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Mr. Patrick Boyle, Project Manager
Research and Test Reactors Licensing Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D.C. 20555-0001

June 30, 2014

Re: Updated Proposed Technical Specifications
Docket No. 50-193

Dear Mr. Boyle:

In response to our telephone conversation on December 3, 2013, and the follow up questions related to that conversation regarding the Rhode Island Nuclear Science Center Reactor License Renewal, enclosure one is forwarded containing the updated proposed Technical Specifications with the appropriate changes made. Enclosure two is the set of follow up questions and responses. Enclosure three is the Fissionable Experiment RAM Release Analysis that was used for setting a limit on the quantity of fissionable material that should be allowed in a reactor experiment.

If there is anything else that you need in order to proceed forward with this project, please do not hesitate to contact me.

Very truly yours,

Michael J. Davis

Michael J. Davis
Assistant Director
Rhode Island Atomic Energy Commission

I certify under penalty of perjury that the representations made above are true and correct.

Executed on: 6/30/14 By: Michael J. Davis

Docket No. 50-193

Enclosures: as stated

A020
NRR

Proposed Technical Specifications 140627

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

1.~~7~~ 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.~~8~~ 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.~~9~~ 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.

1.1~~0~~ 2 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.

1.1~~1~~ 3 Explosive Material

Explosive material is any solid or liquid which is either:

A. 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
B. 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.

1.1~~2~~ 4 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.

1.1~~3~~ 5 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.

1.1~~4~~ 6 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.

1.1~~5~~ 7 May

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

1.1~~6~~ 8 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.

1.1~~7~~ 9 Moveable Experiment

A moveable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

1.~~18~~ 20 Operable

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

1.~~19~~ 21 Operating

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

1.2~~0~~ 2 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.

1.2~~1~~ 3 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:

A. Insertion or removal from the core,
B. Intended or anticipated changes in position, or
C. Credible malfunctions that alter experiment position or configuration.

1.2~~2~~ 4 Reactor Operating

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

1.2~~3~~ 5 Reactor Operator

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

1.2~~4~~ 6 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.

1.2~~5~~ 7 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.

1.2~~6~~ 8 Reactor Secured

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

A. There is insufficient moderator available in the reactor to attain criticality,

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

C. The following conditions exist:

   1. The reactor is shutdown,
   2. The master switch is in the off position and the key is removed from the lock,
   3. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and
   4. No experiments are being moved or serviced that have a reactivity worth of greater than 0.6% dK/K when moved.

1.2~~7~~ 9 Reactor Shutdown

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

1.~~28~~ 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.~~29~~ 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 % dK/K.

1.3~~0~~ 2 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.

1.3~~1~~ 3 Reportable Occurrence

A reportable occurrence is any of the following:

1. A violation of a safety limit,

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,

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

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

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,

6. An uncontrolled or unanticipated change in reactivity in excess of 0.75 %ΔK/K,

7. Abnormal and significant degradation of the fuel cladding,

8. Abnormal and significant degradation of the primary coolant boundary, or

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

Areas, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

1.3~~2~~ 5 Safety Channel

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

1.3~~3~~ 6 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.

1.3~~4~~ 7 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.

1.3~~5~~ 9 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.~~36~~ 40 Shall

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

1.~~37~~ 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.~~38~~ 42 Should

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

1.~~39~~ 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.

1.4~~0~~ 5 Subcritical

The reactor is subcritical if:

~~A. There is insufficient fissile material or moderator present in the core to attain criticality under optimum available conditions of moderation and reflection, or~~

~~B~~A. The control rods are providing sufficient negative reactivity in the core to prevent criticality.

1.4~~1~~ 6 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, ~~and ensuring the safe operation of the reactor~~ which provides confidence that these components will perform their intended functions.

1.4~~2~~ 7 Surveillance Intervals

~~Surveillance intervals are the periods in which each surveillance activity is to be performed. Each interval is defined in a way that provides operational flexibility. The surveillance intervals are:~~

~~A. Daily shall mean during the calendar day on days when the facility is open,~~
~~B. Weekly shall mean an interval not to exceed 10 calendar days,~~
~~C. Monthly shall mean an interval not to exceed 6 weeks,~~
~~D. Quarterly shall mean an interval not to exceed 4 months,~~
~~E. Semiannually shall mean an interval not to exceed 7 1/2 months,~~
~~F. Annually shall mean an interval not to exceed 15 months, and~~
~~G. Biennially shall mean an interval not to exceed 2 1/2 years.~~

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. 5 years (interval not to exceed 6 years).
2. 2 years (interval not to exceed 2 1/2 years).
3. Annual (interval not to exceed 15 months).
4. Semiannual (interval not to exceed 7 1/2 months).
5. Quarterly (interval not to exceed 4 months).
6. Monthly (interval not to exceed 6 weeks).
7. Weekly (interval not to exceed 10 days).
8. Daily (must be done during the calendar day).

1.~~43~~ 8 True Value

The true value is the actual value of a parameter.

1.~~44~~ 9 Unscheduled Shutdown

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

A. Actuation of the reactor safety system,
B. Operator error,
C. Equipment malfunction, or
D. Manual shutdown in response to conditions that could adversely affect safe operation.

~~1.45 Unusual Event~~

~~An unusual event is defined as any of the following:~~

~~A. Permanent changes in the facility organization involving RIAEC membership, the facility Director, or either of the facility Assistant Directors.~~
~~B. Significant changes in the transient or accident analyses as described in the Safety Analysis Report.~~

2.0 Safety Limits and Limiting Safety System Settings

2.1 Safety Limits

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.

Specifications:

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

Bases:

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:

See Comment

Comment [D1]: We are seeing whether or not we can change these valuse slightly.

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 ft 9.6 inches.

2.2.1.3 The limiting safety system setting for the bulk pool temperature shall be 125 ^{O}F.

Bases:

This combination of specifications was set to prevent the cladding temperature from approaching the 530 ^{O}C value at which damage to the fuel cladding could occur, ~~even~~ 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 steady state power level is less than 200 kW,
- The coolant height above the fuel meat is at least 23 ft 6.5 in, and
- The bulk pool temperature is no greater than 130 ^{O}F.

Under these conditions for coolant height and bulk pool temperature, peak channel power would have to reach 1.7812 kW in order for the onset of nucleate boiling to occur, which corresponds to a fuel cladding temperature that is below the 530 ^{O}C value at which damage to the fuel cladding could occur. The hottest channel reaches a peak power of 1.7812 kW when core power is 369 kW. Consequently, there is a margin of

369 kW – 200 kW = 169 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 67.5 ^{O}C during a transient in which the following combination of limits are in place:

- The initial power level is no greater than 100 kW,
- The coolant height above the fuel meat is at least 23 ft 9.1in,
- The bulk pool temperature is no greater than 128 ^{O}F, and
- The transient is terminated by an over power trip at 125 kW.

~~In both, the steady state and transient analyses, the predicted peak cladding temperature is significantly below the damage threshold temperature of 530 °C. The safety margins are :~~

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

~~- Margin for the transient bounded by the limiting conditions is~~ 530 ^{O}C - 67.5 ^{O}C = 462.5 ^{O}C.

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

- Power Level	± 10 kW
- Coolant Height	0.5 in
- Temperature	3 ^{O}F

Consequently, the bases for these specifications are:

Specification 2.2.1.1 sets the limiting safety system setting for reactor thermal power to be 115 kW. The analyses show that cladding damage will not occur under any condition if initial power is no greater than 200 kW. Taking into consideration a 10 kW measurement error, if the LSSS is 115 kW, then the Limiting Trip Value could be as high as

125 kW, which still leaves a ~~safety~~ margin of 75 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 ft 9.6 in. The analyses show that cladding damage will not occur under any condition if the height is no less than 23 ft 6.5 in. Taking into consideration a 0.5 in measurement error, if the LSSS is 23 ft 9.6 in, then the Limiting Trip Value could be as low as 23 ft 9.1 in, which still leaves a ~~safety~~ margin of 2.6 in between the LSSS and the most conservative true value of the coolant height used in the analysis.

Specification 2.2.1.3 sets the limiting safety system setting for the bulk pool temperature to be 125 ^{O}F. The analyses show that cladding damage will not occur under any condition if the pool temperature is no greater than 130 ^{O}F. Taking into consideration a 3 ^{O}F in measurement error, if the LSSS is 125 ^{O}F, then the Limiting Trip Value could be as high as 128 ^{O}F, which still leaves a ~~safety~~ margin of 2 ^{O}F between the LSSS and the most conservative true value of the bulk pool temperature.

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:

See Comment

Comment [D2]: We are seeing whether or not we can change these values slightly

2.2.2.1 The limiting safety system setting for reactor thermal power shall be 2.1 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 ft 9.6 inches.

2.2.2.3 The limiting safety system setting for the primary coolant outlet temperature shall be 120 ^{O}F.

2.2.2.4 The limiting safety system setting for the primary coolant flow rate shall be 1800 gpm.

Bases:

This combination of specifications was set to prevent the cladding temperature from approaching the 530 ^{O}C value at which damage to the fuel cladding could occur, ~~even~~ 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 ft 6.5 in,
- The primary coolant outlet temperature is no greater than 125 ^{O}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 85 ^{O}C during a transient in which the following combination of limits are in place:

- The initial power level is no greater than 2.2 MW,
- The coolant height above the fuel meat is at least 23 ft 9.1in,

- The primary coolant inlet temperature is no greater than 123 ^{o}F,
- The coolant flow rate through the core is a least 1740 gpm, and
- The transient is terminated by an over power trip at 2.3 MW.

~~In both, the steady state and transient analyses, the predicted peak cladding temperature is significantly below the damage threshold temperature of 530 ^{o}C. The safety margins are:~~

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

~~- Margin for the transient bounded by the limiting conditions is~~ 530 ^{o}C - 85 ^{o}C = 445 ^{o}C.

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

- Power Level	± 0.2 MW
- Coolant Height	0.5 in
- Temperature	3 ^{o}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 2.1 MW. The analyses show that cladding damage will not occur under any condition if initial power is no greater than 2.4 MW. Taking into consideration a 0.2 MW measurement error, if the LSSS is 2.1 MW, then the Limiting Trip Value could be as high as 2.3 MW, which still leaves a ~~safety~~ margin of 0.1 MW between the LSSS and the most conservative true value of the power level used in the analysis.

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 ft 9.6 inches. The analyses show that cladding damage will not occur under any condition if the height is no less than 23 ft 6.5 in. Taking into consideration a 0.5 in measurement error, if the LSSS is 23 ft 9.6 in, then the Limiting Trip Value could be as low as 23 ft 9.1 in, which still leaves a ~~safety~~ margin of 2.6 in between the LSSS and the most conservative true value of the coolant height used in the analysis.

Specification 2.2.2.3 sets the limiting safety system setting for the primary coolant outlet temperature to be 120 °F. The analyses show that cladding damage will not occur under any condition if the primary coolant outlet temperature is no greater than 125 °F. Taking into consideration a 3 °F in measurement error, if the LSSS is 120 °F, then the Limiting Trip Value could be as high as 123 °F, which still leaves a ~~safety~~ margin of 2 °F between the LSSS and the most conservative true value of the primary coolant outlet temperature.

Specification 2.2.2.4 sets the limiting safety system setting for the primary coolant flow rate to be 1800 gpm. The analyses show that cladding damage will not occur under any condition if the primary coolant flow rate is at least 1580 gpm. Taking into consideration a 60 gpm in measurement error, if the LSSS is 1800 gpm, then the Limiting Trip Value could be as low as 1740 gpm, which still leaves a ~~safety~~ margin of 160 gpm between the LSSS and the most conservative true value of the primary coolant flow rate.

3.0 Limiting Conditions for Operation

3.1 Core Parameters

3.1.1 Reactivity Limits

Applicability:

This specification applies 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 this specification 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.

Specification:

3.1.1.1 Core

3.1.1.1.1 The core shutdown margin shall be at least 1.0 % dK/K.

3.1.1.1.2 The core excess reactivity shall not exceed 4.7 % dK/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 %dK/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 % dK/K.

3.1.1.3 Experiments

See comment

Comment [D3]: We are still trying to finalize a method for determining the reactivity worths of unknowns.

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:

Total Moveable and Fixed	0.6 %dK/K
Total Moveable	0.08 %dK/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:

Fixed	0.6 % dK.K
Moveable	0.08 % dK/K

Basis:

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 % dK/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, 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:

This specification applies to core configurations during operations above 0.1 MW when the reactor is in the forced convection cooling mode.

Objective:

The objective of this specification 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:

This specification applies to the reactor safety system and ~~safety related~~ instrumentation required for ~~critical operation of the~~ reactor operation.

Objective:

The objective of this specification is to define the minimum set of safety system and ~~safety related~~ instrumentation channels that must be operable in order for ~~the~~ reactor ~~to be made critical~~ operation.

Specification:

3.2.1 The reactor shall not be ~~made critical~~ 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% dK/K per second.

3.2.1.3 The ~~following reactor safety and safety related~~ 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 by~~	~~Power Level~~	~~Less than or Equal to~~	~~105% of Licensed Power~~
~~Low Pool Level~~	~~All~~	~~1~~	~~Scram by~~	~~Pool Level Drop~~	~~Less than or Equal to~~	~~23 ft 9.6 in~~
~~Primary Coolant Inlet Temperature~~	~~>100kW~~	~~1~~	~~Alarm by~~	~~Inlet Temp~~	~~Less than or Equal to~~	~~111 F~~
~~Primary Coolant Outlet Temperature~~	~~>100kW~~	~~1~~	~~Alarm by~~	~~Outlet Temp~~	~~Less than or Equal to~~	~~117 F~~
	~~>100kW~~	~~1~~	~~Scram by~~	~~Outlet Temp~~	~~Less than or Equal to~~	~~120 F~~
~~Pool Temperature~~	~~<=100kW~~	~~1~~	~~Scram by~~	~~Pool Temp~~	~~Less than or Equal to~~	~~125 F~~
~~Primary Coolant Flow Rate~~	~~>100kW~~	~~1~~	~~Scram by~~	~~Primary Flow Rate~~	~~Less than or Equal to~~	~~1800 gpm~~
~~Rate of Change of Power~~	~~All~~	~~1~~	~~Scram by~~	~~Period~~	~~Less than or Equal to~~	4 ~~seconds~~
~~Seismic Disturbance~~	~~All~~	~~1~~	~~Scram if~~	~~Seismic Disturbance Detected~~		
~~Bridge Low Power Position~~	~~>100kW~~	~~1~~	~~Scram if~~	~~Bridge Not Seated at HP End~~		
~~Bridge Movement~~	~~All~~	~~1~~	~~Scram if~~	~~Bridge Movement Detected~~		
~~Coolant Gates Open~~	~~>100kW~~	~~1~~	~~Scram if~~	~~Inlet Gate Open~~		
	~~>100kW~~	~~1~~	~~Scram if~~	~~Outlet Gate Open~~		
~~Detector HV Failure~~	~~All~~	~~1~~	~~Scram if~~	~~Detector HV Decrease~~	~~Less than or Equal to~~	~~50 V~~
	~~All~~	~~1~~	~~Scram if~~	~~Detector HV Decrease~~	~~Less than or Equal to~~	~~50 V~~
	~~All~~	~~1~~	~~Scram if~~	~~Detector HV Decrease~~	~~Less than or Equal to~~	~~50 V~~
~~No Flow Thermal Column~~	~~>100kW~~	~~1~~	~~Scram by~~	~~No Flow Detected~~		
~~Manual Scram~~	~~All~~	~~1~~	~~Scram by~~	~~Button Depressed~~		

	~~All~~	~~1~~	~~Scram by~~	~~Button Depressed~~		
~~Servo Control Interlock~~	~~All~~	~~1~~	~~No Automatic Servo if~~	~~Regulating Blade not Full Out~~		
	~~All~~	~~1~~	~~No Automatic Servo if~~	~~Period~~	~~Less than~~	~~30 seconds~~
~~Shim Safety Withdrawal~~	~~All~~	~~1~~	~~No SS Withdrawal if~~	~~Count Rate~~	~~Less than~~	~~3 cps~~
	~~All~~	~~1~~	~~No SS Withdrawal if~~	~~Test / Select SW not Off~~		
~~Rod Control Communication~~	~~All~~	~~1~~	~~Scram if~~	~~Loss of Communication~~	~~Less than or Equal to~~	~~10 seconds~~

See Comment

Comment [D4]: The order of the trips have been changed, but for each trip, new verbiage is in red, and items that I am proposing that we delete are being crossed out.

