4.4.2.1 Monthly
- 4.4.2.2 Following any maintenance that could affect the operability of the system

4.4.3 It shall be verified that the Confinement System remains operable during an initiation of a facility evacuation alarm concurrent with a loss of normal AC power to the facility.

- 4.4.3.1 Quarterly
- 4.4.3.2 Following any maintenance that could affect the operability of the system.

Basis:

By ensuring that the confinement system is operable prior to each day of operations that could potentially create airborne activity, conditions are verified to be in place to make certain that working in conjunction with the Confinement Ventilation System, ref TS 4.5, any airborne radioactivity release within the Confinement System boundary would be directed to the stack, mixed with dilution air, and detected by the stack radiation monitoring system.

A periodic functional test of the Confinement System under emergency conditions ensures that the Confinement System is capable of supporting the Confinement Ventilation System in the event of an airborne radioactivity release. The testing periods that are specified conform to ANSI 15.1 recommendations.

4.5 Confinement Ventilation System

Applicability:

This specification describes the surveillance requirements for the Confinement Ventilation System.

Objective:

The objective of this specification is to verify that the Confinement Ventilation System is operable and working in conjunction with the Confinement System, ref TS 4.4, maintaining a minimum of -0.5”WC differential pressure across the Confinement System boundary prior to being utilized to support operations activities and remains operable upon initiation of the Emergency Ventilation Mode of operation.

Specification:

4.5.1 It shall be verified each day that the Confinement Ventilation System is operable and working in conjunction with the Confinement System, ref TS 4.4, maintaining a minimum of negative 0.5”WC differential pressure across the Confinement System boundary prior to any of the following conditions:

4.5.1.1 Reactor operations.

4.5.1.2 Handling of irradiated fuel.

4.5.1.3 Experiment handling for an experiment that has a significant fission product, or gaseous effluent activation product inventory.

4.5.1.4 Performing any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.5.1.5 Performing any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.5.2 It shall be verified that the Confinement Ventilation System Emergency Mode activates and maintains greater than a differential pressure of -0.5” WC during an initiation of a facility evacuation alarm.

4.5.2.1 Monthly

4.5.2.2 Following any maintenance that could affect the operability of the system

4.5.3 It shall be verified that the Confinement Ventilation System Emergency Mode activates and maintains greater than a differential pressure of -0.5” WC during an initiation of a facility evacuation alarm concurrent with a loss of normal AC power to the facility.

4.5.3.1 Quarterly

4.5.3.2 Following any maintenance that could affect the operability of the system.

4.5.4 The Emergency Filter Bank shall be verified to be at least 99% efficient for removing iodine:

4.5.4.1 Biennially

4.5.4.2 Following any maintenance that could affect the operability of the system.

4.5.5 The ventilation flow through the Emergency Filter Bank shall be verified to be less than or equal to 1500 SCFM:

4.5.5.1 Biennially

4.5.5.2 Following any maintenance that could affect the operability of the system.

Bases:

By ensuring that the Confinement Ventilation System is functional prior to each day of reactor start-up, conditions are verified to be in place to make certain that any airborne radioactivity release would be directed to the stack and be detected by the stack radiation monitor system and that the system is capable of supporting the Emergency Mode of operation.

A periodic test of the operability of the Confinement Ventilation System in the Emergency Mode of operation ensures that in the event of an airborne radioactivity release, the Confinement Ventilation System Emergency Mode will: 1) activate and realign as required, 2) maintain a flow rate through the filter bank less than or equal to 1500 SCFM and 3) remove at least 99% of the iodine from the exhaust air. The testing periods that are specified conform to ANSI 15.1 recommendations.

A periodic test of the operability of the Confinement Ventilation System in the Emergency Mode of operation concurrent with a loss of AC power ensures that the function of the Confinement Ventilation System in the Emergency Mode of operation will not be impacted by a loss of off-site power.

4.6 Emergency Power System

Applicability:

These specifications describe the surveillance requirements for the Emergency Power System.

Objective:

The objective of these specifications is to verify that the emergency power system is operable and will perform its intended function.

Specifications:

4.6.1 It shall be verified that the Emergency Power System is operable at least daily prior to any of the following conditions:

4.6.1.1 The reactor is operating.

4.6.1.2 Irradiated fuel handling is in progress.

4.6.1.3 Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory.

4.6.1.4 Any work on the core or control rods that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.6.1.5 Any experiment movement that could cause a reactivity change of more than 0.60 %Δk/k is in progress.

4.6.2. Perform an operability test to verify that the Emergency Power System starts and loads (see TS 4.5.3) in the event of a facility power outage.

4.6.2.1. Quarterly

4.6.2.2. Following emergency system load changes

4.6.3. It shall be verified that the available fuel for the emergency generator is at least 50% of full capacity.

4.6.3.1 Monthly

Bases:

Specification 4.6.1: By ensuring that the Emergency Power System is functional prior to each day of reactor start-up, conditions are verified to be in place to make certain that in the event of a loss of facility AC power while the Emergency Mode of operation is required to mitigate a potential release, emergency power would be available for the components of the Confinement and Confinement Ventilation Systems to perform their intended function.

