

RHODE ISLAND NUCLEAR SCIENCE CENTER

Technical Specifications

February 26, 2016

Table of Contents

1.0	Definitions	5
2.0	Safety Limits and Limiting Safety System Settings	12
2.1	Safety Limit	12
2.2	Limiting Safety System Settings	12
3.0	Limiting Conditions for Operation.....	17
3.1	Core Parameters.....	17
3.2	Reactor Control and Safety System.....	20
3.3	Coolant System.....	25
3.4	Confinement System	27
3.5	Ventilation System.....	32
3.6	Emergency Power System	34
3.7	Radiation Monitoring System and Effluents.....	35
3.8	Experiments.....	39
3.9	Reactor Core Components.....	42
4.0	Surveillance Requirements	47
4.1	Core Parameter Surveillance.....	47
4.2	Reactor Control and Safety System Surveillance.....	49
4.3	Coolant System Surveillance	52
4.4	Confinement System Surveillance	55
4.5	Ventilation System Surveillance	57
4.6	Emergency Power System Surveillance	59
4.7	Radiation Monitoring System and Effluent Surveillance.....	59
4.8	Experiment Surveillance	61
4.9	Facility Specific Surveillance.....	62
5.0	Design Features.....	65
5.1	Site and Facility Description.....	65
5.2	Reactor Coolant System Description.....	67
5.3	Reactor Fuel and Core Description.....	69
5.4	Fissionable Material Storage Description	70
6.0	Administrative Controls.....	72
6.1	Organization	72
6.2	Review and Audit.....	76
6.3	Radiation Safety.....	78
6.4	Procedures.....	78
6.5	Experiment Review and Approval.....	78
6.6	Required Actions.....	78
6.7	Reports	79
6.8	Records	80

List of Tables

Table 3.1: Instrumentation Required for Reactor Operation	22
Table 3.2: Historical Wind Rose Data	39

List of Figures

Figure 6.1: Rhode Island Atomic Energy Commission Organization Chart	72
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1.0 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.

1.7 Controlled Access Area

Any temporarily or permanently established area which is clearly demarcated, access to which is controlled, and which affords isolation of the material or persons within it.

1.8 Controlled Area

Areas that are outside of restricted areas but inside the site boundary, access to which can be limited by the licensee for any reason.

1.9 Core Configuration

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

1.10 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.

1.11 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.

1.12 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.

1.13 Explosive Material

Explosive material is any solid or liquid which is either:

- 1.13.1 Categorized as a severe, dangerous, or very dangerous explosion hazard in Sax's Dangerous Properties Of Industrial Materials by Richard J. Lewis, Sr., 11th Ed. (2004), or
- 1.13.2 Is given an identification of Reactivity (Stability) Index of 2, 3, or 4 by the National Fire Protection Association (NFPA) in its publication NFPA 704: Standard System for the Identification of the Hazards of Materials for Emergency Response, 2007 Edition.

1.14 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 must 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.

1.15 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.

1.16 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.

1.17 May

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

1.18 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, and natural convection cooling mode.

1.19 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.

1.20 Operable

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

1.21 Operating

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

1.22 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.

1.23 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:

- 1.23.1 Insertion or removal from the core,
- 1.23.2 Intended or anticipated changes in position, or
- 1.23.3 Credible malfunctions that alter experiment position or configuration.

1.24 Reactor Operating

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

1.25 Reactor Operator

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

1.26 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 reactor operator.

1.27 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.

1.28 Reactor Secured

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

- 1.28.1 There is insufficient moderator available in the reactor to attain criticality,
- 1.28.2 There is insufficient fissile material present in the reactor to attain criticality, or
- 1.28.3 The following conditions exist:
 - 1.28.3.1 The reactor is shutdown,
 - 1.28.3.2 The master switch is in the off position and the key is removed from the lock,
 - 1.28.3.3 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
 - 1.28.3.4 No experiments are being moved or serviced that have a reactivity worth of greater than $0.6\% \Delta k/k$ when moved.

1.29 Reactor Shutdown

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

1.30 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.

1.31 Reference Core Condition

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

1.32 Regulating Rod or Regulating Blade

The regulating rod is a control rod of low reactivity worth used primarily to maintain an intended power level. It is not required to have a scram capability, and may be controlled manually or by servo controller.

1.33 Reportable Occurrence

A reportable occurrence is any of the following:

- 1.33.1 A violation of a safety limit,
- 1.33.2 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 Appendix B of 10CFR20,
- 1.33.3 Operation with a safety system setting less conservative than the limiting setting established in the Technical Specifications,
- 1.33.4 Operation in violation of a limiting condition for operation established in the Technical Specifications,
- 1.33.5 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,
- 1.33.6 An uncontrolled or unanticipated change in reactivity in excess of 0.75 % Δ k/k,
- 1.33.7 Abnormal and significant degradation of the fuel cladding,
- 1.33.8 Abnormal and significant degradation of the primary coolant boundary, or
- 1.33.9 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.

1.34 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.

1.35 Safety Channel

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

1.36 Safety Limits

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

1.37 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.

1.38 Security Area

Permanently established areas which are clearly demarcated, access to which is controlled, and which affords isolation of the material, equipment, and persons within it.

1.39 Senior Reactor Operator

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.

1.40 Shall

The word "shall" is used to denote a requirement.

1.41 Shim Safety Rod or Shim Safety Blade

A shim safety rod 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 a scram capability.

1.42 Should

The word "should" is used to denote a recommendation.

1.43 Shutdown Margin

Shutdown Margin shall mean the minimum amount of negative reactivity inserted into the core when the most reactive control blade and the regulating rod are fully withdrawn, and the remaining control blades are fully inserted into the core.

1.44 Site Boundary

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

1.45 Subcritical

The reactor is subcritical if the control rods are providing sufficient negative reactivity in the core to prevent criticality.

1.46 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.

1.47 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:

- 1.47.1 5 years (interval not to exceed 6 years).
- 1.47.2 2 years (interval not to exceed 2 1/2 years).
- 1.47.3 Annual (interval not to exceed 15 months).
- 1.47.4 Semiannual (interval not to exceed 7 1/2 months).
- 1.47.5 Quarterly (interval not to exceed 4 months).
- 1.47.6 Monthly (interval not to exceed 6 weeks).
- 1.47.7 Weekly (interval not to exceed 10 days).
- 1.47.8 Daily (must be done during the calendar weekday).

1.48 True Value

The true value is the actual value of a parameter.

1.49 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:

- 1.49.1 Actuation of the reactor safety system,
- 1.49.2 Operator error,
- 1.49.3 Equipment malfunction, or
- 1.49.4 Manual shutdown in response to conditions that could adversely affect safe operation.

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 true value of the reactor fuel cladding shall be less than or equal to 530 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 C.

2.2 Limiting Safety System Settings

2.2.1 Limiting Safety System Settings for Natural Convection Mode of 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 fuel meat shall be 23 feet 7 inches.
- 2.2.1.3 The limiting safety system setting for the bulk pool temperature shall be 127 F.

Bases:

This combination of specifications was set to prevent the cladding temperature from approaching the 530 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 fuel meat is at least 23 feet 6.5 inches, and
- The bulk pool temperature is no greater than 130 F.