Protection	Power Level	Channels Required	Function	Set Point
Over Power	All	2	Scram before power is greater than	105% 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 ft 9.6 in above the top of the fuel meat
Manual Scram	All	1	Scram when	Control Room Scram Button Depressed
	~~All~~	~~1~~	~~Scram when~~	~~Bridge~~

				~~Scram Button Depressed~~
Servo Control Interlock	All	1	No regulating rod automatic servo if	Regulating ~~blade~~ 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 count 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	125 F
~~Primary Coolant Inlet Temperature~~	~~>100kW~~	~~1~~	~~Alarm before temperature is greater than~~	~~111 F~~
~~Primary Coolant Outlet Temperature~~	~~>100kW~~	~~1~~	~~Alarm before temperature is greater than~~	~~117 F~~
Primary Coolant Outlet Temperature	>100kW	1	Scram before temperature is greater than	120 F
Primary Coolant Flow Rate	>100kW	1	Scram before flow rate is less than	1800 gpm
Coolant Gates Open	>100kW	1	Scram when	Inlet or outlet gate open
	~~>100kW~~	~~1~~	~~Scram when~~	~~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

See Comment

Comment [D5]: The order of the bases habe been changed to match the order in which the spcification shows up in the table.

Basis:

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% dK/K per second. As part of the Safety Analysis, Argonne National Laboratory analyzed ramp insertions of 0.02% dK/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 ~~safety and safety related~~ instrumentation that is required to be operable when the reactor is operated.

Two independent power level channels are required for both ~~forced and~~ natural and forced convection cooling modes of operation, each of which must be capable of scramming the reactor by 105% licensed power. The basis section of Specification 2.2.1.1 shows that this ensures that the power level ~~safety limit of 2.4 MW~~ 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 ~~safety limit of 23 ft 6.5 inches above the top of the core~~ levels that were used for the steady state and transient analyses. The forced convection steady state power analysis assumed a minimum pool height of 23 ft 6.5 in above the top of the fuel meat. The forced convection transient analyses assumed a minimum pool height of 23 ft 9.1 in above the top of the fuel meat. The low pool level channel LSSS is 23 ft 9.6 in 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 assumed heights during reactor operation.

~~Two~~ One manual scram button~~s~~ that is located in the control room ~~are~~ is required to be operational during both modes of operation. ~~One manual scram button is located in the control room, which provides the operator with a mechanism for manually scramming the reactor. The second scram button is on the reactor bridge, which provides anyone directly over the core with a mechanism for scramming the reactor if there were a reason to do so.~~

One servo control interlock that prevents the regulating ~~blade~~ rod from being put into automatic servo mode unless the blade is fully withdrawn is required for both modes of operation. As a result of this interlock, when the regulating blade 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 ~~blade~~ 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 scramming the reactor when the temperature reaches 125 F. The basis section of Specification 2.2.~~2~~1.3 shows that this ensures that the pool temperature ~~safety limit of 130 F~~ will not ~~be~~ exceede~~d~~ 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 coolant temperature channel is required for forced convection cooling mode operation. This channel alerts the operator in the event that the inlet temperature reaches 111 F. The steady state thermal hydraulic analysis that was done by Argonne National Laboratory for forced convection flow predicts that the inlet temperature would be 115 F for operation at 2.4 MW, with a primary flow of 1580 gpm and an outlet temperature of 125 F.~~

One primary outlet temperature channel is required for forced convection cooling mode operation. This channel is capable of scramming the reactor when the temperature reaches 120 F. The basis section of Specification 2.2.~~1~~ 2.3 shows that this LSSS will ensure~~s~~ that the coolant outlet temperature ~~safety limit~~ will not exceed the ~~of~~ 125 F ~~will not be exceeded~~ 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 ~~safety limit of~~ 1580 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.~~1~~ 2.4 shows that if this channel is set to scram at a limiting safety system setting of 1800 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.

Specifications:

The primary coolant conductivity shall be ≤ 2 μmho / cm ~~when averaged over a quarter of a year~~.

Bases:

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:

Cs-137 and I-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 Cs-137 or I-131. These isotopes are prominent fission products. Consequently, if either of these isotopes are 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.

Specifications:

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

Bases:

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 Na-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:

This specification applies to the operations for which the components of the confinement system must be operable.

Objective:

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.

Specification:

1. The confinement system shall be operable whenever:

    1. The reactor is operating.
    2. Irradiated fuel handling is in progress.

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.
4. Any work on the core or control rods that could cause a reactivity change of more than 0.6~~5~~0% dK/K is in progress.
5. Any experiment movement that could cause a reactivity change of more than 0.6~~5~~0% dK/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:

1. Damage during fuel handling operations.
2. 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:

This specification describes the components of the confinement system that are necessary in order for the system to perform its intended function under normal operating conditions.

Objective:

To assure that the confinement system is capable of detecting a release of airborne radioactive material.

Specification:

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

    1. Normal Personnel Access Door
    2. Roll Up Door
    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:

This specification describes the components of the confinement system that are necessary in order for the system to perform its intended function under emergency operating conditions.

Objective:

To assure that the confinement system is capable of mitigating the consequences of a possible release of airborne radioactive material.

Specification:

1. The following emergency confinement system components shall be operable:

    1. Emergency Confinement System Buttons
    2. Confinement Air Intake Damper
    3. Confinement Air Exhaust Damper
    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:

This specification describes the conditions necessary to assure that normal operating mode confinement is achieved.

Objective:

To assure that the confinement system is functioning sufficiently to prevent airflow from inside confinement to the environment through an uncontrolled pathway.

Specification:

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

1. The Normal Personnel Access Door is closed, except for entry and exit.
2. The Roll Up Door is closed.
3. The Roof Hatch is closed.
4. The Emergency Personnel Access Door is closed, except for entry and exit.
5. The Confinement Dampers are Open.
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:

This specification describes the conditions necessary to assure that emergency operating mode confinement is achieved.

Objective:

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.

Specification:

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

1. The Normal Personnel Access Door is closed, except for entry and exit.
2. The Roll Up Door is closed.
3. The Roof Hatch is closed.
4. The Emergency Personnel Access Door is closed, except for entry and exit.
5. The Confinement Dampers are Closed.
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:

To assure that the normal mode confinement system is capable of performing its intended function.

Specification:

1. The following normal mode ventilation system components shall be operating:

    1. Confinement Exhaust Blower
    2. Confinement Exhaust Filter System, which shall include:

        1. Roughing Filter
        2. Absolute Filter

    3. Confinement Exhaust Stack

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

Bases:

The Confinement Exhaust Blower produces a differential pressure across confinement to ensure that all confinement air pathways are through controlled pathways. The Confinement Exhaust Filter System ensures that the majority of the radioactive particulates that would be likely to be released in the event of a fuel failure would be filtered out prior to being released to the environment, until the emergency operating mode ventilation system is activated. The Confinement Exhaust 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.

3.5.2 Ventilation System Components Required for Emergency Operating Mode

Applicability:

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

Objective:

To assure that the emergency mode confinement system is capable of performing its intended function.

Specification:

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

    1. Emergency Blower
    2. Emergency Filter System, which shall include:

        1. Emergency Filter Intake System Roughing Filter
        2. Emergency Filter System Intake Absolute Filter

3. Emergency Filter System Charcoal Filter
4. Emergency Filter System Exhaust Absolute Filter

3. Dilution Blower
4. Confinement Exhaust Stack

2. The exhaust rate through the emergency filter shall be less than or equal to 1500 cfm.
3. The emergency filter shall be at least 99% efficient at removing iodine.

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 though 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. The Confinement Exhaust 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:

To assure that the confinement system is able to perform its intended function even, when normal electrical power is unavailable.

Specification:

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

Bases:

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:

This specification describes the confinement system components that are required to be connected to an emergency electrical power source.

Objective:

To assure that the confinement system is able to perform its intended function even, when normal electrical power is unavailable.

Specification:

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

    1. Emergency Blower
    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:

This specification applies to the radiation monitoring systems required for critical operation of the reactor, and fuel handling activities.

Objective:

The objective of this specification 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. The following air radiation monitoring instrumentation shall be operable whenever:

The reactor is operating,

Irradiated fuel handling is in progress,

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,

Any work on the core or control rods that could cause a reactivity change of more than 0.6~~5~~0% dK/K is in progress, or

Any experiment movement that could cause a reactivity change of more than 0.6~~5~~0% dK/K is in progress:

3.7.1.1.1 A minimum of one radiation monitor that is capable of warning personnel of high radiation levels in the confinement gaseous or particulate effluent shall be operating.

3.7.1.1.2 If this detector 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.

3.7.1.2 The following fission product radiation monitoring instrumentation shall be whenever:

The reactor is operating,

Irradiated fuel handling is in progress,

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, Any work on the core or control rods that could cause a reactivity change of more than 0.6~~5~~0% dK/K is in progress, or

Any experiment movement that could cause a reactivity change of more than 0.6~~5~~0% dK/K is in progress:

3.7.1.2.1 A minimum of one gamma sensitive radiation monitor that is capable of warning personnel of high radiation levels shall be over the pool.

3.7.1.2.2 If this detector 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.

3.7.1.2.3 If failure of the detector occurs during operations than the replacement detector must be put in place within 15 minutes of the recognition of failure.

3.7.1.3 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.

3.7.1.4 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 3.7.1.2.1 identifies the fission product monitoring instrumentation that is required to be operable when the reactor is operated, and during fuel handling operations.
>
> Specification 3.7.1.2.2 allows for the fission product monitoring instrumentation to be replaced in the event that the normal instrument fails.

Specification 3.7.1.3 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 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 3.7.1.4 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:

This specification applies to the radiation monitoring systems required for critical operation of the reactor, and fuel handling activities.

Objective:

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

Specifications:

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

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

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

3.7.2.1.4 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:

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.

Specifications:

~~Airborne effluents shall be monitored by an air monitor installed, calibrated and maintained in accordance with ANSI 13.1.~~ 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.

Bases:

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

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.

See Comment

Comment [D6]: This specification has been moved to section 4.7.2.2

3.7.2.2 Liquid Effluents

~~Applicability:~~

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

~~Objective:~~

~~The objective 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.~~

~~Specifications:~~

~~The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined before release.~~

~~Bases:~~

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

Applicability:

This specification applies to liquid effluent discharges.

Objective:

The objective 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:

This specification describes 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 this specification is to prevent damage to the reactor, reactor pool, and reactor experimental facilities.

Specification:

1. Corrosives Materials

    1. Corrosive materials shall be doubly contained in corrosion resistant containers.

2. Highly Water Reactive Materials

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

3. Explosive Materials

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

4. Fissionable Materials

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

    2. The maximum quantity of fissionable materials used in an experiment shall be no greater than 96.25 milligrams.

    ~~2~~3. Fissionable materials shall be doubly encapsulated.

    ~~3~~4. Containers for experiments that have fissionable material shall be opened inside confinement.

Basis:

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. ~~The consequence of experiment failure on criticality is bounded by limiting the reactivity worths of experiments. The analysis for this is in SAR Chapter 13 as part of the transient analysis. The radioactive material release is bounded by the analysis in SAR Chapter 13 for the Maximum Hypothetical Accident involving a fuel element failure. Double encapsulation of fissionable materials reduces the probability of the release of radioactive material. The requirement that experiments containing fissionable materials be opened inside confinement ensures that in the event of a fission product gas release, the mitigating actions of the confinement system would be available~~.

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:

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 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.

Specification:

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.

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. Experiment design shall be reviewed to ensure that credible failure of any experiment will not contribute to the failure of:

    1. Other Experiments
    2. Core Components
    3. Principle physical barriers to uncontrolled release of radioactivity

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 % dK/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:

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/cm^2.

Basis:

This limit is based on an analysis that was done by the University of Missouri Research Reactor (MURR). In their analysis, they note that the HFIR Reactor has noticed the presence of small cracks at fast fluences of $1.8 \text{ X } 10^{22}$ nvt, and suggest that "a value of $1 \text{ X } 10^{22}$ 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 $1 \text{ X } 10^{22}$ 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:

To prevent operation with damaged fuel elements.

Specifications:

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

Bases:

The RINSC inspects and tests each fuel element for reactor core box fit in accordance with written procedures to assure operation with fuel elements that are not damaged and meet specifications.

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:

These specifications 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 in^2 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 in^2 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 in^2 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 though this experimental facility is prevented.

Specification 3.9.3.1.3:

~~If the end(s) of the through port that will be used for access have gate valves mounted to them, then in the event of a leak, the port can be easily isolated so that the leak is stopped~~. 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 though the through port reasonably quickly, and the gate valves are in place, the consequence of the leak can be mitigated quickly by closing the valves.

Specification 3.9.3.1.4:

~~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 though the through port reasonably quickly, and the gate valves are in place, the consequence of the leak can be mitigated quickly by closing the valves~~. 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:

These specifications 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 in^2 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 in^2. Consequently, maintaining the limit on the area open between confinement and the beam ports to 1.25 in^2 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:

This specification applies to the surveillance requirements for reactivity limits.

Objective:

The objective of this specification is to ensure that reactivity limits are not exceeded.

Specification:

4.1.1.1 Core Reactivity Limit Surveillance

4.1.1.1.1 The core shutdown margin shall be determined:

Annually

Whenever the core reflection is changed

Whenever the core fuel loading is changed

Following control blade changes.

4.1.1.1.2 The core excess reactivity shall be determined:

Annually

Whenever the core reflection is changed

Whenever the core fuel loading is changed

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:

Annually

Whenever the core reflection is changed

Whenever the core fuel loading is changed

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:

Annually

Whenever the core reflection is changed

Whenever the core fuel loading is changed

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.

Basis:

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. ~~As part of the Safety Analysis, Argonne National Laboratory determined that for the equilibrium core, the temperature and void coefficients are negative over a temperature range of 20 C to 100 C. The fuel temperature coefficient was determined to be negative over a temperature range of 20 C to 600 C.~~

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:

This specification applies to the safety and safety related instrumentation.

Objective:

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

Specification:

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 alarm
4.2.5.4.2 Primary outlet temperature alarm
4.2.5.4.3 Primary outlet temperature scram
4.2.5.4.4 Pool temperature alarm
4.2.5.4.5 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 Primary outlet temperature channel
4.2.6.5 Pool temperature channel

Basis:

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 alarms, 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, t~~T~~he 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 3.1.1.1 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 Cs-137 and I-131 activity in the primary coolant is not significantly above background.

Specifications:

Cs-137 and I-131 activity in the primary coolant shall be measured annually.

Basis:

Specification 4.3.1.2 requires that the Cs-137 and I-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 greater 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.

Specifications:

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 Na-24 activity in the secondary coolant is not significantly above background.

Specification:

Na-24 activity in the secondary coolant shall be measured annually.

Basis:

Specification 4.3.2.1 requires that the Na-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:

4.4.1.1. 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.