Specification 4.6.2 periodically tests the emergency power system to ensure that in the event of a facility power outage, the emergency power system would automatically start, and be capable of handling the load required to power the emergency confinement systems. Initiation of the emergency mode of operation connects the emergency blower and the dilution blower to the emergency power supply. The testing periods that are specified conform to ANSI 15.1 recommendations.

Specification 4.6.3 ensures that there is sufficient fuel to power the emergency generator under full load for approximately 30 hours.

4.7 Radiation Monitoring System and Effluents

4.7.1. Required Radiation Monitoring Systems

Applicability:

These specifications apply to the radiation monitoring systems that are required to be operable during reactor operation and fuel handling activities.

Objective:

The objective of these specifications is to verify the operability of required radiation monitoring instrumentation.

Specifications:

4.7.1.1 The following radiation monitors shall be operable each day prior to the reactor being started up from the shutdown condition, and after the channel has been repaired:

- 4.7.1.1.1 At least one experimental level area radiation monitor
- 4.7.1.1.2 At least one pool top area radiation monitor
- 4.7.1.1.3 The gaseous effluent air monitor
- 4.7.1.1.4 The particulate air monitor

4.7.1.2 The following radiation monitors shall be channel calibrated and channel tested annually:

- 4.7.1.2.1 The experimental level area radiation monitor
- 4.7.1.2.2 The pool top area radiation monitor
- 4.7.1.2.3 The gaseous effluent air monitor
- 4.7.1.2.4 The particulate air monitor

Bases:

Specification 4.7.1.1 indicates the radiation monitors that must be verified to be operable each day prior to reactor start-up. This requirement is consistent with the historical facility requirements.

Specification 4.7.1.2 defines the surveillance interval for calibrating and testing the radiation monitors required to support operations activities. The annual requirement is consistent with the historical facility frequency, and is within the range recommended by ANSI Standard 15.1.

4.7.2. Effluents

4.7.2.1 Airborne Effluents

Applicability:

This specification applies to the monitoring of airborne effluents from the Rhode Island Nuclear Science Center (RINSC).

Objective:

The objective of this specification is to assure that the release of airborne radioactive material from the RINSC will not cause the public to receive doses that are greater than the limits established in 10 CFR Part 20.

Specification:

The annual total effective dose equivalent to the individual member of the public likely to receive the highest dose from air effluents shall be calculated annually.

Basis:

10 CFR Part 20 states the requirements on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 100 mrem per year from these emissions.

Since the Rhode Island Nuclear Science Center is located on Narragansett Bay, the wind does not blow in the same direction more than about 10% of the time as shown in Table 3.3 taken from historical wind rose data.

Thus, during routine operations, no individual would be in the pathway of the plume more than about 10% of the time. Calculations of annual dose equivalent due to the primary airborne effluent, Argon-41, using the COMPLY Code show less than the allowable limitation given in 10 CFR Part 20 for the hypothetical maximum exposed individual member of the general public.

4.7.2.2 Liquid Effluent Sampling

Applicability:

This specification applies to the monitoring of radioactive liquid effluents from the Rhode Island Nuclear Science Center.

Objective:

The objective of this specification is to assure that exposure to the public resulting from the release of liquid effluents will be within the regulatory limits and consistent with as low as reasonably achievable requirements.

Specification:

The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined to be less than the limits set in 10 CFR Part 20 before release.

Basis:

10 CFR Part 20.2003 permits discharges to the sanitary sewer provided that conditions in 10 CFR Part 20.2003 (a) are met.

4.8 Experiments

Applicability:

This specification applies to experiments that are installed inside the reactor, the reactor pool, or inside the reactor experimental facilities.

Objective:

The objective of this specification is to ensure that experiments have been reviewed to verify that the design is within the limitations of the RINSC Technical Specifications and 10 CFR Part 50.59.

Specification:

4.8.1 Experiments shall be reviewed to ensure that the design is within the limitations of the RINSC Technical Specifications and 10 CFR Part 50.59 prior to the experiments initial use.

Basis:

This specification ensures that all experiments will be reviewed to verify that the experiment designs are within the limitations of the RINSC Technical Specifications and 10 CFR Part 50.59 prior to its initial use.

4.9 Facility Specific Surveillance

4.9.1 Beryllium Reflector Elements

Applicability:

These specifications apply to the surveillance of the standard and plug type beryllium reflectors.

Objective:

The objective of these specifications is to prevent physical damage to the beryllium reflectors in the core from accumulated neutron flux exposure.

Specifications:

4.9.1.1. The maximum neutron fluence of any beryllium reflector shall be determined and verified to be less than $1x10^{22}$ neutrons/cm^2 annually.

4.9.1.2. The beryllium reflectors shall be visually inspected and functionally fit into the core grid box on a rotating basis not to exceed five years such that:

4.9.1.2.1. The annual surveillance shall include at least one fifth of the beryllium reflectors that are in the core,

4.9.1.2.2. If a beryllium reflector is removed from use and the time since its last surveillance exceeds five years, it shall be visually inspected and functionally fit into the core grid box prior to being placed in use, and

4.9.1.2.3. If damage is discovered, the damaged reflector shall be removed from service and the surveillance shall be expanded to include all of the beryllium reflectors prior to use, and annually thereafter.