Under these conditions for coolant height and bulk pool temperature, peak channel power would have to reach 1.7812 kW in order for the onset of nucleate boiling to occur, which corresponds to a fuel cladding temperature that is below the 530 C value at which damage to the fuel cladding could occur. The hottest channel reaches a peak power of 1.7812 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 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 fuel meat is at least 23 feet 7 inches,
- The bulk pool temperature is no greater than 130 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:

$$530 \text{ C} - 78.9 \text{ C} = 451.1 \text{ C}.$$

Measurement uncertainty was based on the nominal operating values of 100 kW and 108 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 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 fuel meat 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 2.2.1.3 sets the limiting safety system setting for the bulk pool temperature to be 127 F. The analyses show that cladding damage will not occur under any condition if the pool temperature is no greater than 130 F. Taking into consideration a 3 F in measurement error, if the LSSS is 127 F, then the Limiting Trip Value could never be greater than 130 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 fuel meat shall be 23 feet 7 inches.
- 2.2.2.3 The limiting safety system setting for the primary coolant inlet temperature shall be 122 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 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 2.4 MW,
- The coolant height above the fuel meat is at least 23 feet 6.5 inches,
- The primary coolant outlet temperature is no greater than 125 F, and
- The coolant flow rate through the core is at least 1580 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 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 fuel meat 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:

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

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 outlet 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 fuel meat 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 as 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 in 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 % Δ k/k.
- 3.1.1.1.2 The core excess reactivity shall not exceed 4.7 % Δ k/k.
- 3.1.1.1.3 The temperature coefficient shall be negative.
- 3.1.1.1.4 The reactor shall be subcritical by at least 3.0 % Δ k/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 % Δ k/k.

3.1.1.3 Experiments

- 3.1.1.3.1 The total absolute reactivity worth of experiments shall not exceed the following limits, except when the operation of the reactor is for the purpose of measuring experiment reactivity worth by doing criticality studies:

- 3.1.1.3.1.1 Total Moveable and Fixed 0.6 % Δ k/k
- 3.1.1.3.1.2 Total Moveable 0.08 % Δ k/k

- 3.1.1.3.2 The maximum reactivity worth of any individual experiment shall not exceed the following limits, except when the operation of the reactor is for the purpose of measuring experiment reactivity worth by doing criticality studies:

- 3.1.1.3.2.1 Fixed 0.6 % Δ k/k
- 3.1.1.3.2.2 Moveable 0.08 % Δ k/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. An allowance is made for measuring the reactivity worth of experiments. The reactor can be made critical with experiments of unknown reactivity, so that the criticality data can be used to determine whether or not the reactivity worth of an experiment is within the limits prescribed by this specification.

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. An allowance is made for measuring the reactivity worth of experiments. The reactor can be made critical with experiments of unknown reactivity, so that the criticality data can be used to determine whether or not the reactivity worth of an experiment is within the limits prescribed by this specification.

Specification 3.1.1.1.3 requires that the temperature coefficient be negative. This requirement ensures that a temperature rise due to a reactor transient will not cause a further increase in reactivity. Neutron cross sections in seven energy groups as functions of moderator temperature, fuel temperature, and coolant void fraction were prepared using the WIM/ANL cross section generation code¹. Keff values were computed using the DIF3D diffusion theory code. Coefficients of reactivity were determined from these data. The coolant temperature coefficient was determined to be negative for temperatures between 20 C to 100 C. The fuel temperature coefficient was determined to be negative relative to 20 C for temperatures between 20 C and 600 C.

Specification 3.1.1.1.4 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, and regulating control rods. 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 % $\Delta k/k$ 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 operations above 0.1 MW when the reactor is in the forced convection cooling mode.

Objective:

The objective of these specifications is to ensure that there is sufficient coolant to remove heat from the fuel elements when the reactor is in operation at power levels greater than 0.1 MW.

Specifications:

- 3.1.2.1 All core grid positions shall contain fuel elements, baskets, reflector elements, or experimental facilities during operations at power levels in excess of 0.1 MW in the forced convection cooling mode.

- 3.1.2.2 The pool gate shall be in its storage location during operations at power levels in excess of 0.1 MW in the forced convection cooling mode.

Bases:

Specification 3.1.2.1 requires that all of the core grid spaces be filled when the reactor is operated at higher power levels that require forced convection cooling. 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 gate that is used for separating the sections of the pool, be in its storage location when the reactor is in operation at higher power levels that require forced convection cooling. This requirement ensures that there will be a sufficient heat sink for high power operations, 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:

3.2.1 The reactor shall not be operated unless:

- 3.2.1.1 All 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.
- 3.2.1.2 The reactivity insertion rates of individual shim safety and regulating rods does not exceed 0.02% Δ k/k per second.
- 3.2.1.3 The instrumentation shown in Table 3.1 is operable and capable of performing its intended function:

Protection	Power Level	Channels Required	Function	Set Point
Over Power	All	2	Scram before power is greater than	115% of Licensed Power
Rate of Change of Power	All	1	Scram before period is less than	4 seconds
Detector HV Failure	All	1	Scram before first power channel HV is less than	50 V below suggested operating voltage
	All	1	Scram before second power channel HV is less than	50 V below suggested operating voltage
	All	1	Scram before period channel HV is less than	50 V below suggested operating voltage
Low Pool Level	All	1	Scram before pool level is less than	23 feet 7 inches above the top of the fuel meat
Manual Scram	All	1	Scram when	Control Room Scram Button Depressed
Servo Control Interlock	All	1	No regulating rod automatic servo if	Regulating rod not full out
	All	1	No regulating rod automatic servo if period is less than	30 seconds
Shim Safety Blade Withdrawal Interlock	All	1	No shim safety blade withdrawal if start up channel count rate less than	3 counts per second
	All	1	No shim safety blade withdrawal if Neutron Flux Monitor Test / Select switch is	Not in the Off position
Control Rod Drive Communication	All	1	Scram if loss of communication for greater than	10 seconds
Seismic Disturbance	All	1	Scram when	Seismic Disturbance Detected
Bridge Movement	All	1	Scram when	Bridge Movement Detected
Pool Temperature	<=100kW	1	Scram before temperature is greater than	127 F

Protection	Power Level	Channels Required	Function	Set Point
Primary Coolant Inlet Temperature	>100kW	1	Scram before temperature is greater than	122 F
Primary Coolant Flow Rate	>100kW	1	Scram before flow rate is less than	1560 gpm
Coolant Gates Open	>100kW	1	Scram when	Inlet or outlet gate open
No Flow Thermal Column	>100kW	1	Scram when	No Flow Detected
Bridge Low Power Position	>100kW	1	Scram when	Bridge Not Seated at HP End

Table 3.1: Instrumentation Required for Reactor Operation

Bases:

Specification 3.2.1.1 requires that all shim safety blades be 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 3.2.1.2 requires that the reactivity insertion rates of individual shim safety and regulating rods do not exceed 0.02 % Δ k/k per second. As part of the Safety Analysis, Argonne National Laboratory analyzed ramp insertions of 0.02 % Δ k/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 3.2.1.3 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 scrambling the reactor by 115% 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 of the power channels, and the period channel. 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 fuel meat. The low pool level channel LSSS is 23 feet 7 inches above the fuel meat. Taking into consideration a 0.5 in measurement error, this LSSS ensures that the pool height above the fuel meat 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 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 start up neutron count rate is less than 3 cps is required for both modes of operation. This interlock ensures that the start up 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 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 scrambling the reactor when the temperature reaches 127 F. The basis section of Specification 2.2.1.3 shows that this ensures that the pool temperature will not exceed the 130 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 scrambling the reactor when the temperature reaches 122 F. The basis section of Specification 2.2.2.3 shows that this LSSS will ensure that the coolant outlet temperature will not exceed the 125 F temperature that was used in the thermohydraulic 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 thermohydraulic 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 on each coolant duct is required during forced convection cooling mode operation. These scrams ensure 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.