Bases:

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:

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

Objective:

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

Specification:

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:

4.5.1.1. 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:

This specification describes the surveillance requirements for the emergency power system.

Objective:

The objective of this specification is to verify that the emergency power system is operable and functional.

Specification:

4.6.1. An operability test to verify that the emergency power system starts in the event of a facility power outage shall be performed quarterly.

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 ensures 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:

This specification applies to the radiation monitoring systems that are required to be operable during reactor operation and fuel handling activities.

Objective:

The objective of this specification 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:

~~4.2.7.1~~ 4.7.1.2.1 The experimental level area radiation monitor
~~4.2.7.2~~ 4.7.1.2.2 The pool top area radiation monitor
~~4.2.7.3~~ 4.7.1.2.3 The gaseous effluent air monitor
~~4.2.7.4~~ 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:

These specifications apply to the surveillance of the monitoring equipment used to measure the radioactivity in effluents.

Objective:

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

Specifications:

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

Bases:

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 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.

Specifications:

The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined before release.

Bases:

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:

4.8.1 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:

This specification applies to the surveillance of the standard and plug type beryllium reflectors.

Objective:

To prevent physical damage to the beryllium reflectors in the core from accumulated neutron flux exposure.

Specification:

4.9.1.1. The maximum neutron fluence of any beryllium reflector shall be:

4.9.1.1.1. The fluence 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,

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:

This specification applies to the surveillance of the LEU fuel elements.

Objective:

To verify the physical condition of the fuel elements in order to prevent operation with damaged fuel elements.

Specification:

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,
- 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 4.3.3 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 U~~u~~niversity 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.

Specification:

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 pool 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 N-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 2MW 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 equal to, or greater than 100 kW.

5.2.2 The pool dam shall be in storage when the reactor is in operation.

Basis:

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.

5.3 Reactor Fuel and Core Description

Description:

The RINSC fuel is MTR plate type fuel that has a nominal enrichment of 19.75% U-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 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 U-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.

Specification:

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 U-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.

Basis:

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 5.3.1 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:

This specification applies 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 this specification is to ensure that it is impossible for fissionable material to achieve a critical configuration.

Specification:

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 keff <= 0.9, for all conditions of moderation and reflection using light water.

Bases:

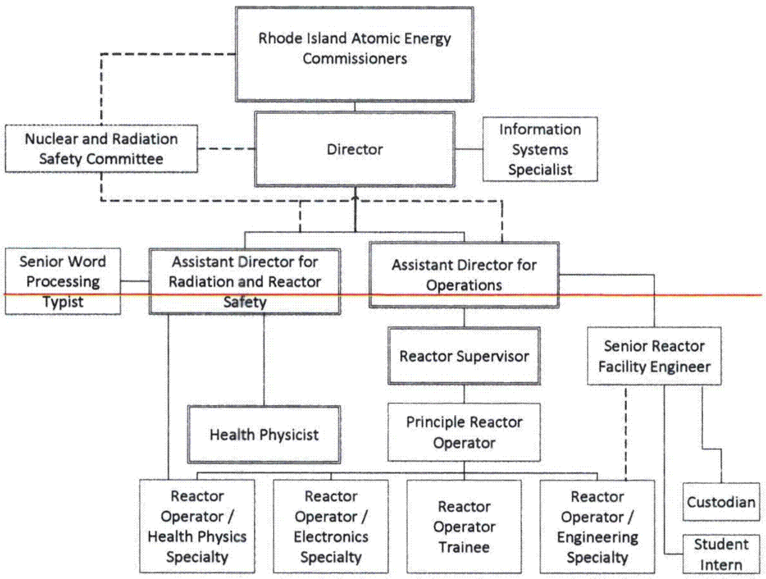
These specifications conform to ANSI 15.1. They ensure that fissionable material that is not in use will remain in a configuration that cannot achieve criticality.

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 select a Director, and appoint individuals to the Nuclear and Radiation Safety Committee (NRSC). The Director is the organization~~al~~ 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 1 shows the organization chart.



~~Figure 1 Organization Chart~~

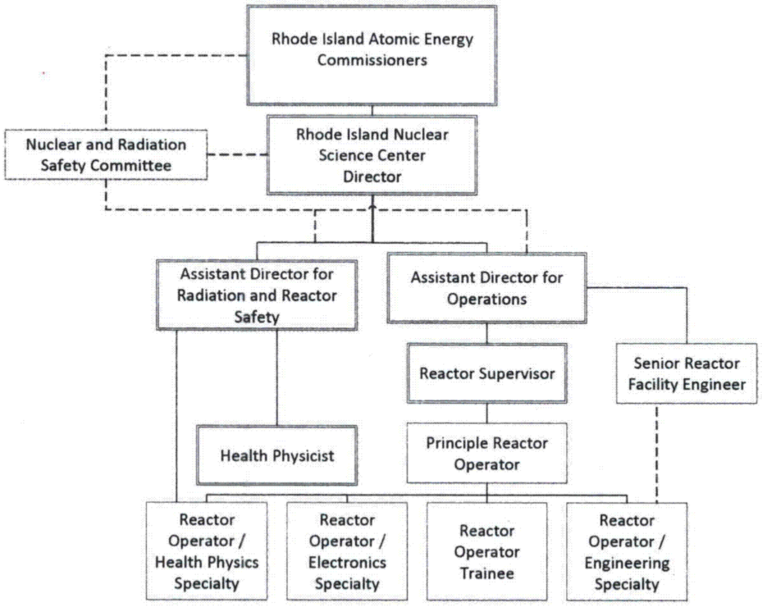


Figure 1 – Organization Chart

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 ~~set the policy~~ provide the general direction for the utilization of the organization.

6.1.2.2 Director

The Director ~~represents~~ 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 ~~that support RIAEC policy~~. 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 ~~day to day~~ 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.

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.~~5~~ 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.~~6~~ 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 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 Experiment installation or removal for experiments that have a reactivity worth greater than 0.75 %dK/K, and~~

6.1.3.2.~~4~~ 3 Recovery from an unscheduled significant reduction in power, and

6.1.3.2.~~5~~ 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 ~~three~~ 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 ~~two~~ four ~~years~~ of the ~~three~~ 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.~~5~~ 6 Senior Reactor Operators

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

6.1.4.1.~~6~~ 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 ~~seven (7)~~ 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.

6.2.2.4 If deficiencies that affect reactor safety are found, a written report shall be submitted to the RIAEC Commissioners within three months after the NRSC has completed its audit.

6.2.3 Review Function

The NRSC shall review the following items:

6.2.3.1 Proposed changes to the Technical Specifications or License, and violations of the Technical Specifications or License,

6.2.3.2 Proposed changes to the NRSC Charter,

6.2.3.3 Proposed changes in reactor safety related instrumentation or systems that have safety significance, and 10 CFR 50.59 evaluations,

6.2.3.4 New procedures, major changes to procedures that have safety significance, and violations of procedures that have safety significance,

6.2.3.5 New experiments,

6.2.3.6 Operating abnormalities that have a safety significance, and

6.2.3.7 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 ~~and approved~~ 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.

~~6.4.3 Substantive changes to the above procedures shall be made only with the approval of the NRSC. Temporary changes to the procedures that do not change their original intent may be made by a Senior Operator. Temporary changes to procedures shall be documented and subsequently reviewed by the NRSC.~~

6.5 Experiment Review and Approval

6.5.1 All new experiments shall be reviewed ~~and approved~~ 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 ~~and approved~~ 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 ~~upper~~ 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.~~3~~ 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.~~1~~ 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.~~2~~ 3 The occurrence shall be reported to the Director or Assistant Director.

6.6.2.~~3~~ 4 If the reactor is shutdown, operations shall not be resumed without authorization from the Director or Assistant Director for Operations.

6.6.2.~~4~~ 5 The occurrence, and corrective action taken shall be reviewed by the NRSC during its next scheduled meeting.

6.6.2.~~5~~ 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.

Follow Up TS Items

TS 1.46 – Surveillance Activities (pg. 9)
I don't think that I mentioned this one during the call. The concern with the definition is the addition of the statement: "ensuring the safe operation of the reactor." This statement is not included in the standard definition since it is technically inaccurate. The surveillance activity provides confidence that the system, structure, or component will perform its intended function when called upon to do so, but the "safe operation of the reactor" is ensured by the licensed operator. My legal staff will not concur with this definition in its current form.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 3.2.1.3 (pg. 22) – RPS trip table
General information, no action required – The choice of a trip set point greater than 100% RTP means that if you get this trip during operations you have also exceeded your license limit requiring NRC notification. Operation at 103% will not generate an automatic scram but is in violation of the license limit as well. Some facilities have adjusted their license limit to allow for the high power trip set point.

We are still thinking about this. Ideal limiting safety system settings would be:

Natural Convection Cooled Operation:

P <= 115% (100 kW) = 115 kW
H>= 23 ft 7 inches
T(bulk) <= 127 F

Forced Convection Cooled Operation:

P <= 115% (2 MW) = 2.3 MW
H>= 23 ft 7 inches
T(outlet) <= 122 F
F >= 1640 gpm

TS 3.7.2.1 – Airborne Effluents (pg. 45)
The basis section discusses the use of the COMPLY Code. The information for this calculation was submitted in an RAI response also needs to be reflected in the updated SAR.

Is this information in the response to RAI 14.110 ?

TS 3.7.2.2 – Liquid Effluents (pg. 45)
The specification reads like as surveillance: "The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined before release." This should be moved to section 4 of the TS and the LCO section 3 should read something like "All liquid waste is collected in the retention tank and discharges are within the regulatory limits."

Please see the revision made to this section of the Proposed Technical Specifications.

TS 3.8.1.4 – Fissionable Materials (pg. 46)
The TS should include a limit that restricts experiment failure consequences to the MHA dose limits. The basis already has a statement that "The radioactive material release is bounded by the analysis in SAR Chapter for the Maximum Hypothetical Accident involving a fuel element failure." This needs to be true for activated and fueled experiments and determined prior to irradiation or installation.

Please see the revision made to this section of the Proposed Technical Specifications.

The basis this was developed by taking the fuel failure accident scenario and recognizing that there would be no water to reduce the concentrations of iodine and noble gases that would be released by the experiment. The quantity of fissionable material was scaled down so that 10 CFR 20 limits would be met. Please see the Fissionable Experiment RAM Release Analysis 140529.

Basis for TS 3.9.3.1.4 (pg. 52) – Beam port ends shall be closed when the through port is not being monitored for leakage.
The basis states that "... the gate valves are in place, the consequence of the leak can be mitigated quickly by closing the valves." Closing of the valves seems to be more closely related to TS 3.9.3.1.3. A discussion about the sealing ability of the port ends would make more sense for the 3.9.3.1.4 basis.

Please see the revision made to this section of the Proposed Technical Specifications.

Basis for TS 4.1.1.1.3 (pg. 57)
The statement related to the ANL safety analysis of the equilibrium core seems to be more closely related to TS 3.1.1.1.3 (core negative temperature coefficient) and can be moved or added to that basis.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 4.2.3/4.2.5.9 (pg. 60) – watchdog scram
Considering the fact that the rod control system does not directly impact your reactor protection system, the one year interval for the watch dog scram can probably be justified. The basis should include a statement about the consequence of the computer system failure that is protected by the watch dog scram and the equipment reliability that implies an annual

surveillance frequency is acceptable. Some facilities perform this scram test as part of the annual rod control system testing. If you reach a safety consequence decision that does not justify annual testing, then adding a test switch that interrupts the communication line would be an optional way to test the scram that does not stress the Ethernet connector.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 4.3.a.1 (pg. 61 TS 4.3.1.1.1) – pH of the primary coolant shall be measured weekly
This TS does not have an equivalent in the draft TS. Weekly trending is valuable and should continue. The basis can include information on the acceptability of the values and related challenges to measuring pH in high purity water. pH and conductivity must both be monitored to prevent corrosion. We can discuss how to work with this. When you don't have a problem these values are sometimes challenging to measure, but monitoring with early detection and response is key to preventing any corrosion damage. As discussed in the phone call, since such a large volume of water is involved it does take time to reduce an elevated reading. But in contrast, operating with elevated conductivity is never a good idea especially if it is on an upward trend. The pH monitoring requirement was maintained when amendment 28 (August 2, 2001) was issued.

pH and conductivity are related and measure the same thing. Specification 3.3.1.1 has a conductivity limit for the primary coolant system. Therefore, pH is redundant. As the water becomes more pure, it becomes less possible to actually measure pH.

TS 4.3.1.1 (pg. 61) – conductivity of the primary coolant
The existing TS requires weekly testing, did you provide a safety analysis for changing to monthly? This is related to my question about the intention for quarterly averaging. When problems develop with Hx leakage, pool integrity, experiment leakage, early indication of fuel damage, the pH and conductivity sampling and trending will help identify a problem before it becomes significant, leading to a longer shut down of the facility.

Conductivity is not really used to measure any of these things. The underlying purpose of having pH and conductivity limits is to minimize the degradation of the aluminum pool liner and structural components. As mentioned above, pH and conductivity are related. As long as the water quality is not outside the conductivity limits for a long period of time, the aluminum structures in the pool will not have significant degradation. Conductivity (or pH) would not be particularly useful for measuring small Hx leakage, pool integrity, experiment leakage, or early indication of fuel damage because there is so much water in the pool that it would take a tremendous change in any of these factors in order to make a measurable change in pool water conductivity.

Heat exchanger leakage is determined by looking for Na-24 in the secondary coolant. Specification 4.3.2.1 requires that this measurement be performed.

Fuel damage is monitored by looking for Cs-137 and I-131 in the primary coolant. Specification 4.3.1.2 requires that this measurement be performed.

Pool integrity is monitored by paying attention to make-up water usage. Section 3.2 Table 3.1 requires a scram in the event of a low pool level.

<u>TS 4.7.1.2 (pg. 70)</u> – annual instrument calibration
The numbering for the radiation equipment starts with 4.2.7.1, it should start with 4.7.1.2.1. Most likely a copy and paste error when the radiation monitoring LCO was moved from section 3.2 to 3.7 and the surveillances were also moved but not re-numbered properly.

This has been corrected in the proposed Technical Specifications. After the content of the proposed Technical Specifications have been determined, an effort will be made to go back through and format the document with a consistent numbering system.

<u>Section 5 of TS</u>
It looks like this section was re-written following a March 2013 phone call. I need to spend a little more time with this section looking at Duke's comments and anything from my contractor to develop a consistent position. I don't expect anything significant to change with this section considering how recently it was revised.

No action necessary.

<u>TS 6.1.1 (pg. 85)</u> – Organization Structure
The statement that the facility director is responsible for the facility license contradicts the statement in TS 6.1.2 that the RIAEC has responsibility for the license. I believe the signature authority lies with the RIAEC as stated in the current operating license.

Recent events at the Rhode Island Science Center have made it clear that there was confusion over what constitutes the Rhode Island Atomic Energy Commission. It is an agency of the State of Rhode Island (Agency number 052), and it is responsible for the facility license.

The entity that holds the license for the Rhode Island Nuclear Science Center Research Reactor is the State of Rhode Island. The state hires a professional management team to manage, maintain, and operate the facility, and to be responsible for the license. The Governor appoints a group of five Commissioners, who provide direction with regard to how to utilize the facility to best serve the State of Rhode Island. The Commissioners are authorized to select their recommendation for who the state should hire to be the Director of the facility. Though the state is not obligated to choose the recommended selection, the Rhode Island Department of Human Resources typically does, unless they deem the selected candidate to have insufficient qualifications according to the state standards for the position. The qualifications required to be a Commissioner do not include any science or regulatory background at all. The Commissioners are uncompensated for their service, and have no legal liability associated with the organization. The Rhode Island Atomic Energy Commission is the state agency that consists of the Commissioners, and the staff which runs the Rhode Island Nuclear Science Center, which is the facility that houses the reactor.