Bases:

Historically, the total lifetime neutron fluence has increased by less than 1% of the maximum limit per year. Consequently, an annual verification of total fluence is reasonable. Additionally, reflector elements are visually inspected and functionally fit into the core grid box in order to verify that there are no observable defects or swelling. The rotating inspection schedule ensures that all of the reflectors in the core will be inspected at least once every five years. Since core element handling represents one of the highest risk opportunities for mechanically damaging the fuel cladding, this schedule is deemed appropriate, given the limited amount of information that is gained from these inspections. The discovery of a damaged reflector triggers an increase in the inspection schedule to an annual period.

4.9.2 Fuel Elements

Applicability:

These specifications apply to the surveillance of in service LEU fuel elements.

Objective:

The objective of these specifications is to verify the physical condition of the fuel elements in order to prevent operation with damaged fuel elements.

Specifications:

4.9.2.1 The fuel elements shall be visually inspected and functionally fit into the core grid box on a rotating basis not to exceed five years such that:

4.9.2.1.1 The annual surveillance shall include at least one fifth of the fuel elements that are in the core,

4.9.2.1.2 The annual surveillance shall include fuel elements that represent a cross section with respect to burn-up,

4.9.2.1.3 If a fuel element is removed from use and the time since its last surveillance exceeds five years, it shall be visually inspected and functionally fit into the core grid box prior to being placed in use, and

4.9.2.1.4 If damage is visually determined or detected by Technical Specification 4.3.1.2 or otherwise discovered, then the surveillance shall be expanded to include all of the fuel elements prior to use, and annually thereafter.

Bases:

Specification 4.9.2.1: TS 4.3.1.2 requires periodic pool water analysis to test for the presence of radioactivity that could potentially indicate a fuel cladding failure. Fuel elements are visually inspected and functionally fit into the core grid box in order to verify that there are no observable fuel defects or swelling. The rotating inspection schedule ensures that all of the fuel elements in the core will be inspected at least once every five years. Since fuel handling represents one of the highest risk opportunities for mechanically damaging the fuel cladding, this schedule is deemed appropriate, given the limited amount of information that is gained from these inspections. The pool water analysis is the most sensitive mechanism for detecting fuel cladding failure. A detected fuel failure triggers an increase in the inspection schedule to an annual period. Fuel inspections include a cross section of elements with respect to burn-up history in order to ensure that each inspection includes high burn-up elements that would be most likely to start to fail over time.

4.9.3 Experimental Facilities

4.9.3.1 Experimental Facility Configuration during Reactor Operation, including a 4.5 hour period after shutdown.

Applicability:

These specifications apply to the surveillance of reactor experimental facilities during reactor operation, including a 4.5 hour period after shutdown.

Objective:

The objective of these surveillances is to ensure that experimental facility configuration required to support reactor operation is set and maintained through operations and for an additional 4.5 hours after shutdown. This ensures that in the event of a Maximum Credible Accident, the rate at which the pool level would decrease would be low enough to make certain that the fuel cladding would not be damaged due to insufficient cooling.

Specifications:

Prior to operating the reactor the following conditions shall be verified, these conditions shall be maintained for a period of 4.5 hours after shutdown:

- 4.9.3.1.1 Each beam port shall have no more than a 1.25 inch diameter opening to confinement.
- 4.9.3.1.2 The drain valve from the through port shall be closed when the through port is in use.
- 4.9.3.1.3 When the through port is in use, gate valves shall be installed on the end(s) of the port that will be used for access.
- 4.9.3.1.4 When the through port is not physically manned and monitored, the ends of the through port shall be closed.

Bases:

Specification 4.9.3.1.1: The LOCA analysis shows that as long as the pool level does not drain through an area greater than 1.48 square inches to confinement, then there will be sufficient time for decay power to drop to a point which will not damage the fuel cladding, provided that the pool level does not drop below the elevation of the bottom of the eight inch beam ports. It also shows that if any single port has a catastrophic failure, the un-damaged ports do not become pool drain pathways. Consequently, limiting the areas of each experimental port that is open to confinement to 1.25 inch diameter is conservative.

Specification 4.9.3.1.2: Shearing the through port is not considered to be a credible accident. Consequently, a leak in the through port is not anticipated to be catastrophic. By keeping the drain valve closed during through port use, that potential leak pathway is blocked and the through ports ends will either be closed or continuously manned IAW 4.9.3.1.3 and 4.9.3.1.4.

Specification 4.9.3.1.3: The LOCA analysis has shown that the amount of time available for performing mitigating actions in the event of a non-catastrophic pool leak is on the order of hours. Consequently, as long as reactor/experimental personnel will become aware of a pool leak though the through port reasonably quickly, with the gate valves in place, the consequence of the leak can be mitigated quickly by closing the valves.

Specification 4.9.3.1.4: This specification ensures that if the through port is not being monitored for the event of a pool leak, the ends are sealed so that the through port is not a LOCA pathway.