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.

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 primary coolant conductivity shall be $\leq 2 \mu\text{mho} / \text{centimeter}$.

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 have 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:

Cesium - 137 and Iodine - 131 activity in the primary coolant shall be maintained at levels that are indistinguishable from background.

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 indicate a fuel cladding leak.

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 3.3.2.1 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

3.4.1 Operations That Require Confinement

Applicability:

These specifications apply to the operations for which the components of the confinement system must be operable.

Objective:

The objective of these specifications is to assure that operations that have the potential to release airborne radioactive material are performed under conditions in which the release to the environment would be detected, and be limited to levels below 10 CFR 20 limits.

Specifications:

3.4.1.1 The confinement system shall be operable whenever:

- 3.4.1.1.1 The reactor is operating.
- 3.4.1.1.2 Irradiated fuel handling is in progress.
- 3.4.1.1.3 Experiment handling is in progress for an experiment that has a significant fission product, or gaseous effluent activation product inventory, and for which the experiment is not inside a container.
- 3.4.1.1.4 Any work on the core or control rods that could cause a reactivity change of more than 0.60 % Δ k/k is in progress.
- 3.4.1.1.5 Any experiment movement that could cause a reactivity change of more than 0.60 % Δ k/k is in progress.

Bases:

The purpose of the confinement system is to mitigate the consequences of airborne radioactive material 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 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.

3.4.2 Components Required to Achieve Confinement

3.4.2.1 Normal Operating Mode Confinement

Applicability:

These specifications describe the components of the confinement system that are necessary in order for the system to perform its intended function under normal operating conditions.

Objective:

The objective of these specifications is to assure that the confinement system is capable of detecting a release of airborne radioactive material.

Specifications:

3.4.2.1.1 The following confinement system components shall be operable whenever the confinement system is required to be operable:

- 3.4.2.1.1.1 Normal Personnel Access Door
- 3.4.2.1.1.2 Roll Up Door
- 3.4.2.1.1.3 Roof Hatch

Bases:

The personnel access door, roll up door, and roof hatch represent the major potential air access ways through confinement. If these components are operable, the major potential air pathways are capable of being controlled to ensure that any airborne radiological release would be detected either by the confinement radiation monitoring system, or by the stack effluent monitoring system.

Under normal operating conditions, the normal operating mode ventilation system controls the general airflow from outside confinement, through confinement, and back out to the environment through the stack.

3.4.2.2 Emergency Operating Mode Confinement

Applicability:

These specifications describe the components of the confinement system that are necessary in order for the system to perform its intended function under emergency operating conditions.

Objective:

The objective of these specifications is to assure that the confinement system is capable of mitigating the consequences of a possible release of airborne radioactive material.

Specifications:

3.4.2.2.1 The following emergency confinement system components shall be operable:

- 3.4.2.2.1.1 Emergency Confinement System Buttons
- 3.4.2.2.1.2 Confinement Air Intake Damper
- 3.4.2.2.1.3 Confinement Air Exhaust Damper
- 3.4.2.2.1.4 Emergency Personnel Access Door

Bases:

Under emergency conditions, operability of any of the emergency confinement system buttons allows the path of the airflow from confinement, through the ventilation system to be changed so that it goes through the emergency filter. Operability of the confinement air intake and exhaust dampers allows the confinement building to be isolated from the outside so that no exhaust confinement air escapes through a pathway other than the emergency pathway. Emergency mode operation of the ventilation system ensures that under emergency conditions, confinement air will be drawn through the emergency filter before being exhausted through the stack. Operability of the filter minimizes the environmental consequence of a potential airborne radioactivity release. Emergency mode operation of the ventilation system also ensures that dilution air will be added to the confinement air from the emergency filter. Operability of the emergency personnel access door allows the reactor operator to have a confinement egress route that does not require the individual to go through the main confinement room. When the door is shut, confinement is maintained.

3.4.3 Conditions Required to Achieve Confinement

3.4.3.1 Normal Operating Mode Confinement

Applicability:

These specifications describe the conditions necessary to assure that normal operating mode confinement is achieved.

Objective:

The objective of these specifications is to assure that the confinement system is functioning sufficiently to prevent airflow from inside confinement to the environment through an uncontrolled pathway.

Specifications:

The following conditions shall be met in order to ensure that the normal confinement is achieved:

- 3.4.3.1.1 The Normal Personnel Access Door is closed, except for entry and exit.
- 3.4.3.1.2 The Roll Up Door is closed.
- 3.4.3.1.3 The Roof Hatch is closed.
- 3.4.3.1.4 The Emergency Personnel Access Door is closed, except for entry and exit.
- 3.4.3.1.5 The Confinement Dampers are Open.
- 3.4.3.1.6 The negative differential pressure inside confinement with respect to the outside is at least 0.5 inches of water.

Bases:

Normal confinement is maintained by keeping all of the doors and the roof hatch closed, except for entry and exit. A negative differential pressure of 0.5 inches of water makes certain that the confinement system is performing its intended function adequately by ensuring that confinement airflow is directed through a defined pathway that is monitored for radiological release. The differential pressure is achieved by circulating air from outside confinement, through the intake damper, and ultimately back out of confinement through the exhaust damper.

3.4.3.2 Emergency Operating Mode Confinement

Applicability:

These specifications describe the conditions necessary to assure that emergency operating mode confinement is achieved.

Objective:

The objective of these specifications is to assure that the confinement system is functioning sufficiently to prevent airflow from inside confinement to the environment through an uncontrolled pathway, and to assure that the confinement airflow pathway to the environment goes through the emergency filter and is mixed with dilution air prior to being exhausted out of the stack.

Specifications:

The following conditions shall be met in order to ensure that the emergency confinement is achieved:

- 3.4.3.2.1 The Normal Personnel Access Door is closed, except for entry and exit.
- 3.4.3.2.2 The Roll Up Door is closed.
- 3.4.3.2.3. The Roof Hatch is closed.
- 3.4.3.2.4 The Emergency Personnel Access Door is closed, except for entry and exit.
- 3.4.3.2.5 The Confinement Dampers are Closed.
- 3.4.3.2.6 The negative differential pressure inside confinement with respect to the outside is at least 0.5 inches of water.

Bases:

Emergency confinement is maintained by closing the confinement intake and exhaust dampers, and by keeping all of the doors and the roof hatch closed, except for entry and exit. This causes all of the make-up confinement air to be drawn in through the spaces around the confinement penetrations, and directed through the Emergency Filter before being exhausted to the stack. A negative differential pressure of 0.5 inches of water makes certain that the confinement system is performing its intended function adequately by ensuring that confinement airflow is directed through the defined pathway that includes the emergency air filter, prior to being released to the environment.

3.5 Ventilation System

3.5.1 Ventilation System Components Required for Normal Operating Mode

Applicability:

This specification describes the ventilation system components that must be operating in order to assure that the normal operating mode confinement is functioning.

Objective:

The objective of this specification is to assure that the normal mode confinement system is capable of performing its intended function.

Specification:

The Confinement Exhaust Blower shall be operating when the reactor is operating.

Basis:

The Confinement Exhaust Blower produces a differential pressure across confinement to ensure that all confinement air pathways are through a controlled pathway that is monitored for radiological release.

3.5.2 Ventilation System Components Required for Emergency Operating Mode

Applicability:

These specifications describe the ventilation system components that must be operating in order to assure that the emergency operating mode confinement is functioning.