The Director of the facility is the organization head. This is the individual that the state has hired to be responsible for the safe operation of the facility, and the license. This corresponds to the Level 1 function described in ANSI 15.4.

The Assistant Director for Operations is the individual that is responsible for the reactor operations side of the facility. This individual corresponds to the Level 2 function described in ANSI 15.4.

The Reactor Supervisor is the individual that is responsible for the day to day operation of the facility. This corresponds to the Level 3 function described in ANSI 15.4.

<u>TS 6.1.1 (pg. 85)</u> – Organization Structure
Individuals such as Information Systems Specialist, Senior Word Processing Typist, Custodian, and Student Intern, are typically excluded from RTR TS organizational charts and should be removed.

Please see the revision made to this section of the Proposed Technical Specifications.

<u>TS 6.1.2.4 (pg. 86)</u> – Assistant Director for Reactor and Radiation Safety
As discussed on the call this TS needs to reflect the final decision on how the position is going to be handled.

The Rhode Island Department of Human Resources has indicated that they have no intention of changing the title for this position. It will remain:

Assistant Director for Radiation and Reactor Safety

<u>TS 6.1.3.1.1 (pg. 87)</u> – Minimum Staffing requirement
See existing TS 6.1.2: The statement allowing the Reactor Operator to leave the control room and proceed to the pool top with the key in the console is not consistent with the regulations or currently acceptable practices.

The rods between the shim safety blades and the shim safety magnets are more than 23 feet long. As a result, the shim safety magnets and the armatures generally require manual alignment. The last step in the pre-start checkout is to center the magnets on the armatures because if they are not centered, the blades are likely to fall off of the magnets when they have been raised approximately half way to the critical blade heights. This allows the operator to make this adjustment with the reactor in shutdown condition. The alternative would be to shut all of the instrumentation off, and then turn it back on again just prior to start up.

TS 6.1.3.2.3 (pg. 88) – Experiment installation or removal
The reactivity worth of 0.75% dk/k requiring an SRO is greater than the maximum worth of an experiment.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 6.1.3.3.1 – Staff Contact List
The TS does not specify the details suggested from ANS 15-1 6.1.3(2)(a) [page 11] listing which personnel and what contact information is required.

RINSC has a staff of seven management / radiation safety / operations people so specifying these details is redundant. However, this change has been made.

TS 6.1.4.1.3 (pg. 88) – Assistant Director for Operations
Is this considered to be a level 2 (responsible for facility operation) or level 3 (day-to-day operations) position? If it is a level 2 position, then the qualifications are not consistent with ANS 15.4.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 6.2.1.1 (pg. 90) - NRSC Composition
The TS lists seven individuals, but the composition is a minimum. Where would additional members come from? I would guess that the four individuals in 6.2.1.1.4 would be a minimum. If that is the intention, then the specification should reflect that information. Likewise the charter specifies a quorum that is seven members, matching the list associated with 6.2.1.1. These TSs do not seem to provide flexibility as they are currently written.

After reviewing the NRSC Charter as it is currently written, in order to have a quorum, a minimum of four members must be present, which must include the RSO or designee, Director Assistant Director for Operations, and the Chairman of the Committee or designee. These individuals are specified so that there is a representative from both, the operations and the radiation safety sides of the house, as well as two independent individuals. There are currently twelve members. For years, we have had tremendous difficulty in getting a quorum. Our strategy for dealing with this has been to increase the size of the membership without increasing the size of the quorum so that it is easier to have a meeting. The proposed Technical Specifications have been re-written so that quorum is four members. Please see the revision made to this section of the Proposed Technical Specifications.

TS 6.4.2 (pg. 92) – Procedure review and approval by NRSC
The NRSC should review procedures, but approval should remain with the RINSC facility staff [see ANS 15.1 section 6.4 last paragraph pg. 12). TS 6.4.3, TS 6.5.1 and TS 6.5.2 contain similar statements and should also be revised. TS 6.5.3 contains the reference to "upper management" but that term is not defined.

While the ANSI standard suggests that the Level 2 management should approve procedures, historically at this facility the Safety Committee Charter has reviewed and approved procedures. The philosophy has been that procedures and experiments should require independent review and approval. These changes have been made. Please see the revisions made to these sections of the Proposed Technical Specifications.

TS 6.6.1.4 (pg. 93) - SL violation report
The report should also include a statement of impact to facility personnel, general public, and any release to the environment.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 6.6.2.1 (pg. 94) – Actions following reportable occurrence
Per 10 CFR 50.36(b)(2) when a limiting condition for operation is not met "the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification." The TS needs to include the action of shutting down the reactor.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 6.7.1.6 (pg. 95) – environmental Monitoring report
The TS needs to include the location and dose at that location. The annual report currently contains this information, but the requirement should be included in the specification.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 6.7.1.7 (pg. 95) – Annual dose report
The limit for public exposure is 100 mREM, so the report should be generated for visitors that exceed 100 mREM not the 500 mREM listed in the TS.

Please see the revision made to this section of the Proposed Technical Specifications.

TS 6.7.2.2 (pg. 96) – Reporting Requirements for Other Significant Events
Do not use the term Unusual Event for the events listed as it is an Emergency Plan term.

Please see the revision made to this section of the Proposed Technical Specifications.

Telephone Conversation TS Items

TS 3.3 – Primary Coolant pH

pH and conductivity are related and measure the same thing. Specification 3.3.1.1 has a conductivity limit for the primary coolant system. Therefore, pH is redundant.

TS 3.3 – Pool Level Scram

Rather than specifying this scram in Section 3.3, it is already specified in Section 3.2 Table 3.1.

Email RAI

I was looking as some of your RAI responses today and I had a question about your response to RAI 7.3 that you provided on June 10, 2010. The question asked for the alignment tolerance for both the bridge movement scram and misalignment. You provided a tolerance for the movement scram based on the teeth spacing, which looks fine. I did not see a response indicating the tolerance for the closure and opening of the high power position switch. This is the one at the end of travel position that ensures the bridge is close enough to couple with the forced flow pumps. I observed the switch during our walk down, but I cannot find the response where you indicate the sensitivity/mechanical tolerance of the switch, i.e. how far away can the bridge be to the high power end and still close the switch. Do you know if you provided that information in an RAI response and if so can you point me to the correct location?

Here is the portion of the RAI response I was looking to find the information but did not see it.

> **The bridge misalignment scram consists of a position switch mounted at the high power end of the reactor pool. When the bridge is not fully seated at the high power end of the pool, the switch is in a state that initiates the "Bridge Lo Pwr Pos" scram. This scram is only active when the reactor is being operated in the forced convection cooling mode. For natural convection cooled operation at 100 kW or less, this scram is bypassed.**

I was expecting to see the tolerance associated with being "fully seated at the high power end of the pool". The next paragraph discusses the spacing in the gears for moving the bridge and identified a ¼ inch tolerance for the bridge movement.

This tolerance has not been specified thus far because it is related to the bridge movement scram. When the bridge is fully seated in the high power end of the pool, motion is physically stopped by the pool wall and cannot go any further. The high power position switch is positioned so that it is closed when the bridge is in this position. At the same time, the pool movement scram switch is reset so that it rests on top of one of the gear teeth. If the bridge moves, the scenario is that the bridge movement scram switch will change state within ¼ inch of movement. That being said, I measured the tolerance on the high power position switch (while holding the bridge movement switch up, effectively bypassing it) and found that the scram occurs when the bridge moves 3/8 inch.

Heartburn Issues

<u>TS 3.8</u> – Experiment Reactivity Determination

We have had a conversation about how to verify that experiments that are installed in the core do not exceed the reactivity limits on experiments. Initially, RINSC proposed that we be allowed to install experiments, and do an approach to critical to determine reactivity worth. This leads to the possibility that we will discover that a proposed experiment is not within the reactivity limits, while at the same time, it has been installed in the core. NRC proposed that modelling be used to estimate the reactivity worths of experiments for which the reactivity cannot be estimated based on similar, previously performed experimental data. The problem with this is that it is not possible to model unknowns.

There is a VERY big push to get the RINSC reactor back to the forefront as a research tool, rather than just being a toy for students to play with. In order to do this, there must be some ability to analyze unknowns, and perform experiments that have never been done before. Consequently, RINSC is reluctant to pretend like it is possible to know the reactivity worths of these types of experiments without making a measurement.

Historically, research reactors would do an approach to critical to determine the reactivity worths of unknowns, and short initial irradiations to see what activation products were going to be produced. My task is to find a way for the RINSC reactor to be able to continue to do this.

Fissionable Experiment RAM Release Analysis 140529

Assumptions

For the fuel failure analysis, an assumption was made that the entire fission fragment inventory for one fuel plate was released. If we start with the assumption that an experiment is fuelled with the amount of fissionable material in one fuel plate, then there would be:

$$\left[\frac{275\ g\ Fissionable\ Material}{Fuel\ Element}\right]\left[\frac{Fuel\ Element}{22\ Fuel\ Plates}\right]\left[\frac{Fuel\ Plate}{Experiment}\right]$$

$$= \left[\frac{12.5\ g\ Fissionable\ Material}{Experiment}\right]$$

Source Term

The source term for the fuel failure analysis was:

1. Fission Rate

   A. The energy associated with each fission that occurs in the reactor is 200 MeV per fission.

   B. Converting from MeV to MW – Seconds:

   $$\left[\frac{200 MeV}{fission}\right]\left[\frac{1.6 X 10^{-13}\ Joule}{MeV}\right]\left[\frac{Watt - Second}{Joule}\right]\left[\frac{MW}{10^{6}\ Watt}\right]$$

   $3.2 \text{ X } 10^{-17}$ MW – second per fission

   C. Therefore the fission rate at 1 MW power is:

   $3.1 \text{ X } 10^{16}$ fission per MW – second

   D. The RINSC reactor operates at a maximum power level of 2 MW, so the fission rate at full power operation is:

   $$\left[\frac{3.1 X 10^{16}\ fission}{MW - second}\right]\left[\frac{2\ MW}{1}\right]$$

   $6.2 \text{ X } 10^{16}$ fission / second

2. Fission Nuclide Production Rate

A. The fission nuclide production rate for the i th fission product nuclide is the product of the fission rate and the fission product yield (γ_i) for the i th fission product:

Fission Nuclide Production Rate = $(6.2 \times 10^{16}$ fission / second)(γ_i)

3. Fission Nuclide Decay Rate

A. The fission nuclide decay rate for the i th fission product nuclide is the product of the decay constant for the i th fission product nuclide (λ_i), and the number of atoms of the nuclide that are present in the core (N_i):

Fission Nuclide Decay Rate = $(\lambda_i)(N_i)$

4. Fission Product Saturation

A. Fission product saturation occurs when the production rate and decay rate are the same. Therefore, for the i th fission product, saturation is when:

$(6.2 \times 10^{16}$ fission / second)($\gamma_i) = (\lambda_i)(N_{i\ sat})$

B. Therefore, if we wanted to estimate the number of atoms of the i th fission product in the core at saturation, it would be:

$$N_{i\ sat} = \left[\frac{6.2 X 10^{16}\ fission}{second}\right]\left[\frac{\gamma_i\ atoms}{fission}\right]\left[\frac{second}{\lambda_i}\right]$$

C. However, if we want to estimate the activity of the i th fission product in the core at saturation, it would be:

Activity (Bq) = $(\lambda_i)(N_{i\ sat}) = (6.2 \times 10^{16}$ fission / second)(γ_i)

D. The activity can be converted to units of Ci by using the conversion factor:

1 Ci = 3.7×10^{10} Bq

E. If we make the simplifying assumption that the activity in the core is evenly spread over all of the fuel plates, the activity per fuel plate would be:

$$\left[\frac{22\,Plates}{Fuel\,Element}\right]\left[\frac{14\,Fuel\,Elements}{Core}\right]=\left[\frac{308\,Plates}{Core}\right]$$

Therefore the activity of the i th fission product per fuel plate is the activity in the core divided by 308 fuel plates

F. As an example, consider I-131 saturation in the core, which has a yield of $\gamma = 0.029$ atoms per fission, and a decay constant of $\lambda = 9.73 \times 10^{-7}$ per second:

1. Saturation Activity in the Core:

$$\left[\frac{6.2X10^{16}\,fission}{second}\right]\left[\frac{0.029\,I-131\,atoms}{fission}\right]$$

$= 1.8 \times 10^{15}$ I-131 atoms per second

$= 1.8 \times 10^{15}$ Bq I-131

$$\left[\frac{1.8X10^{15}\,Bq\,I-131}{1}\right]\left[\frac{Ci}{3.7X10^{10}\,Bq}\right]$$

$= 4.86 \times 10^{4}$ Ci

2. Saturation Activity in a Fuel Plate:

$$\left[\frac{4.86X10^{4}\,Ci}{Core}\right]\left[\frac{Core}{308\,Fuel\,Plates}\right]$$

$= 1.58 \times 10^{2}$ Ci per fuel plate

G. Most of the fission products do not get out of the fuel matrix. Of the isotopes that get into the pool water, there is so much solvent in comparison to solute that the vast majority of the isotopes would stay dissolved in the pool water. The fission products that are both volatile and long lived enough to potentially escape from the fuel to the pool are:

Source Term					
Nuclide	Decay Constant (/s)	Fission Yield (Atoms/Fission)	Core Activity (Bq)	Core Activity (Ci)	Single Plate Activity (Ci)
I-131	9.73E-07	0.029	1.80E+15	4.86E+04	1.58E+02
I-132	8.02E-05	0.043	2.67E+15	7.21E+04	2.34E+02
I-133	9.25E-06	0.065	4.03E+15	1.09E+05	3.54E+02
I-134	2.20E-04	0.08	4.96E+15	1.34E+05	4.35E+02
I-135	2.88E-05	0.064	3.97E+15	1.07E+05	3.48E+02
Kr-85m	4.41E-05	0.013	8.06E+14	2.18E+04	7.07E+01
Kr-85	2.05E-09	0.000255	1.58E+13	4.27E+02	1.39E+00
Kr-87	1.48E-04	0.025	1.55E+15	4.19E+04	1.36E+02
Kr-88	6.95E-05	0.036	2.23E+15	6.03E+04	1.96E+02
Xe-131m	6.67E-07	0.029	1.80E+15	4.86E+04	1.58E+02
Xe-133m	3.50E-06	0.065	4.03E+15	1.09E+05	3.54E+02
Xe-133	1.53E-06	0.065	4.03E+15	1.09E+05	3.54E+02
Xe-135m	7.40E-04	0.064	3.97E+15	1.07E+05	3.48E+02
Xe-135	2.11E-05	0.064	3.97E+15	1.07E+05	3.48E+02

A conservative assumption is made that all of the available activity escapes into the pool water.

Therefore this is the worst case expected release from an experiment with 12.5 grams of fissionable material in it.

Quantity of RAM that Reaches Confinement Air

For the fuel failure analysis, University of Virginia empirical data was used to show that only 0.1% of the noble gases, and 0.01% of the iodine inventory was actually released from the pool into confinement.