4.9.3.2 Accessing an Experimental Facility Configuration Within the 4.5 Hour Period After Shutdown

Applicability:

These specifications apply to the reactor experimental facilities for the 4.5 hour period following a reactor shutdown.

Objective:

The objective of these specifications is to ensure that if it is absolutely necessary to remove a sample from an experimental facility addressed in these TSs, that the appropriate steps are taken to ensure that in the event of a Maximum Credible Accident, the rate at which the pool level would decrease would still be low enough to make certain that the fuel cladding would not be damaged due to insufficient cooling.

Specifications:

4.9.3.2.1 Prior to changing the configuration required by 4.9.3.1, shall verify that the reactor has not operated in the previous 4.5 hours.

4.9.3.2.2 If changing the configuration required by 4.9.3.1 within 4.5 hours after reactor shutdown is absolutely required, then it shall be verified that the follow actions have been completed:

4.9.3.2.2.1 The reactor is in the low power section of the pool, opposite the end of the pool where the beam port extensions are located.

4.9.3.2.2.2 The pool dam is positioned so that the high power section of the pool is isolated in such a way that if a beam port extension were sheared off, the pool level in the low power section would not be affected.

Bases:

Specification 4.9.3.2.1: The LOCA analysis shows that if the reactor were operated for an infinite amount of time at 2 MW, the amount of time that it would take for the power fraction to decay after shutdown to a point where the fuel cladding blister temperature could not be reached, even if the pool level were at the elevation of the bottom of the 8 inch beam ports, would be 4.5 hours. The analysis also shows that the maximum area of an opening between a beam port and confinement that limits this drain time to 4.5 hours is 1.48 square inches (equivalent to a 1.37 inch diameter opening). Consequently, maintaining the limit on the size of the opening between confinement and the beam ports to 1.25 inches in diameter for a period of 4.5 hours after shutdown ensures that in the event of a catastrophic beam port failure, the drain time would provide sufficient time for power to decay to a point below which the fuel could not be damaged.

Specification 4.9.3.2.2: In the event that access to a beam port is needed within 4.5 hours after shutdown, a provision is made so that the core can be isolated from the beam port end of the pool. With the core in the low power end of the pool, and the pool dam in place, if a beam port extension were sheared off, and a catastrophic beam port failure were to occur, the coolant level above the core would not be affected.

5.0 Design Features

Applicability

The following specifications apply to various design features of the facility and associated equipment which are not surveillable and therefore not addressed in Technical Specification sections 2.0, 3.0 or 4.0.

Objective

The objective of these specifications is to capture important design features that, in the event it became necessary to change, repair or replace the system and or components addressed in this section, design features addressed in the facility license and those important to safety would be maintained in accordance with original design and safety analysis parameters.

5.1 Site and Facility Specifications

5.1.1 The reactor facility is located in Narragansett, Rhode Island on a 3 acre section of the University of Rhode Island (URI), Narragansett Bay Campus (NBC).

5.1.2 The facility consists of a Confinement Building (also referred to as the reactor building), including the basement area and an office wing and lab building. The Confinement Building and Confinement Building basement serve as the restricted area.

5.2 Reactor Fuel

5.2.1 Each fuel element shall contain 22 plates containing uranium silicide fuel enriched to less than 20% in the isotope U-235 clad with aluminum.

5.2.2 Each fuel element shall contain no more than 283 grams of U-235.

5.3 Reactor Fuel Storage

5.3.1 All dry new fuel storage facilities shall have a configuration where k_{eff} is less than 0.8 under water flooded conditions.

5.3.2 A maximum of four fuel elements shall be stored in the fuel safe with no two elements in adjacent positions in the storage rack or in adjacent rows.

5.3.3 All irradiated fuel and experimental fissionable material not installed in the reactor core shall be stored in the reactor pool in storage racks in a configuration that ensures adequate cooling and is designed to maintain k_{eff} less than 0.9 under all conditions of moderation and reflection.

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5.4 Reactor Core

5.4.1 The reactor core box consists of a grid plate with a 9x7 array of 3 inch square modules designed to receive various components (ex. fuel elements, reflectors, experimental baskets, detectors) and four aluminum side walls. The four corner positions also serve as structural support posts.

5.4.2 The standard core consists of 14 fuel assemblies arranged symmetrically between the shim safety blades in the center of the core box. An alternate core with an additional 3 fuel assemblies installed at the thermal column end of the core is also approved for use.

5.4.3 All core designs shall insure that the temperature coefficient is negative.

5.5 Confinement (Reactor) Building

5.5.1 The free volume of the Confinement Building (volume of the building minus volume of the pool structure, including the water in the pool) shall be 181,955 cu ft.

5.6 Reactor Pool

5.6.1 The reactor pool is made of concrete with an aluminum liner.

5.7 Confinement Building Ventilation

5.7.1 The confinement building ventilation system emergency filtration train absolute filters shall be certified by the manufacturer to have an efficiency of not less than 99.97% when tested with 0.3 micron diameter dioctylphthalate smoke.

5.7.2 The confinement exhaust stack terminates at a minimum height equal to or greater than the confinement building.

Bases

The bases for these specifications is to ensure that design features considered in the facility license and those important to safety shall be maintained in accordance with original design and safety analysis parameters. These specifications shall be considered in all facility change reviews in accordance with the requirements of 10 CFR 50.59.