Objective:

The objective of these specifications is to assure that the emergency mode confinement system is capable of performing its intended function.

Specifications:

3.5.2.1 The following emergency mode ventilation system components shall be operating whenever the ventilation system is required to be operable:

3.5.2.1.1 Emergency Blower

3.5.2.1.2 Emergency Filter System, which shall include:

3.5.2.1.2.1 Emergency Filter Intake System Roughing Filter

3.5.2.1.2.2 Emergency Filter System Intake Absolute Filter

3.5.2.1.2.3 Emergency Filter System Charcoal Filter

3.5.2.1.2.4 Emergency Filter System Exhaust Absolute Filter

3.5.2.1.3 Dilution Blower

3.5.2.1.4 Confinement Exhaust Stack

3.5.2.2 The exhaust rate through the emergency filter shall be less than or equal to 1500 cfm.

3.5.2.3 The emergency filter shall be at least 99% efficient at removing iodine.

3.5.2.4 The Emergency Filter System Exhaust Absolute Filter shall be certified by the manufacturer to have a minimum efficiency of 99.97% for removing 0.3 micron diameter particulates.

Bases:

Under emergency conditions, the Confinement Exhaust Blower turns off, and differential pressure across confinement is maintained by the Emergency Blower. The Emergency Blower directs confinement air through the Emergency Filter 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 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 System Exhaust Absolute Filter prevents charcoal particulates from the charcoal filter from being released to the building exhaust air stream. The Dilution Blower 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. Though the safety analysis does not take credit for the Confinement Exhaust Stack, the stack ensures that the plume of confinement air that is released to the environment, is released at an elevation of 115 feet above ground level, which provides for an opportunity for the air to disperse prior to the plume reaching ground level.

3.6 Emergency Power System

3.6.1 Required Emergency Power Sources

Applicability:

This specification describes the emergency electrical power sources that are required in order to ensure that power is available to confinement system components that are necessary to make certain that the confinement system is able to perform its intended function in the event of an electrical power outage.

Objective:

The objective of this specification is to assure that the confinement system is able to perform its intended function even, when normal electrical power is unavailable.

Specification:

An emergency electrical power source shall be operable whenever the confinement system is required to be operable.

Basis:

Operability of the emergency electrical power source ensures that the blower systems that are necessary in order to maintain emergency operation mode confinement will remain operable, even in the event of a facility electrical power outage.

3.6.2 Components Required to be Supplied with Emergency Power

Applicability:

These specifications describe the confinement system components that are required to be connected to an emergency electrical power source.

Objective:

The objective of these specifications is to assure that the confinement system is able to perform its intended function even, when normal electrical power is unavailable.

Specifications:

3.6.2.1 The following confinement system components shall be connected to an emergency power source:

3.6.2.1.1 Emergency Blower

3.6.2.1.2 Dilution Blower

Bases:

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 air intake and exhaust dampers are pneumatically operated and will fail closed to isolate the confinement room. 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 source will supply the emergency and dilution blowers with electricity so that they will continue to operate, and the emergency confinement system 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.

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:

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 or particulate effluent shall be operating whenever:

3.7.1.1.1.1 The reactor is operating,

3.7.1.1.1.2 Irradiated fuel handling is in progress,

3.7.1.1.1.3 Experiment handling is in progress for an experiment that has a significant fission product inventory, and for which the experiment is not inside a container,

3.7.1.1.1.4 Any work on the core or control rods that could cause a reactivity change of more than 0.60 % $\Delta k/k$ is in progress, or

3.7.1.1.1.5 Any experiment movement that could cause a reactivity change of more than 0.60 % $\Delta k/k$ is in progress:

- 3.7.1.1.2 If the detector described in specification 3.7.1.1.1 fails during operation, a suitable alternative gaseous or particulate air monitor may be used, or an hourly grab sample analysis may be made 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 over the pool whenever:
 - 3.7.1.1.3.1 The reactor is operating,
 - 3.7.1.1.3.2 Irradiated fuel handling is in progress,
 - 3.7.1.1.3.3 Experiment handling is in progress for an experiment that has a significant fission product inventory, and for which the experiment is not inside a container,
 - 3.7.1.1.3.4 Any work on the core or control rods that could cause a reactivity change of more than 0.60 % $\Delta k/k$ is in progress, or
 - 3.7.1.1.3.5 Any experiment movement that could cause a reactivity change of more than 0.60 % $\Delta k/k$ is in progress:
- 3.7.1.1.4 If the detector described in specification 3.7.1.1.3 fails, a suitable gamma sensitive alternative meter with alarming capability may be placed at the top of the bridge in lieu of the normal detector.
- 3.7.1.1.5 Passive radiation monitors provided by a certified vendor shall be used to provide area radiation monitoring inside confinement when the reactor is in operation, and during fuel handling operations.
- 3.7.1.1.6 Passive radiation monitors provided by a certified vendor shall be used to provide environmental radiation monitoring when the reactor is in operation, and during fuel handling operations.

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 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, and during fuel handling operations.

Specification 3.7.1.1.2 allows for the air monitoring instrumentation to be replaced, or for grab samples 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 fission product monitoring instrumentation that is required to be operable when the reactor is operated, and during fuel handling operations.

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

Specification 3.7.1.1.5 requires that passive area radiation monitoring be in place during reactor operation and fuel handling activities. Active monitoring is performed in accordance with the RINSC Radiation Safety Program, and is tailored to meet the specific needs of each experiment as they are performed. The passive monitoring provides a mechanism for reconstructing maximum doses as a backup to the active systems used.

Specification 3.7.1.1.6 requires that passive environmental radiation monitoring be in place during reactor operation and fuel handling activities. This provides a mechanism for determining doses outside confinement.

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:

- 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.
- 3.7.1.2.4 Alarm set points may be adjusted higher with the approval of the Director or one of the Assistant Directors.

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.

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 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.

Basis:

10 CFR 20.1101(d) states, in part, “to implement the ALARA requirements of § 20.1101 (b), and notwithstanding the requirements in § 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon - 222 and its daughters, shall be established by licensees other than those subject to § 50.34a, 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 10 mrem (0.1 mSv) 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.

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

Table 3.2: 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 ALARA limitation given in 10 CFR 20.1101 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.

Basis:

Liquid effluent discharges are made on a periodic basis. This specification ensures that these discharges are within regulatory release limits.

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.

3.8.1.2 Highly Water Reactive Materials

Highly water reactive materials shall not be placed inside the reactor, the reactor pool, or inside the reactor experimental facilities.

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 96.25 milligrams.

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.

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 and 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 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 design shall be reviewed to ensure that credible failure of any experiment will not result in releases or exposures in excess of limits established in 10 CFR 20.
- 3.8.2.2 Experiment design shall be reviewed 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 design shall be reviewed 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

3.8.2.4 Experiments which could increase reactivity by flooding shall not remain in the core, or adjacent to the core unless the minimum core shutdown margin required would be satisfied with the experiment in the flooded condition.

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.

In order to determine the reactivity worth of a new experiment for which there is no data based on similar experiments, the only way to determine the reactivity worth of the experiment is to perform an approach to critical with the experiment loaded in the core. In that case, it is possible that an experiment could be found to have enough positive reactivity that if additional positive reactivity were added due to flooding, the shutdown margin would be less than 1.0 % $\Delta k/k$. In that event, Technical Specification 3.8.2.4 requires that the experiment be removed immediately.

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 flux shall be 1×10^{22} neutrons/cm².

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.

Objective:

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

Specification:

Fuel elements shall be inspected for physical defects and reactor core box fit in accordance with manufactured specifications.