In the case of an experiment failure, the release could potentially be directly to the confinement room, in which case 100 % of the noble gases and iodines would be released to confinement. Consequently, the release to confinement would be:

Release to Confinement Air				
Nuclide	Single Plate Activity (Ci)	Release to Confinement (Ci)	Release to Confinement (microCi)	Confinement Concentration (microCi / cc)
I-131	1.58E+02	1.58E+02	1.58E+08	3.06E-02
I-132	2.34E+02	2.34E+02	2.34E+08	4.54E-02
I-133	3.54E+02	3.54E+02	3.54E+08	6.87E-02
I-134	4.35E+02	4.35E+02	4.35E+08	8.45E-02
I-135	3.48E+02	3.48E+02	3.48E+08	6.76E-02
Kr-85m	7.07E+01	7.07E+01	7.07E+07	1.37E-02
Kr-85	1.39E+00	1.39E+00	1.39E+06	2.69E-04
Kr-87	1.36E+02	1.36E+02	1.36E+08	2.64E-02
Kr-88	1.96E+02	1.96E+02	1.96E+08	3.80E-02
Xe-131m	1.58E+02	1.58E+02	1.58E+08	3.06E-02
Xe-133m	3.54E+02	3.54E+02	3.54E+08	6.87E-02
Xe-133	3.54E+02	3.54E+02	3.54E+08	6.87E-02
Xe-135m	3.48E+02	3.48E+02	3.48E+08	6.76E-02
Xe-135	3.48E+02	3.48E+02	3.48E+08	6.76E-02

Confinement Building

A negative pressure is maintained in the confinement building so that all of the air that exits the buiding will exit through a stack. If an airborne RAM release is detected, the Emergency Air Handling System is activated, and the airflow is directed through an emergency filter prior to reaching the stack.

During facility re-licensing, the volume of the confinement building was determined to be approximately 203,695 cubic feet. The volume of the pool structure and water was determined to be 21,740 cubic feet, leaving 181,955 cubic feet of open space. The control room takes up about 3,612 cubic feet of this space. Converting the free volume of the confinement room to cubic centimeters:

$$\left[\frac{181955\,ft^3}{1}\right]\left[\frac{(12\,in)^3}{ft^3}\right]\left[\frac{(2.54\,cm)^3}{(in)^3}\right]=5.15\,X\,10^9\,cm^3$$

Concentration of RAM in the Confinement Air

If we assume that the quantity of RAM that reaches the confinement air is spread uniformly throughout confinement, The concentration of each nuclide in the confinement building air would be:

(μCi of Nuclide in Confinement) / (5.15 X 10^9 cm^3) = μCi / cm^3

If we continue with our I-131 example:

1. We concluded that there was 1.58E+8 μCi released to the confinement air.

2. Therefore, the concentration of I-131 inside the confinement building is:

$$(1.58E+8\ \mu Ci) / (5.15 \times 10^9\ cm^3) = 3.06 \times 10^{-2}\ \mu Ci / cm^3$$

Therefore, the average concentration of each of the major nuclides would be:

Release to Confinement Air				
Nuclide	Single Plate Activity (Ci)	Release to Confinement (Ci)	Release to Confinement (microCi)	Confinement Concentration (microCi / cc)
I-131	1.58E+02	1.58E+02	1.58E+08	3.06E-02
I-132	2.34E+02	2.34E+02	2.34E+08	4.54E-02
I-133	3.54E+02	3.54E+02	3.54E+08	6.87E-02
I-134	4.35E+02	4.35E+02	4.35E+08	8.45E-02
I-135	3.48E+02	3.48E+02	3.48E+08	6.76E-02
Kr-85m	7.07E+01	7.07E+01	7.07E+07	1.37E-02
Kr-85	1.39E+00	1.39E+00	1.39E+06	2.69E-04
Kr-87	1.36E+02	1.36E+02	1.36E+08	2.64E-02
Kr-88	1.96E+02	1.96E+02	1.96E+08	3.80E-02
Xe-131m	1.58E+02	1.58E+02	1.58E+08	3.06E-02
Xe-133m	3.54E+02	3.54E+02	3.54E+08	6.87E-02
Xe-133	3.54E+02	3.54E+02	3.54E+08	6.87E-02
Xe-135m	3.48E+02	3.48E+02	3.48E+08	6.76E-02
Xe-135	3.48E+02	3.48E+02	3.48E+08	6.76E-02

Emergency Filter

When the Emergency Air Handling System is activated, all of the air from the confinement room is exhausted through an emergency filter. The emergency filter consists of:

1. Roughing Filter
2. HEPA Filter
3. Charcoal Filter
4. HEPA Filter

Each of the HEPA filters is 99.97% efficient for removing particles that are 0.3 microns or greater. The charcoal filter is required by RINSC Technical Specifications to be 99.97% efficent for removing iodine. Therefore:

1. The concentration of airborne particles that are 0.3 microns or greater in the confinement room that will reach the stack is:

   (0.03%)(0.03%)(Confinement Concentration)

   = (0.0009%)(Confinement Concentration)

   = (0.00009)(Confinement Concentration)

   = (9×10^{-5})(Confinement Concentration)

2. The concentration of iodine in the confinement room that will reach the stack is:

   (0.03%)(Confinement Concentration)

   = (0.0003)(Confinement Concentration)

3. The noble gases are unaffected by the HEPA filters, but are slowed by the charcoal filter. We will assume that all of the noble gases are released to the stack.

4. If we continue with our I-131 example:

   A. We concluded that there the concnetration of I-131 in confinement was 3.06E-2 μCi / cm^3 in the confinement air.

   B. Therefore, the concentration of I-131 that is exhausted from the emergency air filter and reaches the stack is:

   (0.0003)(3.06E-2 μCi / cm^3) = 9.19E-6 μCi / cm^3

Concentration of RAM in the Emergency Air Filter Exhaust

If we assume that the fraction of the iodine that is exhausted by the emergency filter is 0.0003, and that all of the noble gases make it through the filter, the concentrations of RAM that reach the building exhaust stack are:

Release to Stack			
Nuclide	Confinement Concentration (microCi / cc)	Emergency Air Filter Release Fraction	Emergency Air Filter Exhaust Concentration (microCi / cc)
I-131	3.06E-02	0.03%	9.19E-06
I-132	4.54E-02	0.03%	1.36E-05
I-133	6.87E-02	0.03%	2.06E-05
I-134	8.45E-02	0.03%	2.54E-05
I-135	6.76E-02	0.03%	2.03E-05
Kr-85m	1.37E-02	100%	1.37E-02
Kr-85	2.69E-04	100%	2.69E-04
Kr-87	2.64E-02	100%	2.64E-02
Kr-88	3.80E-02	100%	3.80E-02
Xe-131m	3.06E-02	100%	3.06E-02
Xe-133m	6.87E-02	100%	6.87E-02
Xe-133	6.87E-02	100%	6.87E-02
Xe-135m	6.76E-02	100%	6.76E-02
Xe-135	6.76E-02	100%	6.76E-02

RAM Release Rate

Based on empirical data, when the Emergency Air Handling System is running, the average air flow rate out of the confinement building and through the emergency filter is:

Year	Clean-Up Blower Flow Rate (cfm)
2008	643
2009	1487
2010	1397
2011	775
2012	968
Average	1054

$$\left[\frac{1054\,ft^3}{\min}\right]\left[\frac{(12\,in)^3}{ft^3}\right]\left[\frac{(2.54\,cm)^3}{(in)^3}\right]\left[\frac{\min}{60\,s}\right]=4.97X10^5\ cm^3/s$$

Despite the fact that there is a dilution blower, it is irrelevant because we are interested in the RAM release rate rather than the concentration that is being released from the stack.

Consequently, the release rate of the RAM from the stack is:

$$\left[\frac{Emergency\,Filter\ Exhaust\,Concentration\,\mu Ci}{cm^3}\right]\left[\frac{Emergency\,Filter\ Exhaust\ Flowrate\,cm^3}{s}\right]=\frac{\mu Ci}{s}$$

Continuing with the I-131 example, the concentation of I-131 in the emergency filter exhaust was 9.19 X 10^{-10} µCi / cm^3, so the I-131 release rate is:

$$\left[\frac{9.19X10^{-10}\,\mu Ci}{cm^3}\right]\left[\frac{4.97X10^5\,cm^3}{s}\right]=4.57X10^{-4}\,\mu Ci/s$$

Overall:

RAM Release Rate		
Nuclide	Emergency Air Filter Exhaust Concentration	Stack RAM Release
I-131	9.19E-06	4.57E+00
I-132	1.36E-05	6.77E+00
I-133	2.06E-05	1.02E+01
I-134	2.54E-05	1.26E+01
I-135	2.03E-05	1.01E+01
Kr-85m	1.37E-02	6.83E+03
Kr-85	2.69E-04	1.34E+02
Kr-87	2.64E-02	1.31E+04
Kr-88	3.80E-02	1.89E+04
Xe-131m	3.06E-02	1.52E+04
Xe-133m	6.87E-02	3.41E+04
Xe-133	6.87E-02	3.41E+04
Xe-135m	6.76E-02	3.36E+04
Xe-135	6.76E-02	3.36E+04

Atmospheric Dispersion

The assumptions made for release to the atmosphere are:

1. Release is at ground level
2. Conditions are Pasquill Type F
3. Wind speed is one meter per second
4. Wind direction is constant over the entire duration of the release

In reality, the exhaust from the emergency filter goes up a 115 foot stack, which would increase the dispersion before the plume reaches ground level. Conservatively, it is assumed that the plume is released at ground level.

Conditions are assumed to be Pasquill Type F. Atmospheric stability is a measure of the turbulence in the plume, and it affects the rate of dispersion of the plume. The more turbulent the air is, the greater the dispersion rate. There are six classifications of atmospheric stability, ranging from Pasquill Type A through Pasquill Type F, in which A is extremely unstable, and F is moderately stable. Consequenly, the assumption of Pasquill Type F is conservative because it minimizes the dispersion rate, and maximizes the airborne RAM concentrations at ground level.

The wind speed is assumed to be one meter per second. Higher wind speeds increase dilution because the RAM released per unit time is added to a larger volume of air passing by the release point. Consequently, this assumption is conservative.

Wind direction is assumed to be constant, As a result, all of the concentration of RAM will be along one line of direction, rather than dispersed across more than one direction. Consequently, this assumption is conservative.

Atmoshperic dispersion calculations estimate the concentration of some material in air for a given release rate, under specified atmospheric conditions, at some distance away from the source. A Gaussian Straight Line Plume Model is used. Typical factors used in this calculation are:

1. X is the concetration of the material in the air (Activity / Volume)
2. Q is the material release rate (Activity / Time)
3. σ_y is the horizontal dispersion coefficient (Distance)
4. σ_Z is the vertical dispersion coefficient (Distance)
5. h is the stack height (Distance)
6. u is the average wind speed of the plume (Distance / Time)
7. t is the plume travel time to the point of interest (Time)
8. x is the downwind distance (Distance)
9. y is the horizontal distance at right angles to the plume centerline (Distance)
10. z is the height above the ground (Distance)

We are assuming that the release is at ground level (h = 0), and we are interested in the RAM concentration at ground level (z = 0). The highest concentration will be along the plume centerline (y = 0). Therefore, the general equation used to calculate the concentration for only the downwind sector, and for only one wind speed and one stability class is:

$$\frac{\chi}{Q} = \frac{1}{\pi \sigma_y \sigma_z u}$$

The minimum distance between the reactor core and the site boundary is 48 meters. RINSC has the authority to prevent the public from entering this boundary. Consequently, we are interested in the concentration at 48 meters from the source. The dispersion coefficients take this distance into account.

The dispersion coefficents are quantitative measures of how much the plume has spread out in the horizontal (y) and vertical (z) directions. The material concentration in the plume as a function of distance from the plume centerline is a Gaussian distribution, with maximum concentration at the centerline. The dispersion coeficients are the standard deviations in each direction. Therefore, 68% of the plume is within σ_y distance form the centerline in the y – direction, and σ_z distance from the centerline in the z – direction. typically the plume spreads out more in the horizontal direction than it does in the vertical direction. Consequently σ_y is generally larger than σ_z. Based on dispersion coefficient curves given in the US Atomic Energy Commission's "Meteorology and Atomic Energy, 1968", pp. 102 – 103 for Pasquill Type F conditions, the dispersion coeficients for a point 48 meters from the source are:

σ_y = 2.0 meters
σ_z = 1.2 meters

Therefore the atmospheric dispersion equation becomes:

$$\frac{\chi}{Q} \approx \frac{1}{\pi \sigma_y \sigma_z u} = \frac{1}{(\pi)(2.0m)(1.2m)(1m/s)} = 0.133 s/m^3$$

$$\left[\frac{0.133s}{m^3}\right]\left[\frac{m^3}{(100cm)^3}\right] = 1.33X10^{-7} s/cm^3$$

For ground releases, a correction is needed to account for building wake effects, particularly if the point of interest is close to the building. The wake of the building increases the horizontal and vertical dispersion. The wake effect reduces the concentration by a factor of three, which reduces the concentration reduction factor (X / Q) to an effective concentration reduction factor $(X / Q)_{eff}$ of:

$$(X / Q)_{eff} = (1 / 3)(1.33 \text{ X } 10^{-7} \text{ s / cm}^3) = 4.43 \text{ X } 10^{-8} \text{ s / cm}^3$$

The release rate (Q) for each isotope is the release rate from the emergency air filter. Therefore the concentrations 48 meters down wind (X) is predicted to be:

$$(4.43 \text{ X } 10^{-8} \text{ s / cm}^3)(\text{Isotope Release Rate } \mu\text{Ci / s}) = \text{Concentration } \mu\text{Ci / cm}^3$$

Continuing with the I-131 example, the release rate of I-131 from the emergency filter was 4.57 μCi / s, so the concentration of I-131 that is predicted to be 48 meters down wind of the facility is:

$$\left[\frac{4.57\ \mu Ci}{s}\right]\left[\frac{4.43\ X\ 10^{-8}}{cm^3}\right] = 2.02\ X\ 10^{-7}\ {}^{\mu Ci}\!/_{cm^3}$$

This means that the concentrations of the radionuclides at the site boundary are:

Site Boundary Concentration			
Nuclide	**Emergency Air Filter Exhaust Concentration (microCi / cc)**	**Stack Release Rate (microCi / s)**	**Site Boundary Concentration (microCi / cc)**
I-131	9.19E-06	4.57E+00	2.02E-07
I-132	1.36E-05	6.77E+00	3.00E-07
I-133	2.06E-05	1.02E+01	4.53E-07
I-134	2.54E-05	1.26E+01	5.57E-07
I-135	2.03E-05	1.01E+01	4.46E-07
Kr-85m	1.37E-02	6.83E+03	3.02E-04
Kr-85	2.69E-04	1.34E+02	5.92E-06
Kr-87	2.64E-02	1.31E+04	5.81E-04
Kr-88	3.80E-02	1.89E+04	8.36E-04
Xe-131m	3.06E-02	1.52E+04	6.73E-04
Xe-133m	6.87E-02	3.41E+04	1.51E-03
Xe-133	6.87E-02	3.41E+04	1.51E-03
Xe-135m	6.76E-02	3.36E+04	1.49E-03
Xe-135	6.76E-02	3.36E+04	1.49E-03

Dose Calculation Background Information

Health effects of radiation dose are separated into two categories:

A. Stochastic Effects – These effects are probabilistic, and are due to random ionization events. Consequently, there is no threshold for these effects, and the probability of occurrence is proportional to the dose received. Cancer is an example of these types of effects.

B. Non-Stochastic Effects – These effects depend on the amount of dose received beyond a minimum threshold, and the amount of damage depends on the magnatude of the dose. Skin erythmia is an example of a non-stochastic effect.

The objective of dose limits are to minimize the risk of stochastic effects, and to prevent the occurrence of non-stochastic effects. The dose limits have been designed to be independent of whether or not the radiation dose is uniform or non-uniform. This is achieved by having "effective dose" limits in which the "effective dose" takes into consideration the risk due to the irradiation of each individual organ and equates it to the risk associated with a uniform irradiation of the whole body.

Important definitions that can be found in 10 CFR 20:

A. Allowable Limit on Intake (ALI) – This is the amoung of RAM taken into the body via ingestion or inhalation that would lead to a committed effective dose equivalent of 5 Rem, or 50 Rem to any individual tissue or organ.