6.0 Administrative Controls

6.1 Organization

6.1.1 Organization Structure

The Rhode Island Nuclear Science Center (RINSC) Reactor shall be licensed to the State of Rhode Island. The Rhode Island Atomic Energy Commission is the state agency that shall have responsibility for the safe operation of the reactor. The Governor of the state shall appoint five Commissioners to the Rhode Island Atomic Energy Commission (RIAEC) who shall have the authority to recommend the selection of a Director, and appoint individuals to the Nuclear and Radiation Safety Committee (NRSC). The Director shall be the organizational head, and shall be responsible for the reactor facility license. The Assistant Director for Operations shall be responsible for the reactor programs and operation of the facility. The Assistant Director for Radiation and Reactor Safety shall be responsible for the safety programs of the facility. The RINSC staff shall operate and maintain the facility. The Nuclear and Radiation Safety Committee (NRSC) shall be an independent review and audit committee. Figure 6.1 shows the organization chart.

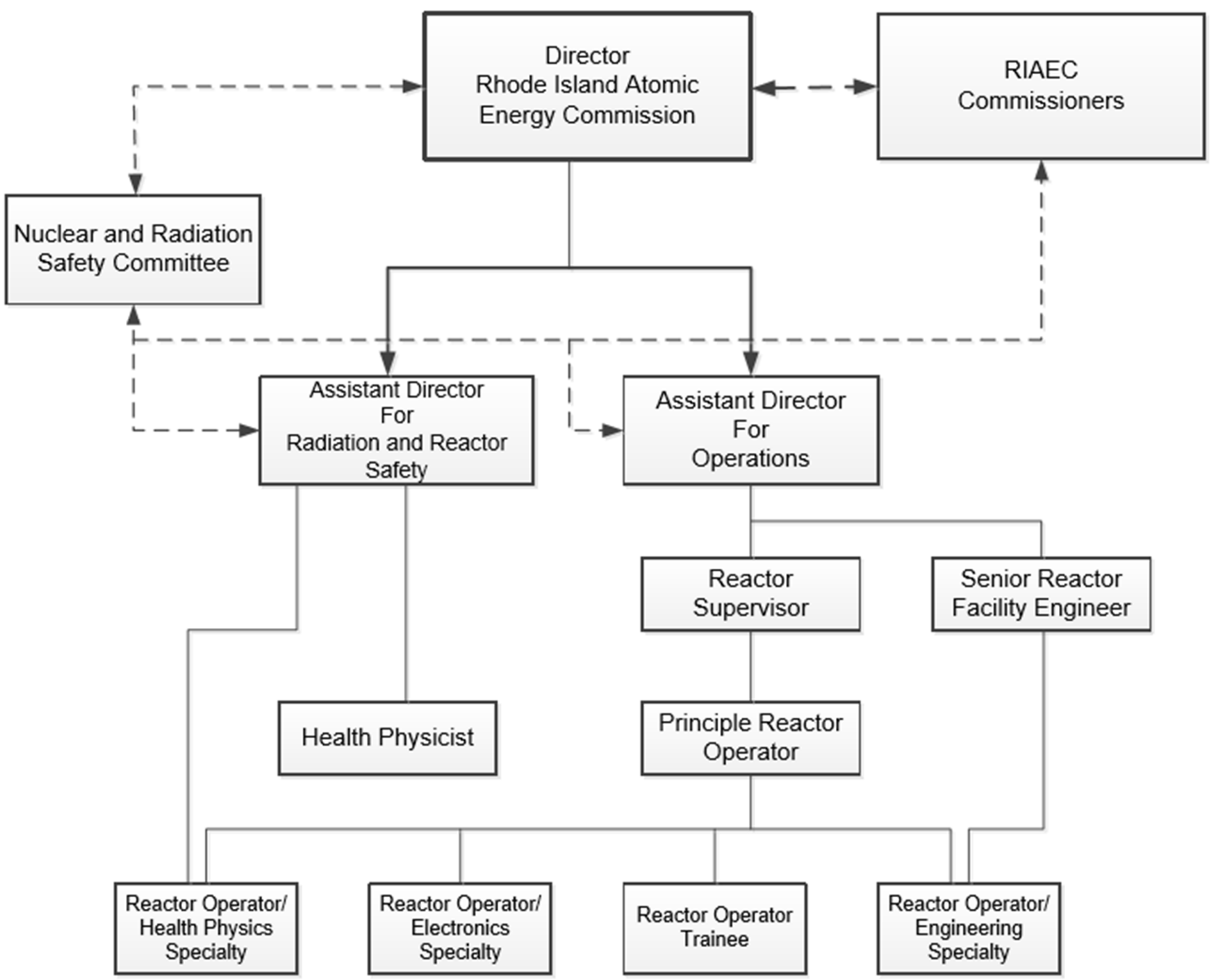

Figure 6-1 Rhode Island Atomic Energy Commission Organization Chart

6.1.2 Responsibility

6.1.2.1 Rhode Island Atomic Energy Commission (RIAEC)

The Rhode Island Atomic Energy Commission is the state agency that serves as the liaison between the State of Rhode Island, and the federal regulating authority. RIAEC, led by the Director, shall have the ultimate responsibility for the RINSC Reactor license. The RIAEC Commissioners provide the general direction for the utilization of the facility.