Basis:

Fuel elements are inspected, and tested for core box fit in accordance with written procedures in order to assure that the reactor is operated with fuel elements that are not damaged.

3.9.3 Experimental Facilities

3.9.3.1 Experimental Facility Configuration During Reactor Operation

Applicability:

These specifications apply to the reactor experimental facilities during reactor operation.

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:

- 3.9.3.1.1 Each beam port shall have no more than an area of 1.25 square inches open to confinement during reactor operation.
- 3.9.3.1.2 When the reactor is in operation, the drain valve to the through port shall be closed.
- 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 being monitored for a leak condition, 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 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 square inches is conservative.

Specification 3.9.3.1.2:

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. The through port has three potential pool leak pathways. The first is the through port drain. By keeping this drain closed during operation, that potential leak pathway is blocked, and the potential for an unnoticed pool leak through this experimental facility is prevented.

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 personnel will become aware of a pool leak through 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 4.5 Hours 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:

3.9.3.2.1 If there is no need to open a beam port within 4.5 hours after reactor shutdown, then the 1.25 square inch area opening to confinement shall be maintained until that time period has passed.

3.9.3.2.2 If there is a need to open a beam port within 4.5 hours after reactor shutdown, then:

3.9.3.2.2.1. 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.

3.9.3.2.2.2. The pool gate 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 3.9.3.2.1

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. Consequently, maintaining the limit on the area open between confinement and the beam ports to 1.25 square inches 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 3.9.3.2.2

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 gate 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

4.1 Core Parameter Surveillance

4.1.1 Reactivity Limit Surveillance

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 Surveillance

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 temperature coefficient shall be shown to be negative at the initial start-up after a fuel type change.

4.1.1.2 Control Rod Reactivity Limit Surveillance

4.1.1.2.1 The reactivity worth of 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

4.1.1.2.2 The reactivity worth of the shim safety rods shall be determined:

- 4.1.1.2.2.1 Annually
- 4.1.1.2.2.2 Whenever the core reflection is changed
- 4.1.1.2.2.3 Whenever the core fuel loading is changed
- 4.1.1.2.2.4 Whenever maintenance is performed that could have an effect on the reactivity worth of the control rod

4.1.1.3 Experiment Reactivity Limit Surveillance

- 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 temperature coefficient be shown to be negative at the initial start-up after a fuel type change. A negative temperature coefficient makes power increases self limiting by inserting a negative reactivity effect as fuel and coolant temperatures rise.

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

Specification 4.1.1.2.2 requires that the shim safety rod reactivities be determined annually, and whenever there is a change in core loading or core reflection. These determinations provide assurance that the rod worths do not degrade due to rod changes, or changes in core configuration.

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 worths be re-determined every time that a recurring core configuration change, such as equilibrium core re-fuelling, occurs.

4.2 Reactor Control and Safety System Surveillance

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 All shim safety reactivity insertion rates shall be measured:

- 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 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

- 4.2.4 The following reactor safety and safety related instrumentation shall be verified to be operable 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 rate scram

- 4.2.5 The following reactor safety and safety related instrumentation alarms, scrams, and interlocks shall be 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

 - 4.2.5.3 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 alarms and scrams:
 - 4.2.5.4.1 Primary inlet temperature scram
 - 4.2.5.4.2 Pool temperature scram

 - 4.2.5.5 The following coolant system channel flow scrams:
 - 4.2.5.5.1 Primary flow scram
 - 4.2.5.5.2 Inlet and outlet coolant gates open scrams
 - 4.2.5.5.3 No flow thermal column scram

 - 4.2.5.6 Low pool level scram

4.2.5.7 The following bridge scrams:

- 4.2.5.7.1 Bridge manual scram
- 4.2.5.7.2 Bridge movement scram
- 4.2.5.7.3 Bridge low power position scram

4.2.5.8 Seismic scram

4.2.5.9 Rod control communication watchdog scram

4.2.6 The following reactor safety and safety related instrumentation shall be calibrated annually:

- 4.2.6.1 Power level channels
- 4.2.6.2 Primary flow channel
- 4.2.6.3 Primary inlet temperature channel
- 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 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 rate 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. The watchdog system monitors the communication link between the control rod drive display computer, and the stepper motor controllers that provide rod motion. None of the nuclear instrumentation protections scrams are affected if the communication link is broken. The impact of a loss of communication is that control rod motion stops, which is analog equivalent of having a control rod drive switch or wire between the drive switch and stepper motor fail.

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 System Surveillance

4.3.1 Primary Coolant System

4.3.1.1 Primary Coolant Conductivity Surveillance

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. Specification 3.3.1.1 sets a limit on the average conductivity when averaged over one quarter of a year. Consequently, a monthly measurement falls within the ANSI recommended schedule, and allows for a running average based on three data points per quarter.

4.3.1.2 Primary Coolant Activity Surveillance

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 annually.

Basis:

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

4.3.1.3 Primary Coolant Level Inspection Surveillance

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 system 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 level reaches one inch less than 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.

4.3.1.4 Primary Coolant System Inspection Surveillance

Applicability:

This specification applies to the surveillance of the primary cooling system components.

Objective:

The objective of this specification is to provide a periodic verification that there are no obvious defects in any of the system components.

Specification:

The components of the primary coolant system shall be inspected annually.

Basis:

Specification 4.3.1.4 requires that the primary coolant system be inspected on an annual basis to ensure that the integrity of the pool and other cooling system components are not degraded. This schedule is consistent with the historical inspection schedule for the facility.

4.3.2 Secondary Coolant System

4.3.2.1 Secondary Coolant Activity Surveillance

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 annually.

Basis:

Specification 4.3.2.1 requires that the Sodium - 24 activity in the secondary coolant be tested on an annual basis. This schedule is consistent with the schedule recommended by ANSI 15.1.

4.3.2.2 Secondary Coolant System Inspection Surveillance

Applicability:

This specification applies to the surveillance of the secondary cooling system components.

Objective:

The objective of this specification is to provide a periodic verification that there are no obvious defects in any of the system components.

Specification:

The components of the secondary coolant system shall be inspected annually.

Basis:

Specification 4.3.2.2 requires that the primary coolant system be inspected on an annual basis. This schedule is consistent with the historical inspection schedule for the facility.

4.4 Confinement System Surveillance

4.4.1 Normal Operating Mode Confinement System

Applicability:

This specification describes the surveillance requirements for the normal operating mode confinement system.

Objective:

The objective of this specification is to verify that the normal operating mode confinement system is functional prior to reactor start-up.

Specification:

The conditions required to achieve normal operating mode confinement that are specified in section 3.4.3.1 shall be verified to be met prior to the each day of reactor start-up.

Basis:

If the conditions specified in section 3.4.3.1 are met, then the normal operating mode confinement system is functioning. By ensuring that the normal operating mode confinement 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, mixed with dilution air, and detected by the stack radiation monitor system.

4.4.2 Emergency Operating Mode Confinement System

Applicability:

These specifications describe the surveillance requirements for the emergency operating mode confinement system.

Objective:

The objective of these specifications is to verify that the emergency operating mode confinement system is functional.