B. Derived Air Concentration (DAC) – This is the concentration of a given radionuclide in air which if inhaled at a rate of 1.2 m^3 per hour for one working year (2000 hours) would result in reaching the ALI.

C. Derived Air Concentration - Hour (DAC - Hour) –This is the product of the concentration of RAM in the air expressed as a fraction or multiple of the DAC, and the exposure time expressed in hours.

D. Absorbed Dose (D) – This is a measure of the radiation energy that is absorbed per unit mass of material of interest.

E. Dose Equivalent (H) – This is the product of the absorbed dose in tissue, quality factor (Q), and all other necessary modifying factors at the location of interest. The units of dose equivalent are the rem and sievert (Sv). In general:

$$H = DQ$$

F. Quality Factor (Q) – This is a regulatoraly defined factor to account for the fact that the type and energy of the incident radiation has an effect on the amount of biological damage that is produced per unit of absorbed energy (absorbed dose).

G. Tissue Dose Equivalent (H_T) – This is the dose equivalent to a specific tissue or organ due to external sources.

H. Committed Dose Equivalent ($H_{T,50}$) – This is the dose equivalent to organs or tissues of reference (T) that will be received from a single intake of radioactive material by an individual that will be accumulated over the 50-year period following the intake.

I. Effective Dose Equivalent (H_E) – This equates the risk of a non-uniform external dose, or internal dose to the risk associated with a dose that is distributed uniformly over the whole body. A regulatoraly defined weighting factor (W_T) is used for each organ, and the overall effective dose equivalent is:

$$H_E = \Sigma\, W_T H_T$$

This is the sum of the products of the dose equivalent to the organ or tissue (H_T) and the weighting factors (W_T) applicable to each of the body organs or tissues that are irradiated ($H_E = \Sigma W_T H_T$).

J. Committed Effective Dose Equivalent ($H_{E,50}$) – This is the effective dose equivalent accumulated over a 50 period as a result of a single intake of radioactive material. In general:

$$H_{E,50} = \Sigma W_T H_{T,50}$$

This is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues ($H_{E,50} = \Sigma W_T H_{T,50}$).

K. Deep Dose Equivalent (DDE) – This is the whole body dose at a depth of 1 cm due to an external exposure.

L. Total Effective Dose Equivalent (TEDE) – This is the sum of the DDE due to an external dose, and the CEDE due to an internal dose from an intake of radioactive material. In general:

TEDE = DDE + CEDE

Regulatory Limits

Regulatory limits on dose are defined in 10 CFR 20:

Occupational Dose Limits

A. TEDE = 5 rem / yr [10 CFR 20.1201(a)(1)(i)]

B. DDE + CDE to any individual organ or tissue = 50 rem / yr [10 CFR 20.1201(a)(1)(ii)]

C. The DAC and ALI may be used to determine the individual's dose and to demonstrate compliance with dose limits. [10 CFR 20.1201(d)]

D. If the only intake of radionuclides is by inhalation, the total effective dose equivalent limit is not exceeded if the sum of the deep-dose equivalent divided by the total effective dose equivalent limit, and one of the following, does not exceed unity [10 CFR 20.1202(b)]:

  (1) The sum of the fractions of the inhalation ALI for each radionuclide, or
  (2) The total number of derived air concentration-hours (DAC-hours) for all radionuclides divided by 2,000, or
  (3) The sum of the calculated committed effective dose equivalents to all significantly irradiated[1] organs or tissues (T) calculated from bioassay data using appropriate biological models and expressed as a fraction of the annual limit.

E. If the identity and concentration of each radionuclide in a mixture are known, the fraction of the DAC applicable to the mixture for use in calculating DAC-hours must be either [10 CFR 20.1204(e)]:

  (1) The sum of the ratios of the concentration to the appropriate DAC value (e.g., D, W, Y) from appendix B to part 20 for each radionuclide in the mixture; or
  (2) The ratio of the total concentration for all radionuclides in the mixture to the most restrictive DAC value for any radionuclide in the mixture.

F. In order to calculate the committed effective dose equivalent, the licensee may assume that the inhalation of one ALI, or an exposure of 2,000 DAC-hours, results in a committed effective dose equivalent of 5 rems (0.05 Sv) for radionuclides that have their ALIs or DACs based on the committed effective dose equivalent. [10 CFR 20.1204(h)(1)]

G. When the ALI (and the associated DAC) is determined by the nonstochastic organ dose limit of 50 rems (0.5 Sv), the intake of radionuclides that would result in a committed effective dose equivalent of 5 rems (0.05 Sv) (the stochastic ALI) is listed in parentheses in table 1 of appendix B to part 20. In this case, the licensee may, as a simplifying assumption, use the stochastic ALIs to determine committed effective dose equivalent. However, if the licensee uses the stochastic ALIs, the licensee must also demonstrate that the limit in § 20.1201(a)(1)(ii) is met. [10 CFR 20.1204(h)(2)]

Dose Limits for Individual Members of the Public

A. TEDE = 100 mrem / yr [10 CFR 20.1301(a)(1)]

B. A licensee shall show compliance with the annual dose limit in § 20.1301 by [10 CFR 20.1302(b)]:

   (1) Demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit; or

   (2) Demonstrating that:

      (i) The annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in table 2 of appendix B to part 20; and

      (ii) If an individual were continuously present in an unrestricted area, the dose from external sources would not exceed 0.002 rem (0.02 mSv) in an hour and 0.05 rem (0.5 mSv) in a year.

Therefore, based on the regulations, we must show that:

A. The occupational doses to individuals inside confinement are no greater than:

   1. TEDE = 5 rem
   2. DDE + CDE to any individual organ or tissue = 50 rem / yr

B. The dose to the public at the site boundary is no greater than:

   1. TEDE = 100 mrem / yr

Use of the DAC to determine the Deep Dose Equivalent (DDE)

The ALI and DAC for any given nuclide can be found in 10 CFR 20 Appendix B. For Kr-85, the inhalation value of the DAC for occupational exposure are given to be:

A. DAC = $1 \text{ X } 10^{-4}$ μCi / cm^3

   This means that if an individual is immersed in a concentration of $1 \text{ X } 10^{-4}$ μCi / cm^3 Kr-85 for 2000 hours, they would receive a DDE of 5 Rem whole body.

If and individual is immersed in an air concentration equivalent to one DAC, the average dose rate that they would be receiving would be:

A. Whole Body:

$$\left[\frac{5rem}{2000hr}\right] = 2.5mrem / DAC - hr$$

We can express the concentration of RAM in air as a fraction or multiple of the DAC:

DAC Fraction (Multiple) = (Air Concentration) / (DAC)

If we had a concentration of 2.69 X 10^{-7} μCi / cm^3 of Kr-85 in the confinement air, the DAC fraction (multiple) if the occupational DAC were 1 X 10^{-4} μCi / cm^3 would be:

DAC Fraction (Multiple) = (Air Concentration) / (DAC)

= (2.69 X 10^{-7} μCi / cm^3) / (1 X 10^{-4} μCi / cm^3)

= 2.69 X 10^{-3} DAC

If we know the concentration of RAM in air as a fraction or multiple of the DAC, then we can determine the dose rate that that an individual immersed in the air would recieve. Continuing with the Kr-85 example:

$$\left[\frac{2.5\,mrem}{DAC-hr}\right]\left[\frac{2.69X10^{-3}\,DAC}{1}\right]=6.73X10^{-3}\,mrem/hr$$

If there is more than one nuclide in the air, then the DAC fractions can be added together before determining the dose rate. Consider:

A. Suppose that the air has a concentration of 2.69 X 10^{-7} μCi / cm^3 of Kr-85, and 1.37 X 10^{-5} μCi / cm^3 of Kr-85m in it. The DAC fractions are:

(Air Concentration) / (DAC)

Where the DAC is defined in 10 CFR 20 for each isotope

B. For Kr-85 the DAC fraction has been previously calculated to be 2.69 X 10^{-3} DAC.

C. For Kr-85m, given that the DAC is 2 X 10^{-5} μCi / cm^3, the DAC fraction is:

(1.37 X 10^{-5} μCi / cm^3) / (2 X 10^{-5} μCi / cm^3) = 0.685 DAC

D. Therefore the total DAC fraction is:

Total DAC Fraction = 2.69 X 10^{-3} DAC + 0.685 DAC = 0.688 DAC

E. Therefore the deep dose equivalent is:

$$\left[\frac{2.5\,mrem}{DAC-hr}\right]\left[\frac{0.688\,DAC}{1}\right]=1.72\ mrem/hr$$

Confinement Deep Dose Equivalent

Facility evacuation drills show that in the event of an evacuation, the confinement buidling can be evacuated within 5 minutes. Thererfore the projected dose from the gas release for individuals inside confinement when the fuel failure occurs is:

$$\left[\frac{2.5\,mrem}{DAC-hr}\right]\left[\frac{DAC\,Fraction}{1}\right]\left[\frac{5\min}{1}\right]\left[\frac{hr}{60\min}\right]=DDE\,(mrem)$$

For the isotopes of interest:

Confinement Air Immersion Dose					
Nuclide	Confinement Concentration (microCi / cc)	Occupational DAC (microCi / cc)	DAC Fraction	Immersion DAC (DAC-hr)	Deep Dose Equivalent mrem
I-131	3.06E-02	2.00E-08	1.53E+06	1.28E+05	3.19E+05
I-132	4.54E-02	3.00E-06	1.51E+04	1.26E+03	3.15E+03
I-133	6.87E-02	1.00E-07	6.87E+05	5.72E+04	1.43E+05
I-134	8.45E-02	2.00E-05	4.23E+03	3.52E+02	8.80E+02
I-135	6.76E-02	7.00E-07	9.66E+04	8.05E+03	2.01E+04
Kr-85m	1.37E-02	2.00E-05	6.87E+02	5.72E+01	1.43E+02
Kr-85	2.69E-04	1.00E-04	2.69E+00	2.24E-01	5.61E-01
Kr-87	2.64E-02	5.00E-06	5.28E+03	4.40E+02	1.10E+03
Kr-88	3.80E-02	2.00E-06	1.90E+04	1.58E+03	3.96E+03
Xe-131m	3.06E-02	4.00E-04	7.66E+01	6.38E+00	1.60E+01
Xe-133m	6.87E-02	1.00E-04	6.87E+02	5.72E+01	1.43E+02
Xe-133	6.87E-02	1.00E-04	6.87E+02	5.72E+01	1.43E+02
Xe-135m	6.76E-02	9.00E-06	7.51E+03	6.26E+02	1.57E+03
Xe-135	6.76E-02	1.00E-05	6.76E+03	5.63E+02	1.41E+03

The total DDE is the sum of the dose equivalents of each ofthe nuclides. Therefore the overall DDE due to being immersed for five minutes inside confinement is:

$\sum$ Individual DDE = 4.95 X 10^5 mrem

Site Boundary Deep Dose Equivalent

Assuming that an individual spends two hours at the site boundary, the deep dose equivalent would be the sum of the dose equivalents from eachof the isotopes of interest.

For the isotopes of interest:

Site Boundary Air Immersion Dose					
Nuclide	Site Boundary Concentration (microCi / cc)	Occupational DAC (microCi / cc)	DAC Fraction	Immersion DAC (DAC-hr)	Deep Dose Equivalent mrem
I-131	2.02E-07	2.00E-08	1.01E+01	2.02E+01	5.05E+01
I-132	3.00E-07	3.00E-06	9.99E-02	2.00E-01	4.99E-01
I-133	4.53E-07	1.00E-07	4.53E+00	9.06E+00	2.26E+01
I-134	5.57E-07	2.00E-05	2.79E-02	5.57E-02	1.39E-01
I-135	4.46E-07	7.00E-07	6.37E-01	1.27E+00	3.18E+00
Kr-85m	3.02E-04	2.00E-05	1.51E+01	3.02E+01	7.55E+01
Kr-85	5.92E-06	1.00E-04	5.92E-02	1.18E-01	2.96E-01
Kr-87	5.81E-04	5.00E-06	1.16E+02	2.32E+02	5.81E+02
Kr-88	8.36E-04	2.00E-06	4.18E+02	8.36E+02	2.09E+03
Xe-131m	6.73E-04	4.00E-04	1.68E+00	3.37E+00	8.42E+00
Xe-133m	1.51E-03	1.00E-04	1.51E+01	3.02E+01	7.55E+01
Xe-133	1.51E-03	1.00E-04	1.51E+01	3.02E+01	7.55E+01
Xe-135m	1.49E-03	9.00E-06	1.65E+02	3.30E+02	8.26E+02
Xe-135	1.49E-03	1.00E-05	1.49E+02	2.97E+02	7.43E+02

Therefore, if a member of the general public remains at the site boundary for two hours, they will receive a dose of:

$\sum$ Individual DDE = $4.55 \text{ X } 10^3$ mrem

Use of the DAC to determine the Committed Dose Equivalent to the Thyroid (CDE)

The ALI and DAC for any given nuclide can be found in 10 CFR 20 Appendix B. For I-131, the inhalation values for occupational exposure are given to be:

A. ALI = 50 μCi

This means that an intake of 50 μCi of I-131 will lead to a CEDE of 5 Rem, or 50 Rem to any individual tissue or organ. Since iodine concentrates in the thyroid, the ALI is based on a dose of 50 Rem to the thyroid.

B. DAC = $2 \text{ X } 10^{-8}$ μCi / cm^3

This means that if an individual inhales concentration of $2 \text{ X } 10^{-8}$ μCi / cm^3 I-131 at a rate of 1.2 m^3 per hour for 2000 hours, they would intake enough of the radionuclide to receive a CEDE of 5 Rem whole body, or 50 Rem to any individual tissue or organ:

$$\left[\frac{2X10^{-8}\,\mu Ci}{cm^3}\right]\left[\frac{1.2\,m^3}{hr}\right]\left[\frac{(100\,cm)^3}{m^3}\right]\left[\frac{2000\,hr}{1}\right]=48\,\mu Ci\approx 50\,\mu Ci$$

If an individual is immersed in an air concentration equivalent to one DAC, the average dose rate that they would be receiving would be:

A. Whole Body:

$$\left[\frac{5rem}{2000hr}\right]=2.5mrem/DAC-hr$$

B. Individual Tissue or Organ:

$$\left[\frac{50rem}{2000hr}\right]=25mrem/DAC-hr$$

We can express the concentration of RAM in air as a fraction or multiple of the DAC:

DAC Fraction (Multiple) = (Air Concentration) / (DAC)

If we had a concentration of 3.06×10^{-6} µCi / cm^3 of I-131 in the confinement air, the fraction (multiple) of the DAC for I-131 if the occupational DAC were 2×10^{-8} µCi / cm^3 would be:

DAC Fraction (Multiple) = (Air Concentration) / (DAC)

= (3.06×10^{-6} µCi / cm^3) / (2×10^{-8} µCi / cm^3)

= 153 DAC

If we know the concentration of RAM in air as a fraction or multiple of the DAC, then we can determine the dose rate that that an individual immersed in the air would recieve. Continuing with the I-131 example:

A. Since the ALI for I-131 is based on a committed thyroid dose of 50 rem per year, the dose rate associated with an air concentration of one DAC is 25 mrem / DAC - hr.