6.1.2.2 Director

The Director of the RIAEC is the organization head, and shall be responsible for the license, and for developing and directing all of the administrative and technical programs. The Director shall be responsible for ensuring facility compliance with federal and state licenses and regulations, and for all activities in the reactor facility which may affect reactor operations or involve radiation hazards. This individual is level 1 management.

6.1.2.3 Assistant Director for Operations

The Assistant Director for Operations shall be responsible for implementing the operations programs and managing the operation of the RINSC facility. The Assistant Director shall be responsible for ensuring that operation of the reactor is compliant with the provisions of the RINSC License and Technical Specifications. This individual is level 2 management.

6.1.2.4 Assistant Director for Radiation and Reactor Safety

The Assistant Director for Radiation and Reactor Safety shall be responsible for implementing and managing the Radiation Safety Program. The Assistant Director shall ensure that that the public and facility personnel are safeguarded from undue exposure to radiation, and that the facility is compliant with federal and state radiation safety regulation. This individual is level 2 management.

6.1.2.5 Reactor Supervisor

The Reactor Supervisor shall be responsible for the day to day operation of the facility. This individual is level 3 management.

6.1.2.6 Senior Reactor Operators

The Senior Reactor Operator on duty during reactor operations shall be responsible for directing the licensed activities of Reactor Operators. The Senior Reactor Operator shall ensure that the operability of the reactor is compliant with the RINSC License and Technical Specifications during operation, and that any experiments performed during operation have been reviewed and approved by the NRSC, and are installed in accordance with any limitations prescribed by the approved experiment. The Senior Reactor Operator shall also ensure that experimenters follow facility procedures.

6.1.2.7 Reactor Operators

The Reactor Operator on duty during reactor operations shall be responsible for manipulating the controls of the reactor. The Reactor Operator shall direct the actions of Reactor Operator Trainees, and ensure that the reactor is operated within the limits of the RINSC Technical Specifications.

6.1.3 Staffing

6.1.3.1 Minimum Staffing Requirements

6.1.3.1.1 The minimum staffing requirements when the reactor is not secured there shall be a licensed Reactor Operator in the control room.

6.1.3.1.2 The minimum staffing requirements when all of the shim safety rods are not fully inserted into the core shall be two individuals present in the facility:

6.1.3.1.2.1 A Reactor Operator in the control room, and

6.1.3.1.2.2 A second individual present in the facility that has security access to the confinement building and is capable of scramming the reactor, initiating a facility evacuation, and notifying RINSC staff members and appropriate response agencies.

6.1.3.1.3 If the Senior Reactor Operator on duty is not serving as the Reactor Operator or the second individual present in the facility, they shall be readily available on call.

6.1.3.2 A Senior Reactor Operator shall be present in the facility as defined in section 5.1 during any of the following operations:

6.1.3.2.1 The initial reactor start-up and approach to power for the day,

6.1.3.2.2 Fuel element, reflector element, or control rod core position changes,

6.1.3.2.3 Recovery from an unscheduled shutdown or an unscheduled power reduction in excess of 25%.

6.1.3.3 Staff Contact List

6.1.3.3.1 A staff contact list that includes management, radiation safety, and other operations personnel shall be available in the control room for use by the Reactor Operator.

6.1.4 Selection and Training of Personnel

6.1.4.1 Qualification

6.1.4.1.1 Rhode Island Atomic Energy Commissioners

The RIAEC Commissioners shall be aware of the general operational and emergency aspects of the reactor facility.

6.1.4.1.2 Director

At the time of the appointment to the position, the Director shall have a minimum of six years of nuclear experience. The individual shall have a Bachelor of Science degree or higher in an engineering or scientific field, or an equivalent combination of education and experience. The degree may fulfill up to four years of the six years of nuclear experience required.

6.1.4.1.3 Assistant Director for Operations

At the time of the appointment to the position, the Assistant Director shall have a minimum of six years of nuclear experience. The individual shall have a Bachelor of Science degree or higher in an engineering or scientific field, or an equivalent combination of education and experience. The degree may fulfill up to four of the six years of nuclear experience required.

6.1.4.1.4 Assistant Director for Radiation and Reactor Safety

At the time of the appointment to the position, the Assistant Director shall have a minimum of three years of health physics experience. The individual shall have a Bachelor of Science degree or higher in an engineering or scientific field, or an equivalent combination of education and experience. The degree may fulfill up to two years of the three years of nuclear experience required.

6.1.4.1.5 Reactor Supervisor

At the time of the appointment to the position, the Reactor Supervisor shall have a minimum of three years of nuclear experience, and have the training to satisfy the requirements for being a licensed Senior Reactor Operator. A maximum of two years of full time academic training may be substituted for two of the three years of nuclear experience.

6.1.4.1.6 Senior Reactor Operators

Senior Reactor Operators shall be licensed pursuant to 10 CFR Part 55.