Specifications:

4.4.2.1 A functional test of the emergency operating mode confinement system shall be performed:

4.4.2.1.1 Quarterly

4.4.2.1.2 After any maintenance that could affect the operability of the system

4.4.2.2 The functional test of the emergency operating mode confinement system shall verify that the conditions required to achieve emergency operating mode confinement are met when an evacuation button is depressed. The following actions shall occur when an evacuation button is depressed:

4.4.2.2.1 The evacuation horn sounds

4.4.2.2.2 The following dampers close:

- 4.4.2.2.2.1 Confinement Air Intake Damper
- 4.4.2.2.2.2 Confinement Air Exhaust Damper

- 4.4.2.2.3 The negative differential pressure inside confinement with respect to the outside is at least 0.5 inches of water.
- 4.4.2.2.4 The confinement room HVAC and air conditioners de-energize.

Bases:

A periodic functional test of the emergency confinement system ensures that in the event of an airborne radioactivity release, the emergency confinement system is capable of being activated. The testing periods that are specified conform to ANSI 15.1 recommendations.

4.5 Ventilation System Surveillance

4.5.1 Normal Operating Mode Ventilation System

Applicability:

This specification describes the surveillance requirements for the normal operating mode ventilation system.

Objective:

The objective of this specification is to verify that the normal operating mode ventilation system is operable prior to reactor start-up.

Specification:

The confinement exhaust blower shall be verified to be in operation prior to each day of reactor start-up:

Bases:

By ensuring that the normal operating mode 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.

4.5.2 Emergency Operating Mode Ventilation System

Applicability:

This specification describes the surveillance requirements for the emergency operating mode ventilation system.

Objective:

The objective of this specification is to verify that the emergency operating mode ventilation system is operational and functional.

Specifications:

4.5.2.1. A test of the operability of the emergency operating mode ventilation system shall be performed:

4.5.2.1.1. Quarterly

4.5.2.1.2. After any maintenance that could affect the operability of the system

4.5.2.2. The test of the operability of the emergency operating mode ventilation system shall verify that the following actions occur when an evacuation button is depressed:

4.5.2.2.1. The following blowers are de-energized:

4.5.2.2.1.1. Confinement Exhaust Blower

4.5.2.2.1.2. Rabbit System Blower

4.5.2.2.1.3. Off Gas System Blower

4.5.2.2.2. The following blowers are energized:

4.5.2.2.2.1. Emergency Exhaust Blower

4.5.2.2.2.2. Dilution Blower

4.5.2.3. The flow rate at the exhaust of the emergency exhaust blower shall be verified to be less than or equal to 1500 cfm:

4.5.2.3.1. Annually

4.5.2.3.2. After any maintenance that could affect the operability of the system

4.5.2.4. The emergency filter efficiency shall be verified to be at least 99% efficient for removing iodine:

4.5.2.4.1. Biennially

4.5.2.4.2. After any maintenance that could affect the operability of the system

Bases:

A periodic test of the operability of the emergency ventilation system ensures that in the event of an airborne radioactivity release, the emergency confinement system is capable of being activated. The verification of the emergency exhaust blower flow rate, and the emergency filter efficiency ensure that the filter will perform its intended function. The testing periods that are specified conform to ANSI 15.1 recommendations.

4.6 Emergency Power System Surveillance

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 functional.

Specifications:

- 4.6.1. A quarterly operability test shall be performed in order to verify that the emergency power system starts in the event of a facility power outage.
- 4.6.2. A functional test of the emergency power system under load shall be performed:
 - 4.6.2.1. Biennially
 - 4.6.2.2. Following emergency system load changes
- 4.6.3. The fuel tank levels for the emergency generator shall be verified to be at least 50% full on a monthly basis.

Bases:

Specifications 4.6.1 and 4.6.2 are periodic tests of the emergency power system that 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 system. 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 Effluent Surveillance

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 verified to be operable 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:

- 4.7.1.1.1 The experimental level area radiation monitor
- 4.7.1.1.2 The 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 reactor safety and safety related instrumentation shall be calibrated 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 prior to the initial reactor start-up of each day. This requirement is consistent with the historical facility requirements.

Specification 4.7.1.2 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.

4.7.2. Effluents

4.7.2.1 Liquid Effluent Radiation Monitoring System

Applicability:

This specification applies to the surveillance of the monitoring equipment used to measure the radioactivity in effluents.

Objective:

The objective of this specification is to assure that an accurate assessment of the radiological material release from the facility via effluents can be made.

Specification:

The monitoring equipment used to measure the radioactive concentrations in the waste retention tanks shall be calibrated annually.

Basis:

An annual calibration period is within the range that is recommended by ANSI Standard 15.1 for radiation monitors.

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 before release.

Basis:

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

4.8 Experiment Surveillance

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 50.59.

Specification:

Experiments shall be reviewed to ensure that the design is within the limitations of the RINSC Technical Specifications and 10 CFR 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 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 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 surveillance each year 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, then 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 fuel 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 the 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 surveillance each year shall include at least one fifth of the fuel elements that are in the core,
 - 4.9.2.1.2. The surveillance each year 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 detected by Technical Specification 4.3.3 or otherwise discovered, then the surveillance shall be expanded to include all of the fuel elements prior to use, and annually thereafter.

Bases:

RINSC Technical Specification 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.

5.0 Design Features

5.1 Site and Facility Description

Description:

The Rhode Island Nuclear Science Center (RINSC) site is located on a 3 acre section of a 27 acre auxiliary campus of the University of Rhode Island. The 27 acre site was formerly a military reservation prior to becoming the Bay Campus of the university. The parcel of land is located in the town of Narragansett, Rhode Island, on the west shore of the Narragansett Bay, approximately 22 miles south of Providence, and approximately 6 miles north of the entrance of the bay from the Atlantic Ocean. The controlled area is within the site boundary, and is surrounded by bollards, gates, or trees. This allows control over vehicle access into the area.

The facility is one of a number of buildings located on the Bay Campus of the University of Rhode Island, and is located inside the controlled area. The RINSC facility consists of a reactor room and an office wing with one entrance between them. The facility serves as the restricted area in which personnel access is controlled.

The reactor room acts as the confinement space. A negative pressure is maintained in the room during reactor operation. The reactor room air exhausts to a stack.

In addition to housing the reactor pool, the reactor room is equipped with a crane.

Applicability:

These specifications apply to the RINSC site and facility.

Objective:

The objective of these specifications is to specify facility design features.

Specifications:

- 5.1.1 The reactor room shall have a minimum air volume of 5.15E9 cubic centimeters.
- 5.1.2 The reactor room crane shall not be positioned directly over the high power section of the reactor pool unless the reactor has been shutdown for a minimum of 4.5 hours.

Bases:

The site for the RINSC facility was originally a military fort that was used to guard the west side of the Narragansett Bay against naval attack. The foundation for the reactor building is one of the fort buildings that was designed to withstand mortar shells. There were a series of large military guns that were set up on gun pads along the bay. The reactor pool is constructed on top of one of these pads. Consequently, the foundation for the reactor room, and the base on which the reactor pool rests is very stable.

The controlled area is designed to control vehicle access near the reactor building. Bollards, gates, and natural barriers are used in order to achieve this. The facility building serves as the restricted area. Personnel access into the facility is limited to individuals that have been authorized to have a key to the facility, and visitors or users that are given access during normal facility hours of operation.

The reactor room functions as confinement. The negative pressure in the room provides a controlled pathway for airflow into, and out of the reactor room, and ensures that there will be no uncontrolled discharge into the environment. The air is monitored for radioactive gases and particulates, and is exhausted out of the stack. The stack was not used as part of the analysis done for estimating radioactive material release due to fuel failure. Consequently, there is no specification for minimum stack height.

Specification 5.1.1 defines the minimum confinement air volume. This volume was used in the calculations for estimating the concentrations of radioactive effluents that would be present in the confinement air due to a fuel failure.