B. Therefore the committed dose to the thyroid is:

$$\left[\frac{25mrem}{DAC-hr}\right]\left[\frac{153DAC}{1}\right]=3825mrem/hr$$

If there is more than one nuclide in the air with the same dose rate associated with exposure (either whole body or individual organ), then the DAC fractions can be added together before determining the dose rate. Consider:

A. Suppose that the air has a concentration of 3.06×10^{-6} µCi / cm^3 of I-131, and 4.54×10^{-6} µCi / cm^3 of I-132 in it. The DAC fractions are:

(Air Concentration) / (DAC)

Where the DAC is defined in 10 CFR 20 for each isotope

B. For I-131 the DAC fraction has been previously calculated to be 153 DAC.

C. For I-132, given that the DAC is 3 X 10^{-6} µCi / cm^3, the DAC fraction is:

(4.54 X 10^{-6} µCi / cm^3) / (3 X 10^{-6} µCi / cm^3) = 1.51 DAC

D. Therefore the total DAC fraction is:

Total DAC Fraction = 153 DAC + 1.51 DAC = 154.51 DAC

E. For the iodines, the committed dose to the thyroid is also dependent on the amount of time that the individual is immersed. If an individual were only in the concentration of iodine for 5 minutes (0.083 hr), then the DAC fraction can be reduced:

Immersion DAC = (DAC)(Immersion Time)

Immersion DAC = (154.51 DAC)(0.083 hr) = 12.82 DAC - hr

F. Both of these DACs are based on a committed thyroid dose of 50 rem per year, which means that the dose rate associated with an air concentration of one DAC is 25 mrem / DAC - hr.

E. Therefore the committed dose to the thyroid for an individual that is immersed for 5 minutes in air with a concentration of 3.06 X 10^{-6} µCi / cm^3 of I-131 and 4.54 X 10^{-6} µCi / cm^3 of I-132 would be:

$$\left[\frac{25\,mrem}{DAC-hr}\right]\left[\frac{12.82\,DAC-hr}{1}\right]=320.5\,mrem$$

Confinement Committed Dose to the Thyroid (CDE)

Halogens are an inhalation hazard because they are absorbed into the body. The halogen of interest in the case of a fuel failure is iodine. Iodine concentrates in the thyroid. Consequently, the DAC for each isotope of Iodine is based on the amount of isotope that will result in a 50 rem dose to the thyroid over a 2000 hour year. We have calculated that an individual immersed in air with a concentration of RAM in it equivalent to one DAC would lead to an external dose rate of 25 mrem / hr to the thyroid.

The DAC fractions associated with the iodine nuclides in confinement are:

Nuclide	Release to Confinement Air µCi	Confinement Air Concentration µCi / cm³	DAC µCi / cm³	DAC Fraction
I-131	1.58E+08	3.06E-02	2.00E-08	1.53E+06
I-132	2.34E+08	4.54E-02	3.00E-06	1.51E+04
I-133	3.54E+08	6.87E-02	1.00E-07	6.87E+05
I-134	4.35E+08	8.45E-02	2.00E-05	4.23E+03
I-135	3.48E+08	6.76E-02	7.00E-07	9.66E+04

Since the doses associated with all of these nuclides are thyroid doses, we can sum the DAC fractions and multiply by 25 mrem / DAC – hr.

$\sum$ DAC Fractions = 2.33E+6 DAC

Dose Rate = (by 25 mrem / DAC – hr)(2.33E+6 DAC) = 5.83E+7 mrem / hr

Facility evacuation drills show that in the event of an evacuation, the confinement buidling can be evacuated within 5 minutes. Thererfore the projected dose to the thyroid from the iodine release for individuals inside confinement when the fuel failure occurs is:

$$\left[\frac{5.83\,X\,10^{7}\,mrem}{hr}\right]\left[\frac{5\,min}{1}\right]\left[\frac{hr}{60\,min}\right] = 4.86\,X\,10^{6}\,mrem$$

Site Boundary Committed Dose to the Thyroid (CDE)

The DAC fractions associated with the iodine nuclides at the site boundary are:

Nuclide	Confinement Air Concentration µCi / cm³	DAC µCi / cm³	DAC Fraction
I-131	2.02E-07	2.00E-08	1.01E+01
I-132	3.00E-07	3.00E-06	9.99E-02
I-133	4.53E-07	1.00E-07	4.53E+00
I-134	5.57E-07	2.00E-05	2.79E-02
I-135	4.46E-07	7.00E-07	6.37E-01

Since the doses associated with all of these nuclides are thyroid doses, we can sum the DAC fractions and multiply by 25 mrem / DAC – hr.

$\sum$ DAC Fractions = 1.54 X 10^1 DAC

Dose Rate = (by 25 mrem / DAC – hr)(1.54 X 10^1 DAC) = 3.85 X 10^2 mrem / hr

Thererfore the projected dose to the thyroid from the iodine release from fuel failure to individuals who are at the site boundary for two hours is:

$$\left[\frac{3.85\ X\ 10^{2}\ mrem}{hr}\right]\left[\frac{2\ hr}{1}\right] = 7.70\ X\ 10^{2}$$

Determination of the Committed Effective Dose Equivalent (CEDE):

The committed effective dose equivalent (CEDE) can be calculated by determining the amount of RAM uptake, and applying a CEDE factor. The amount of uptake is determined by assuming that the individual breaths 1.2 m^3 per hour and multiplying the volume of air inhaled by the concentration of RAM. The CEDE factors are taken from the "Effective" column on Table 2.1 in EPA-520/1-88-020.

Continuing with the I-131 example of an idividual that is immersed for 5 minutes in air with a concentration of 3.06 X 10^{-6} μCi / cm^3 of I-131, and breathing at a rate of 1.2 m^3 per hour:

A. The volume of air that an idividual that is breathing at a rate of 1.2 m^3 per hour intakes is:

$$\left[\frac{1.2m^3}{hr}\right]\left[\frac{5\min}{1}\right]\left[\frac{hr}{60\min}\right]\left[\frac{(100cm)^3}{m^3}\right]=1X10^5\,cm^3$$

B. The amount of I-131 intake is assumed to be all of the I-131 that has been breathed in. Therefore, the intake is the product of the concentration of I-131 in the air, and the volume of air taken in:

(3.06 X 10^{-6} μCi / cm^3 of I-131)(1 X 10^5 cm^3 air taken in) = 0.306 μCi I-131

C. The CEDE per μCi I-131 intake is taken from the "Effective" column in Table 2.1 in EPA-520/1-88-020 to be 33 mrem / μCi. Therefore the CEDE to the thyroid due to I-131 for this individual would be:

(0.306 μCi I-131)(33 mrem / μCi I-131) = 10.1 mrem

Confinement Committed Effective Dose Equivalent (CEDE):

Confinement Internal Dose							
Nuclide	**Confinement Concentration (microCi / cc)**	**Occupational DAC (microCi / cc)**	**DAC Fraction**	**Evacuation DAC (DAC-hr)**	**Intake (microCi)**	**CEDE per microCi**	**CEDE (mrem)**
I-131	3.06E-02	2.00E-08	1.53E+06	1.28E+05	3.06E+03	33	1.01E+05
I-132	4.54E-02	3.00E-06	1.51E+04	1.26E+03	4.54E+03	0.011	5.00E+01
I-133	6.87E-02	1.00E-07	6.87E+05	5.72E+04	6.87E+03	5.85	4.02E+04
I-134	8.45E-02	2.00E-05	4.23E+03	3.52E+02	8.45E+03	0.13	1.10E+03
I-135	6.76E-02	7.00E-07	9.66E+04	8.05E+03	6.76E+03	1.23	8.32E+03

The total CEDE is the sum of the dose equivalents for each of the individual isotopes:

$\sum$ Individual CEDE = 1.51×10^5 mrem

Site Boundary Committed Effective Dose Equivalent (CEDE):

Site Boundary Internal Dose							
Nuclide	Site Boundary Concentration (microCi / cc)	Occupational DAC (microCi / cc)	DAC Fraction	Evacuation DAC (DAC-hr)	Intake (microCi)	CEDE per microCi	CEDE (mrem)
I-131	2.02E-07	2.00E-08	1.01E+01	2.02E+01	2.02E-02	33	6.67E-01
I-132	3.00E-07	3.00E-06	9.99E-02	2.00E-01	3.00E-02	0.011	3.30E-04
I-133	4.53E-07	1.00E-07	4.53E+00	9.06E+00	4.53E-02	5.85	2.65E-01
I-134	5.57E-07	2.00E-05	2.79E-02	5.57E-02	5.57E-02	0.13	7.25E-03
I-135	4.46E-07	7.00E-07	6.37E-01	1.27E+00	4.46E-02	1.23	5.48E-02

The total CEDE is the sum of the dose equivalents for each of the individual isotopes:

$\sum$ Individual CEDE = 9.94×10^{-1} mrem

Determination of the Total Effective Dose Equivalent (TEDE):

The total effective dose equivalent (TEDE) is the sum of the dose due to external sources (DDE) and internal sources (CEDE):

TEDE = DDE + CEDE

Confinement Total Effective Dose Equivalent (TEDE):

TEDE = 4.95×10^5 mrem + 1.51×10^5 mrem = 6.46×10^5 mrem

Site Boundary Total Effective Dose Equivalent (TEDE):

TEDE = 4.55×10^3 mrem + 9.94×10^{-1} mrem = 4.55×10^3 mrem

Conclusion

10 CFR 20 provides radiation dose limits to radiation workers, and to the general public. For radiation worker, the limits are:

50 rem / yr to an individual organ (CDE)

5 rem / yr whole body (TEDE)

For members of the general public, the limits are:

100 mrem / yr

The doses that an individual is predicted to receive due to a fuel failure event in which the core has reached saturation activity is based on the following assumptions:

A. Individuals inside confinement recognize the problem and evacuate within five minutes.
B. Individuals that are exposed outside confinement remain at the site boundary for two hours.

A summary of the predicted doses, and any regulatory limit associated with the dose is:

Dose Summary		Dose Limits	
Confinement Dose		Occupational Limits	General Public Limits
Committed Dose to Thyroid	4.86E+06 mrem	50 rem / yr	
CEDE	1.51E+05 mrem		
Immersion	4.95E+05 mrem		
TEDE	6.46E+05 mrem	5 rem / yr	100 mrem / yr
Site Boundary Dose			
Committed Dose to Thyroid	7.70E+02 mrem	50 rem / yr	
CEDE	9.94E-01 mrem		
Immersion	4.55E+03 mrem		
TEDE	4.55E+03 mrem	5 rem / yr	100 mrem / yr

Therefore, for radiation worker inside confinement in which the saturation activity has been reached with 12.5 grams of fissionable material is released to confinement, the predicted doses that would be expected if it took five minutes to evacuate confinement would be:

CDE to Thyroid = 4860 Rem

TEDE = 646 REM

Both of these are significantly above the regulatory limits. Scaling down to the regulatory limits:

CDE to Thyroid:

$$Scale\ Factor = \left[\frac{Regulatory\ Limit}{Predicted\ Dose}\right]$$

$$= \left[\frac{50 \ Rem \ per \ Year}{4860 \ Rem}\right] = 0.01$$

This suggests that the limit on the quantity of fissionable material in experiments should be:

(12.5 grams fissionable material)(0.01) = 0.125 grams fissionable material

TEDE

$$Scale \ Factor = \left[\frac{Regulatory \ Limit}{Predicted \ Dose}\right]$$

$$= \left[\frac{5 \ Rem \ per \ Year}{646 \ Rem}\right] = 0.0077$$

This suggests that the limit on the quantity of fissionable material in experiments should be:

(12.5 grams fissionable material)(0.0077) = 0.09625 grams fissionable material

For the general public at the site boundary, assuming a two hour occupancy time:

TEDE

$$Scale \ Factor = \left[\frac{Regulatory \ Limit}{Predicted \ Dose}\right]$$

$$= \left[\frac{100 \ mrem \ per \ Year}{4550 \ mrem}\right] = 0.022$$

This suggests that the limit on the quantity of fissionable material in experiments should be:

(12.5 grams fissionable material)(0.022) = 0.275 grams fissionable material

Of all of these, the most restrictive amount is 0.09625 grams of fissionable material.

Fissionable Experiment Back Check:

Assumptions

1. Assume that an experiment with 0.09625 grams of fissionable material has been irradiated long enough for the fission products in it to be at saturation levels.

2. Assume that when this experment malfunctions, all of the fission products are relased to confinement.

Source Term

0.09625 grams of fissionable material represents 0.77 % of the amount of fissionable material in a fuel plate. Therefore, the experiment activity is:

(0.0077)(activity in a single plate)

Using the single plate activities that were calculated for the fuel failure accident analysis, the activities for the isotopes of interest in the fissionable experiment are:

Source Term						
Nuclide	Decay Constant (/s)	Fission Yield (Atoms/Fission)	Core Activity (Bq)	Core Activity (Ci)	Single Plate Activity (Ci)	Experiment Activity (Ci)
I-131	9.73E-07	0.029	1.80E+15	4.86E+04	1.58E+02	1.21E+00
I-132	8.02E-05	0.043	2.67E+15	7.21E+04	2.34E+02	1.80E+00
I-133	9.25E-06	0.065	4.03E+15	1.09E+05	3.54E+02	2.72E+00
I-134	2.20E-04	0.08	4.96E+15	1.34E+05	4.35E+02	3.35E+00
I-135	2.88E-05	0.064	3.97E+15	1.07E+05	3.48E+02	2.68E+00
Kr-85m	4.41E-05	0.013	8.06E+14	2.18E+04	7.07E+01	5.45E-01
Kr-85	2.05E-09	0.000255	1.58E+13	4.27E+02	1.39E+00	1.07E-02
Kr-87	1.48E-04	0.025	1.55E+15	4.19E+04	1.36E+02	1.05E+00
Kr-88	6.95E-05	0.036	2.23E+15	6.03E+04	1.96E+02	1.51E+00
Xe-131m	6.67E-07	0.029	1.80E+15	4.86E+04	1.58E+02	1.21E+00
Xe-133m	3.50E-06	0.065	4.03E+15	1.09E+05	3.54E+02	2.72E+00
Xe-133	1.53E-06	0.065	4.03E+15	1.09E+05	3.54E+02	2.72E+00
Xe-135m	7.40E-04	0.064	3.97E+15	1.07E+05	3.48E+02	2.68E+00
Xe-135	2.11E-05	0.064	3.97E+15	1.07E+05	3.48E+02	2.68E+00

Quantitiy of RAM that Reaches Confinement Air

Since it is not postulated that the experiment would neccessarily be under water, the reduction factors that were used in the fuel failure analysis are not used. Consequently it is assumed that 100% of the activity generated reaches confinemetn air.

Conenctration of RAM in the Confinement Air

The free volume of the confinement room was determined to be:

5.15×10^9 cm^3

Therefore, spreading the isotope activities evenly over this volume of air provkdes the following concentrations:

Release to Confinement Air				
Nuclide	Experiment Activity (Ci)	Release to Confinement (Ci)	Release to Confinement (microCi)	Confinement Concentration (microCi / cc)
I-131	1.21E+00	1.21E+00	1.21E+06	2.36E-04
I-132	1.80E+00	1.80E+00	1.80E+06	3.50E-04
I-133	2.72E+00	2.72E+00	2.72E+06	5.29E-04
I-134	3.35E+00	3.35E+00	3.35E+06	6.51E-04
I-135	2.68E+00	2.68E+00	2.68E+06	5.21E-04
Kr-85m	5.45E-01	5.45E-01	5.45E+05	1.06E-04
Kr-85	1.07E-02	1.07E-02	1.07E+04	2.07E-06
Kr-87	1.05E+00	1.05E+00	1.05E+06	2.03E-04
Kr-88	1.51E+00	1.51E+00	1.51E+06	2.93E-04
Xe-131m	1.21E+00	1.21E+00	1.21E+06	2.36E-04
Xe-133m	2.72E+00	2.72E+00	2.72E+06	5.29E-04
Xe-133	2.72E+00	2.72E+00	2.72E+06	5.29E-04
Xe-135m	2.68E+00	2.68E+00	2.68E+06	5.21E-04
Xe-135	2.68E+00	2.68E+00	2.68E+06	5.21E-04

Emergency Filter

The emergency filter has no effect on the noble gases, but removes 99.97% of the iodines.