6.1.4.1.7 Reactor Operators

Reactor Operators shall be licensed pursuant to 10 CFR Part 55.

6.1.4.2 Initial Training and Licensing

Personnel that require a Reactor Operator or Senior Reactor Operator license shall be trained in accordance with the facility Operator Training Program.

6.1.4.3 Re-Qualification and Re-Licensing

As a condition of maintaining their operating licenses, Reactor and Senior Reactor Operators shall participate in the facility Operator Re-Licensing Program.

6.1.4.4 Medical Certification

Facility senior management shall certify that the health of each Reactor Operator and Senior Reactor Operator is such that they will be able to perform their assigned duties. This certification shall be maintained in accordance with 10 CFR Part 55.21.

6.2 Review and Audit

6.2.1 Nuclear and Radiation Safety Committee (NRSC) Composition and Qualifications

6.2.1.1 Composition

The NRSC shall be comprised of a minimum of seven individuals:

6.2.1.1.1 The Director

6.2.1.1.2 The Assistant Director for Operations

6.2.1.1.3 The Assistant Director for Radiation and Reactor Safety

6.2.1.1.4 Four members that are not RIAEC commissioners or staff

6.2.1.2 Qualification

The collective qualification of the NRSC members shall represent a broad spectrum of expertise in science and engineering.

6.2.1.3 Alternates

Qualified alternates may serve in the absence of regular members.

6.2.2 Nuclear and Radiation Safety Committee Charter

The NRSC shall have a written Charter that specifies:

6.2.2.1 Meeting frequency of not less than once per year.

6.2.2.2 Quorum shall consist of a minimum of four (4) members, including the Assistant Director for Radiation and Reactor Safety or designee, and the Director or Assistant Director for Operations.

6.2.2.3 NRSC Minutes shall be reviewed and approved at the next committee meeting.

6.2.3 Review Function

All review results will be documented in the NRSC meeting minutes. The NRSC shall review the following items:

6.2.3.1 Proposed changes to the Technical Specifications,

6.2.3.2 Violations of the Technical Specifications,

6.2.3.3 Proposed changes to the License,

6.2.3.4 Violations of the License,

6.2.3.5 Proposed changes to the NRSC Charter, 6.2.3.6 10 CFR Part 50.59 evaluations,

6.2.3.7 New procedures,

6.2.3.8 Major changes to procedures that have safety significance,

6.2.3.9 Violations of procedures that have safety significance,

6.2.3.10 New experiments,

6.2.3.11 Operating abnormalities that have a safety significance, and

6.2.3.12 Reportable occurrences.

6.2.4 Audit Function

The audit function is normally performed in conjunction with a scheduled NRSC meeting. The non-RIAEC staff members of the NRSC shall audit the following items either before or after the meeting and identify any discrepancies for resolution:

6.2.4.1 Reactor operations shall be audited at least annually to verify that the facility is operated in a manner consistent with public safety and within the terms of the facility license.

6.2.4.2 The Operator Re-Qualification Program shall be audited at least biennially,

6.2.4.3 The Emergency Plan and Emergency Plan Implementing Procedures shall be audited at least biennially,

6.2.4.4 Actions taken to correct any deficiencies found in the facility equipment, systems, structures, or methods of operation that could affect reactor safety shall be audited at least annually, and

6.2.4.5 The Radiation Safety Program shall be audited at least annually.

6.2.4.6 Results of the audit will be captured in the NRSC meeting minutes

6.3 Radiation Safety

The facility shall have a qualified, designated individual that is responsible for implementing the Radiation Safety Program in accordance with 10 CFR Part 20. The Assistant Director for Radiation and Reactor Safety shall be the individual in the organization that fulfills this requirement. A qualified alternative may serve in this capacity if the Assistant Director is unavailable for an extended period of time.

6.4 Procedures

6.4.1 Written procedures shall be used that are adequate to assure the safe operation of the reactor, but should not preclude the use of independent judgment and action should the situation require such.

6.4.2 The procedures for the following activities shall be reviewed by the NRSC, and approved by level 1 or level 2 management prior to use:

6.4.2.1 Startup, operation, and shutdown of the reactor,

6.4.2.2 Fuel loading, unloading, and movement within the reactor,

6.4.2.3 Maintenance of major components of systems that could have an effect on reactor safety,

6.4.2.4 Surveillance checks, calibrations, and inspections that are required by the Technical Specifications, or have a significant effect on reactor safety,

6.4.2.5 Radiation safety,

6.4.2.6 Administrative controls for operations, maintenance, and experiments that could affect reactor safety or core reactivity,

6.4.2.7 Implementation of the Emergency and Security plans, and.

6.4.2.8 Receipt, use, and transfer of byproduct material.

6.5 Experiment Review and Approval

6.5.1 All new experiments shall be reviewed by the NRSC, and approved by level 1 or level 2 management prior to bringing the reactor to power with the experiment loaded.

6.5.2 Substantive changes to previously approved experiments shall be reviewed by the NRSC, and approved by level 1 or level 2 management prior to bringing the reactor to power with the experiment loaded.

6.5.3 Minor changes that do not significantly alter the experiment may be approved by a Senior Reactor Operator or level 1, 2, or 3 management.