Specification 5.1.2 defines the minimum time required to have elapsed after reactor shutdown in order to move the reactor room crane directly over the top of the high power section of the reactor pool. The maximum credible accident for the facility is a beam port shear, leading to a loss of coolant accident (LOCA). As part of the LOCA analysis, it was determined that if the reactor is shutdown after infinite operation for 4.5 hours, core power will have decayed to a level at which it is impossible for decay heat to damage the fuel. Thus, this specification makes it impossible for something dropped from the crane to shear a beam port, and lead to fuel overheating.

5.2 Reactor Coolant System Description

Description:

The reactor pool is made of concrete and has an aluminum liner. The primary coolant is light water that is provided by the local town water supply. One end of the pool is designated as the high power end of the pool because the primary inlet and outlet pipes extend into the pool at that end, allowing for forced convection cooling. The thickness of the pool wall is greater at that end than at the low power end of the pool. The central section of the pool is separated from the high power section by two pool wall extensions that protrude approximately two feet into the pool, opposite each other. This allows a pool dam to be put into place so that the high power section can be isolated from the rest of the pool, and drained without draining the rest of the pool. Likewise, there is a pair of pool wall extensions that separate the center section of the pool and the low power end of the pool, which allows the low power end to be drained without draining the rest of the pool.

The core is suspended in the pool from a moveable bridge, that allows the core to be positioned anywhere along the length of the pool, while being centered along the width. The core may be operated up to 100 kW at any position in the pool, however, for operations above 100 kW, the core must be fully seated at the high power end so that the forced convection pipes and ducts are coupled.

The primary inlet and outlet pipes extend into the pool at the high power end, approximately twelve feet below the pool surface, and couple to the inlet and outlet ducts, which are attached to the core suspension frame. Forced convection cooling is achieved by bringing cooled water from the primary inlet pipe down the inlet duct which opens over the top of the core. Suction causes the water to go through the core into a plenum beneath the core, up the outlet duct, and into the primary outlet pipe.

The forced convection cooling system outlet pipe goes from the reactor pool to the delay tank, where cooling water's progress through the cooling system is held up for approximately 70 seconds in order to reduce the Nitrogen - 16 concentration in the water. From the delay tank, the forced cooling system is divided into two loops.

Each cooling loop consists of a primary and secondary system. Each primary system takes heated water from the delay tank, through a primary pump, through the primary side of a heat exchanger, and back to the forced convection cooling system inlet piping, where the two systems merge before returning to the pool. The piping for the primary cooling system is aluminum. Nominal temperatures and pressures are less than 130 F and less than 100 psig respectively.

The secondary sides of the primary heat exchangers use city water to remove the heat from the primary sides. For each loop, secondary water from the heat exchanger is circulated to a cooling tower, through the secondary pump, and back to the heat exchanger. The piping for the secondary cooling system is polyvinyl chloride.

Both of the cooling towers use air cooling to reduce the temperature of the secondary water.

For 2 MW reactor operation, only one of the cooling loops is required to provide sufficient cooling, though it is possible to run both loops simultaneously.

Applicability:

These specifications apply to the RINSC reactor coolant system.

Objective:

The objective of these specifications is to ensure that there will be adequate reactor core cooling, as well as adequate radiation shielding at the pool top.

Specifications:

- 5.2.1 The reactor shall be seated in the high power end of the pool when the reactor is in operation at power levels greater than 100 kW.
- 5.2.2 The pool dam shall be in storage when the reactor is in operation.

Bases:

The reactor pool serves as the biological shield which protects personnel from exposure to direct radiation from the reactor. The pool water elevation above the top of the core is such that personnel may safely observe the operating core from the top of the pool.

Natural convection cooling has been shown to be a sufficient heat removal mechanism for reactor operation at power levels less than or equal to 100 kW. However, forced convection cooling is required for operation at power levels above 100 kW.

Specification 5.2.1 requires that the reactor be fully seated in the high power section of the pool for operation at power levels above 100 kW. This ensures that the cooling system inlet and outlet pipes and ducts are coupled so that the forced convection cooling system is functional.

Specification 5.2.2 requires that the pool dam be in storage when the reactor is in operation. This ensures that there is a sufficient pool water volume to dissipate the heat that is generated by the core. It also ensures that the full volume of water that was used to calculate the drain time in the event of a loss of coolant accident is available.

5.3 Reactor Fuel and Core Description

Description:

The RINSC fuel is MTR plate type fuel that has a nominal enrichment of 19.75% Uranium - 235. The chemical composition of the fuel is U_3Si_2 . Each fuel assembly consists of 22 fuel plates, bound by side plates that hold the fuel plates evenly spaced apart. At each end of the assembly, the side plates are attached to square end boxes, that are capable of being inserted into a core grid box. The cladding, side plates, and end boxes are aluminum. Each fresh fuel assembly is loaded with 275g Uranium - 235 nominal.

The core grid box is consists of a 5 15/16 inch thick grid plate that has a 9 X 7 array of square holes, and a box that has four walls that surround the grid plate in such a way that the plate serves as the bottom of the box with the top end open. The grid box is suspended from the top of the pool by four corner posts that occupy the corner grid spaces. The box is oriented so that the open end faces up toward the top of the pool. The reactor core is configured by inserting fuel element end boxes into grid spaces, so that each fuel assembly is standing up inside the box.

The standard core consists of 14 assemblies in a 3 X 5 array in the center of the grid box, with the central grid space available as an experimental facility. The remaining grid spaces are either filled with graphite or beryllium reflector assemblies, or incore experimental facilities. A non-standard core configuration with 17 fuel elements is also possible. In this configuration, the standard core configuration has been modified so that the three central reflector assemblies on the thermal column edge of the core are substituted with fuel assemblies. This core configuration is more conservative than the 14 element core because the core power is spread over three additional assemblies.

Both core configurations include 4 shim safety control blades, and a regulating rod. The shim safety blades are located between the fuel and the reflector assemblies on both of the edges of the fuel array that consist of 5 assemblies. There are two blades on each side of the fuel. The blades are housed in shrouds that are part of the core grid box. The shrouds ensure that the blades have unfettered movement in and out of the core. The regulating rod is positioned one grid space out from the fuel, along the central axis of the fuel on the thermal column side of the core.

Applicability:

These specifications apply to the RINSC reactor fuel and core.

Objective:

The objective of these specifications is to ensure that the fuel elements are fabricated, and that the core is configured in a manner that is consistent with the characteristics used in the safety analysis.

Specifications:

- 5.3.1 Each fuel element shall be fabricated with 22 plates per element.
- 5.3.2 The reactor shall not be operated unless there are a minimum of 14 fuel elements loaded in the core.
- 5.3.3 Each fuel element shall have a nominal Uranium - 235 loading of 275 grams.
- 5.3.4 The reactor shall not be operated unless all of the grid spaces are filled.
- 5.3.5 The core fuel arrangement shall be symmetrical.

Bases:

Specification 5.3.1 requires that each fuel element be fabricated with 22 fuel plates. Specification 5.3.2 requires that there be a minimum of 14 fuel elements in the core during operation. This combination of specifications ensures that the core has the minimum number of flow channels that were used in the thermal-hydraulic analysis.

Specification 5.3.3 requires that that each fuel element has a nominal U-235 loading of 275 grams. In the fuel failure analysis, the failure scenario considers an event in which the entire radioisotope inventory of one fuel plate is released into the pool. The inventory per plate is dependent on the fuel loading, and the number of fuel plates over which reactor power is distributed. This specification, in conjunction with specifications 5.3.1 and 5.3.2 ensures that the fuel plate inventory is consistent with the fuel failure analysis.