Concentration of RAM in the Emergency Filter Air Exhaust

Release to Stack			
Nuclide	Confinement Concentration (microCi / cc)	Emergency Air Filter Release Fraction	Emergency Air Filter Exhaust Concentration (microCi / cc)
I-131	2.36E-04	0.03%	7.08E-08
I-132	3.50E-04	0.03%	1.05E-07
I-133	5.29E-04	0.03%	1.59E-07
I-134	6.51E-04	0.03%	1.95E-07
I-135	5.21E-04	0.03%	1.56E-07
Kr-85m	1.06E-04	100%	1.06E-04
Kr-85	2.07E-06	100%	2.07E-06
Kr-87	2.03E-04	100%	2.03E-04
Kr-88	2.93E-04	100%	2.93E-04
Xe-131m	2.36E-04	100%	2.36E-04
Xe-133m	5.29E-04	100%	5.29E-04
Xe-133	5.29E-04	100%	5.29E-04
Xe-135m	5.21E-04	100%	5.21E-04
Xe-135	5.21E-04	100%	5.21E-04

RAM Release Rate

Empirical data has been used to show that the average flow rate through the emergency filter is:

$4.97 \text{ X } 10^5 \text{ cm}^3 / \text{s}$

Therefore, the RAM release rate for each isotope of interest is:

(Emergency Filter Air Exhaust Concentration)(Emergency Filter Flow Rate)

Overall:

RAM Release Rate		
Nuclide	**Emergency Air Filter Exhaust Concentration (microCi / cc)**	**Stack RAM Release Rate (microCi / s)**
I-131	7.08E-08	3.52E-02
I-132	1.05E-07	5.22E-02
I-133	1.59E-07	7.88E-02
I-134	1.95E-07	9.70E-02
I-135	1.56E-07	7.76E-02
Kr-85m	1.06E-04	5.26E+01
Kr-85	2.07E-06	1.03E+00
Kr-87	2.03E-04	1.01E+02
Kr-88	2.93E-04	1.46E+02
Xe-131m	2.36E-04	1.17E+02
Xe-133m	5.29E-04	2.63E+02
Xe-133	5.29E-04	2.63E+02
Xe-135m	5.21E-04	2.59E+02
Xe-135	5.21E-04	2.59E+02

Atmospheric Dispersion

The concentration at the site boundary was determined with the following assumptions:

1. Release is at ground level
2. Conditions are Pasquill Type F
3. Wind speed is one meter per second
4. Wind direction is constant over the entire duration of the release

Based on these assumptions, the dispersion factor $(X/Q)_{eff}$ was found to be:

$(X/Q)_{eff} = 4.43 \text{ X } 10^{-8} \text{ s / cm}^3$

The site boundary concnetrations of each isotope of interest isfound by:

$(X/Q)_{eff}$ * (Isotope Release Rate μCi / s)

Overall:

Site Boundary Concentration			
Nuclide	**Emergency Air Filter Exhaust Concentration (microCi / cc)**	**Stack Release Rate (microCi / s)**	**Site Boundary Concentration (microCi / cc)**
I-131	7.08E-08	3.52E-02	1.56E-09
I-132	1.05E-07	5.22E-02	2.31E-09
I-133	1.59E-07	7.88E-02	3.49E-09
I-134	1.95E-07	9.70E-02	4.29E-09
I-135	1.56E-07	7.76E-02	3.43E-09
Kr-85m	1.06E-04	5.26E+01	2.32E-06
Kr-85	2.07E-06	1.03E+00	4.56E-08
Kr-87	2.03E-04	1.01E+02	4.47E-06
Kr-88	2.93E-04	1.46E+02	6.44E-06
Xe-131m	2.36E-04	1.17E+02	5.19E-06
Xe-133m	5.29E-04	2.63E+02	1.16E-05
Xe-133	5.29E-04	2.63E+02	1.16E-05
Xe-135m	5.21E-04	2.59E+02	1.14E-05
Xe-135	5.21E-04	2.59E+02	1.14E-05

Confinement Deep Dose Equivalent

The DAC is the air concentration at which an individual would receive a deep dose equivalent of 5 rem whole body if they were immersed in for 2000 hours. This is defined in 10 CFR 20 for each relevant isotope.

The DAC fraction is:

Isotope Concentration / DAC

The DAC – Hour is:

DAC Fraction * Exposure Time in Hours

By definition, that DAC represents the concentration that would cause one to receive a deep dose equivalent of:

$$\left[\frac{5\ Rem}{2000\ hr}\right] = \frac{2.5\ mrem}{DAC - hr}$$

Therefore the DDE due to immersion in a radioactive gas can be found by:

$$DDE = \left[\frac{2.5\ mrem}{DAC - hr}\right] [DAC\ Fraction]$$

The DDE can be calculated for each isotope of interest, and summed to get the total DDE. Assuming that it takes five minutes for people inside confinement to evacuate after the experiment fails, this works out to be:

Confinement Air Immersion Dose					
Nuclide	Confinement Concentration (microCi / cc)	Occupational DAC (microCi / cc)	DAC Fraction	Immersion DAC (DAC-hr)	Deep Dose Equivalent mrem
I-131	2.36E-04	2.00E-08	1.18E+04	9.83E+02	2.46E+03
I-132	3.50E-04	3.00E-06	1.17E+02	9.72E+00	2.43E+01
I-133	5.29E-04	1.00E-07	5.29E+03	4.41E+02	1.10E+03
I-134	6.51E-04	2.00E-05	3.25E+01	2.71E+00	6.78E+00
I-135	5.21E-04	7.00E-07	7.44E+02	6.20E+01	1.55E+02
Kr-85m	1.06E-04	2.00E-05	5.29E+00	4.41E-01	1.10E+00
Kr-85	2.07E-06	1.00E-04	2.07E-02	1.73E-03	4.32E-03
Kr-87	2.03E-04	5.00E-06	4.07E+01	3.39E+00	8.47E+00
Kr-88	2.93E-04	2.00E-06	1.46E+02	1.22E+01	3.05E+01
Xe-131m	2.36E-04	4.00E-04	5.90E-01	4.91E-02	1.23E-01
Xe-133m	5.29E-04	1.00E-04	5.29E+00	4.41E-01	1.10E+00
Xe-133	5.29E-04	1.00E-04	5.29E+00	4.41E-01	1.10E+00
Xe-135m	5.21E-04	9.00E-06	5.78E+01	4.82E+00	1.21E+01
Xe-135	5.21E-04	1.00E-05	5.21E+01	4.34E+00	1.08E+01

The total DDE due to immersion is:

3.81×10^3 mrem

Site Boundary Deep Dose Equivalent

Taking dilution into account, and assuming that an individual spends two hours at the site boundary, the DDE due to immersion is:

Site Boundary Air Immersion Dose					
Nuclide	Site Boundary Concentration (microCi / cc)	Occupational DAC (microCi / cc)	DAC Fraction	Immersion DAC (DAC-hr)	Deep Dose Equivalent mrem
I-131	1.56E-09	2.00E-08	7.78E-02	1.56E-01	3.89E-01
I-132	2.31E-09	3.00E-06	7.69E-04	1.54E-03	3.84E-03
I-133	3.49E-09	1.00E-07	3.49E-02	6.97E-02	1.74E-01
I-134	4.29E-09	2.00E-05	2.15E-04	4.29E-04	1.07E-03
I-135	3.43E-09	7.00E-07	4.90E-03	9.81E-03	2.45E-02
Kr-85m	2.32E-06	2.00E-05	1.16E-01	2.32E-01	5.81E-01
Kr-85	4.56E-08	1.00E-04	4.56E-04	9.12E-04	2.28E-03
Kr-87	4.47E-06	5.00E-06	8.94E-01	1.79E+00	4.47E+00
Kr-88	6.44E-06	2.00E-06	3.22E+00	6.44E+00	1.61E+01
Xe-131m	5.19E-06	4.00E-04	1.30E-02	2.59E-02	6.48E-02
Xe-133m	1.16E-05	1.00E-04	1.16E-01	2.32E-01	5.81E-01
Xe-133	1.16E-05	1.00E-04	1.16E-01	2.32E-01	5.81E-01
Xe-135m	1.14E-05	9.00E-06	1.27E+00	2.54E+00	6.36E+00
Xe-135	1.14E-05	1.00E-05	1.14E+00	2.29E+00	5.72E+00

The total DDE due to immersion is:

$3.50 \text{ X } 10^1$ mrem

Confinement Committed Dose to the Thyroid

If an individual is immersed in an air concentration equivalent to one DAC, the average dose rate that they would be receiving would be:

A. Whole Body:

$$\left[\frac{5rem}{2000hr}\right] = 2.5mrem/DAC-hr$$

B. Individual Tissue or Organ:

$$\left[\frac{50rem}{2000hr}\right] = 25mrem/DAC-hr$$

Therefore, by definition, that DAC tells us the concentration in which one would receive a CDE to an individual organ of 50 rem, if the individual were immersed for 200 hours. For this scenario, iodine concentrates in the thyroid so we are interested in the committed dose to the thyroid (Xe is inert, and therefore does not contribute to the CDE).

Consequently the CDE due to immersion in a radioactive gas can be found by:

$$CDE = \left[\frac{25\ mrem}{DAC - hr}\right][DAC\ Fraction]$$

The CDE can be calculated for each isotope of interest, and summed to get the total CDE. Assuming that it takes five minutes for people inside confinement to evacuate after the experiment fails, this works out to be:

Confinement CDE DAC Fractions				
Nuclide	Release to Confinement Air µCi	Confinement Air Concentration µCi / cm³	DAC µCi / cm³	DAC Fraction
I-131	1.21E+06	2.36E-04	2.00E-08	1.18E+04
I-132	1.80E+06	3.50E-04	3.00E-06	1.17E+02
I-133	2.72E+06	5.29E-04	1.00E-07	5.29E+03
I-134	3.35E+06	6.51E-04	2.00E-05	3.25E+01
I-135	2.68E+06	5.21E-04	7.00E-07	7.44E+02

Total CDE is:

$\sum$ DAC Fraction = 1.80E+4

Dose Rate = (by 25 mrem / DAC – hr)(1.80E+4) = 4.49E+5 mrem / hr

Occupancy is assumed to be 5 min = 0.0833 hr

CDE = (4.49E+5 mrem / hr) (0.0833 hr) = 3.74E+4 mrem

This means that the committed dose equivalent to the thyroid is 3.74E+4 mrem.

Site Boundary Committed Dose to the Thyroid

The stie boundary CDE, assuming that an individual is at the boundary for two hours is:

Site Boundary CDE DAC Fractions			
Nuclide	**Confinement Air Concentration μCi / cm^3**	**DAC μCi / cm^3**	**DAC Fraction**
I-131	1.56E-09	2.00E-08	7.78E-02
I-132	2.31E-09	3.00E-06	7.69E-04
I-133	3.49E-09	1.00E-07	3.49E-02
I-134	4.29E-09	2.00E-05	2.15E-04
I-135	3.43E-09	7.00E-07	4.90E-03

Total CDE is:

$\sum$ DAC Fraction = 1.19E-1

Dose Rate = (by 25 mrem / DAC – hr)(1.19E-1) = 2.96E+0 mrem / hr

Occupancy is assumed to be 2 hr

CDE = (2.96E+0 mrem / hr) (2 hr) = 5.93E+0 mrem

This means that the committed dose equivalent to the thyroid is 5.93 mrem.

Confinement Committed Effective Dose Equivalent

The Standard Man is estimated to have a breath intake rate of 1.2 m^3 per hour.

Therefore, the volume of air intake in a 5 min period is:

$$\left[\frac{1.2\,m^3}{hr}\right]\left[\frac{5\min}{1}\right]\left[\frac{hr}{60\min}\right]\left[\frac{(100\,cm)^3}{m^3}\right]=1X10^5\,cm^3$$

The intake of each isotope is assumed to be all of the isotope that was breathed in, which is the volume of air intake during the 5 minute period, multiplied by the concentration of the isotope:

$$[1\,X\,10^5\ cm^3\ Air\ Intake][Confinement\ Concentration\ of\ Isotope]$$

For the iodine isotopes of interest, this is:

Confinement Internal Dose							
Nuclide	Confinement Concentration (microCi / cc)	Occupational DAC (microCi / cc)	DAC Fraction	Evacuation DAC (DAC-hr)	Intake (microCi)	CEDE per microCi	CEDE (mrem)
I-131	2.36E-04	2.00E-08	1.18E+04	9.83E+02	2.36E+01	33	7.78E+02
I-132	3.50E-04	3.00E-06	1.17E+02	9.72E+00	3.50E+01	0.011	3.85E-01
I-133	5.29E-04	1.00E-07	5.29E+03	4.41E+02	5.29E+01	5.85	3.09E+02
I-134	6.51E-04	2.00E-05	3.25E+01	2.71E+00	6.51E+01	0.13	8.46E+00
I-135	5.21E-04	7.00E-07	7.44E+02	6.20E+01	5.21E+01	1.23	6.40E+01

The total CEDE is the sum of the individual CEDEs for each isotope:

$\sum$ Individual CEDE = 1.16 E+3 mrem

Site Boundary Committed Effective Dose Equivalent

Site Boundary Internal Dose							
Nuclide	Site Boundary Concentration (microCi / cc)	Occupational DAC (microCi / cc)	DAC Fraction	Evacuation DAC (DAC-hr)	Intake (microCi)	CEDE per microCi	CEDE (mrem)
I-131	1.56E-09	2.00E-08	7.78E-02	1.56E-01	1.56E-04	33	5.13E-03
I-132	2.31E-09	3.00E-06	7.69E-04	1.54E-03	2.31E-04	0.011	2.54E-06
I-133	3.49E-09	1.00E-07	3.49E-02	6.97E-02	3.49E-04	5.85	2.04E-03
I-134	4.29E-09	2.00E-05	2.15E-04	4.29E-04	4.29E-04	0.13	5.58E-05
I-135	3.43E-09	7.00E-07	4.90E-03	9.81E-03	3.43E-04	1.23	4.22E-04

The total CEDE is the sum of the individual CEDEs for each isotope:

$\sum$ Individual CEDE = 7.65 E-3 mrem

Confinement Total Effective Dose Equivalent (TEDE):

TEDE = DDE + CEDE

TEDE = 3.81 X 10^3 mrem + 1.16 E+3 mrem = 4.97 X 10^3 mrem

Site Boundary Total Effective Dose Equivalent (TEDE):

TEDE = 3.50 X 10^1 mrem + 7.65 E-3 mrem = 3.51 X 10^1 mrem

Dose Summary

Dose Summary			Dose Limits	
Confinement Dose			Occupational Limits	General Public Limits
	Committed Dose to Thyroid	3.74E+04 mrem	50 rem / yr	
	CEDE	1.16E+03 mrem		
	Immersion	3.81E+03 mrem		
	TEDE	4.97E+03 mrem	5 rem / yr	100 mrem / yr
Site Boundary Dose				
	Committed Dose to Thyroid	5.93E+00 mrem	50 rem / yr	
	CEDE	7.65E-03 mrem		
	Immersion	3.50E+01 mrem		
	TEDE	3.51E+01 mrem	5 rem / yr	100 mrem / yr

Conclusion

If we assume that an experiment with 0.09625 grams of fissionable material has been irradiated long enough for the fission products in it to be at saturation levels malfunctions, and that all of the fission products are relased to confinement, 10 CFR 20 limits will still be met if it takes 5 minutes to evacuate confinement, and personnel down wind at the site boundary are located there for 2 hours.