6.6 Required Actions

6.6.1 Action to be Taken in the Event of a Safety Limit Violation

6.6.1.1 The reactor shall be shut down and reactor operations shall not be resumed until authorization is obtained from the NRC.

6.6.1.2 Immediate notification shall be made to the Director and to the NRSC members.

6.6.1.3 Notification shall be made to the NRC in accordance with paragraph 6.7.2 of these specifications.

6.6.1.4 A safety limit violation report shall be prepared. The report shall include:

6.6.1.4.1 A complete analysis of the causes of the event,

6.6.1.4.2 The extent of possible damage to facility components, systems, or structures

6.6.1.4.3 A statement regarding the impact of the event on the facility personnel.

6.6.1.4.4 A statement regarding the impact of the event on the public.

6.6.1.4.5 A description of any radioactive material release to the environment.

6.6.1.4.6 Corrective actions taken to prevent or reduce the probability of recurrence.

6.6.1.5 The safety limit violation report shall be submitted to the NRSC for review and appropriate action.

6.6.1.6 The safety limit violation report shall be submitted to the NRC in accordance with Paragraph 6.7.2 of these specifications in support of a request for authorization to resume reactor operations.

6.6.2 Action to be Taken in the Event of a Reportable Occurrence Other Than a Safety Limit Violation

6.6.2.1 The reactor shall be shutdown.

6.6.2.2 The Senior Reactor Operator shall be notified promptly and corrective action shall be taken immediately to place the facility in a safe condition until the cause of the reportable occurrence is determined and corrected.

6.6.2.3 The occurrence shall be reported to the Director or Assistant Director.

6.6.2.4 Operations shall not be resumed without authorization from the Director or Assistant Director for Operations.

6.6.2.5 The occurrence, and corrective action taken shall be reviewed by the NRSC during its next scheduled meeting.

6.6.2.6 Notification shall be made to the NRC in accordance with Paragraph 6.7.2 of these specifications.

6.7 Reports

6.7.1 Annual Report

A written report shall be submitted annually to the NRC following the 30th of June of each year, and shall include a summary of reactor operating experience. The following information shall be provided as a minimum:

6.7.1.1 A summary of the number of hours that the reactor was critical for the period, the energy produced for the period, and the cumulative total energy output since initial criticality;

6.7.1.2 A summary of the unscheduled shutdowns that occurred during the period, the causes of the shutdowns, and if applicable, corrective action taken to preclude recurrence;

6.7.1.3 A summary of any major maintenance performed during the period that has safety significance, and the reasons for any corrective maintenance required;

6.7.1.4 A summary of 10 CFR Part 50.59 safety evaluations made during the reporting period;

6.7.1.5 A summary of the amount of radioactive effluents, and to the extent possible, an estimate of the individual radionuclides that have been released or discharged to the environs outside the facility as measured at or prior to the point of release.

If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient for the summary.

6.7.1.6 A summary of the results of environmental surveys performed outside the facility during the reporting period that includes the locations of the surveys; and

6.7.1.7 A summary of annual radiation exposures in excess of 500 mrem received by facility personnel, or 100 mrem received by non-staff members, or 10 mrem received by members of the general public.

6.7.2 Special Reports

6.7.2.1 Reporting Requirements for Reportable Occurrences

In the event of a reportable occurrence, the following notifications shall be made:

6.7.2.1.1 Within one working day after the occurrence has been discovered, the NRC Headquarters Operation Center shall be notified by telephone, with written follow-up confirmation within 24 hours, at the number listed in 10 CFR Part 20 Appendix D, and

6.7.2.1.2 Within 14 days after the occurrence has been discovered, a written report that describes the circumstances of the event shall be sent to the NRC Document Control Desk at the address listed in 10 CFR Part 50.4.

6.7.2.2 Other Reporting Requirements

6.7.2.2.1 A written report shall be submitted to the NRC within 30 days after the following occurs:

6.7.2.2.1.1 Permanent changes in the facility organization involving level 1 or 2 personnel.

6.7.2.2.1.2 Significant changes in the transient or accident analysis as described in the Safety Analysis Report

6.8 Records

6.8.1 Records that shall be retained for a period of at least five years

6.8.1.1 Reactor operating records,

6.8.1.2 Principal maintenance activities,

6.8.1.3 Surveillance activities required by the Technical Specifications,

6.8.1.4 Facility radiation monitoring surveys,

6.8.1.5 Experiments performed with the reactor,

6.8.1.6 Fuel inventories and transfers,

6.8.1.7 Changes to procedures, and

6.8.1.8 NRSC meeting minutes, including audit findings.

6.8.2 Records that shall be retained for a period of at least one certification cycle

6.8.2.1 Current Reactor Operator re-qualification records shall be maintained for each individual licensed to operate the reactor until their license is terminated.

6.8.3 Records that shall be retained for the life of the facility

6.8.3.1 Gaseous and liquid radioactive effluents released to the environs,

6.8.3.2 Off-site environmental monitoring surveys,

6.8.3.3 Personnel radiation exposures,

6.8.3.4 Drawings of the reactor facility, and

6.8.3.5 Reportable occurrences.