Specification 5.3.4 requires that all of the core grid spaces be filled during reactor operation. This specification ensures that coolant flow through the core does not bypass the flow channels between the fuel plates by flowing through a vacant grid space.

Specification 5.3.5 requires the core arrangement to be symmetrical. This specification ensures that the fuel is arranged in a manner that is consistent with the arrangements considered for the thermal-hydraulic and transient analyses, in which peak power and peak cladding temperature were predicted.

5.4 Fissionable Material Storage Description

Description:

Irradiated fuel is stored in two types of fuel storage racks in the reactor pool:

- Fixed racks that are mounted on the pool wall
- Moveable racks that rest on the pool floor

Each fixed rack has 9 spaces for fuel storage arranged in a linear array. Each moveable rack has 18 spaces for fuel storage arranged in a 9 X 2 array.

Non-irradiated fuel is typically stored in the RINSC fuel safe.

Non-fuel fissionable materials are either kept where they are in use, or are stored in the reactor pool or fuel safe depending on size constraints and what is most reasonable from an ALARA standpoint.

Applicability:

These specifications apply to the fissionable material storage facilities used for storing materials while they are not in use, or in an approved shipping container.

Objective:

The objective of these specifications is to ensure that it is impossible for fissionable material to achieve a critical configuration.

Specifications:

- 5.4.1. Fissionable material that is not in use or not in an approved shipping container shall be in storage.
- 5.4.2. Fissionable material storage facilities shall have $k_{eff} \leq 0.9$, for all conditions of moderation and reflection using light water.

Bases:

These specifications conform to ANSI 15.1. They ensure that fissionable material that is not in use will remain in a configuration that cannot achieve criticality.

6.0 Administrative Controls

6.1 Organization

6.1.1 Organization Structure

The Rhode Island Nuclear Science Center (RINSC) Reactor is 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 appoints five Commissioners to the Rhode Island Atomic Energy Commission (RIAEC) who have the authority to recommend the selection a Director, and appoint individuals to the Nuclear and Radiation Safety Committee (NRSC). The Director is the organizational head, and is responsible for the reactor facility license. The Assistant Director for Operations is responsible for the reactor programs and operation of the facility. The Assistant Director for Radiation and Reactor Safety is responsible for the safety programs of the facility. The RINSC staff operates and maintains the facility. The Nuclear and Radiation Safety Committee (NRSC) is an independent review and audit committee. Figure 6.1 shows the organization chart.

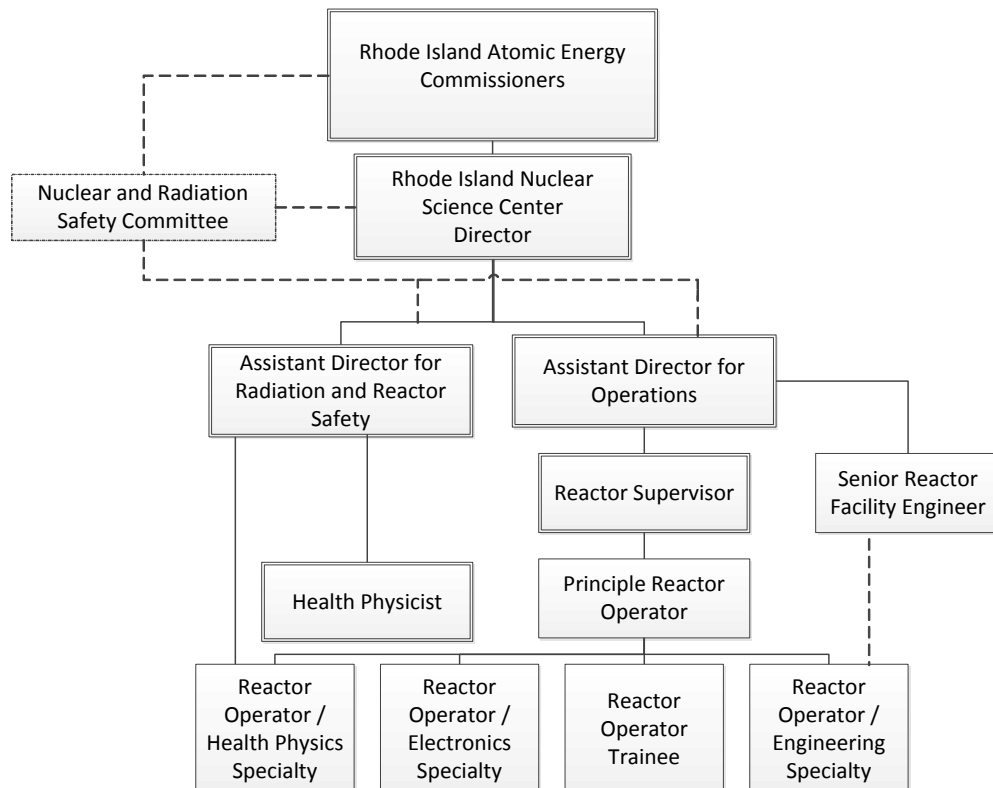


Figure 6.1: Rhode Island Atomic Energy Commission Organization Chart

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 has 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 is responsible for the license, and for developing and directing all of the administrative and technical programs. The Director is 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 is responsible for implementing the operations programs and managing the operation of the RINSC facility. The Assistant Director ensures 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 Reactor and Radiation Safety

The Assistant Director for Reactor and Radiation Safety is responsible for implementing and managing the Radiation Safety Program. The Assistant Director ensures 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 is 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 is responsible for directing the licensed activities of Reactor Operators. The Senior Reactor Operator ensures 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 NRSC. The Senior Reactor Operator also ensures that experimenters follow facility procedures.

6.1.2.7 Reactor Operators

The Reactor Operator on duty during reactor operations is responsible for manipulating the controls of the reactor. The Reactor Operator directs the actions of Reactor Operator Trainees, and ensures 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 but all of the shim safety control rods are fully inserted into the core shall be a Reactor Operator in the control room or at the pool top.

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 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 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 significant reduction in power, and

6.1.3.2.4 Recovery from an unscheduled shutdown.

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 Reactor and Radiation 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 55.

6.1.4.1.7 Reactor Operators

Reactor Operators shall be licensed pursuant to 10 CFR 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 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 Reactor and Radiation 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

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 Proposed changes in reactor safety related instrumentation or systems that have safety significance,
- 6.2.3.7 10 CFR 50.59 evaluations,
- 6.2.3.8 New procedures,
- 6.2.3.9 Major changes to procedures that have safety significance,
- 6.2.3.10 Violations of procedures that have safety significance,
- 6.2.3.11 New experiments,
- 6.2.3.12 Operating abnormalities that have a safety significance, and
- 6.2.3.13 Reportable occurrences.

6.2.4 Audit Function

The non-RIAEC staff members of the NRSC shall audit the following items:

- 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.

6.3 Radiation Safety

The facility shall have a qualified, designated individual that is responsible for implementing the Radiation Safety Program. The Assistant Director for Reactor and Radiation Safety is 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 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:

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 control,

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 If the reactor was in operation while a limiting condition for operation was not met, 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 If the reactor is shutdown, 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 the following information:

- 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 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 visitors.

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 at the number listed in 10 CFR 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 50.4.

6.7.2.2 Reporting Requirements for Unusual Events

Within 30 days following an unusual event, 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 50.4.

6.8 Records

6.8.1 Records to 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 to be retained for a period of at least one certification cycle

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 to 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.