14.93 The proposed TS contain TS 3.4, 3.5, 3.6, "Confinement and Emergency Exhaust System and Emergency Power." The proposed TS is difficult to understand because it combines the requirements for three systems into one specification without clearly stating the requirements for each system. Explain the reason for combining all of these requirements into one specification, and explain the reason for the multiple numbers in the title of the TS. Revise the proposed TS to either separate the limiting conditions for operation (LCOs) for the three systems into three separate TS, or revise the proposed TS to clearly state the requirements for each of the three systems.

Fifth Response Submitted November 26, 2010

It is difficult to define the components that should be included as part of the confinement system versus those that should be included as part of the ventilation system because the ventilation blowers and filters are critical components of the confinement system. However, these systems have been broken apart in an attempt to make this section of the Technical Specifications follow the format outlined in ANSI 15.1. These specifications will be written as follows:

3.4 Confinement

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 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 0.65% dK/K is in progress.
5. Any experiment movement that could cause a reactivity change of more than 0.65% 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:

    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:

    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 is 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

3.6.1 Required Emergency Power Sources

Applicability:

This specification describes the emergency electrical power sources that are necessary in order to ensure that power is available to confinement system components that are necessary to ensure 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.

14.94 The "Applicability" and "Objective" sections of TS 3.4, 3.5, 3.6 mention fuel handling, handling of radioactive material, and any operation that could cause the spread of airborne radioactivity in the confinement area. The "Specification" section only contains requirements for reactor operation. Explain why the TS does not contain requirements for fuel handling, handling of radioactive material, and any operation that could cause the spread of airborne radioactivity in the confinement area. Revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

See the response to RAI 14.93. These Technical Specifications have been re-written to conform to ANSI 15.1. Section 3.4.1 addresses the conditions under which the confinement system is required to be operable.

14.95 The "Specification" section of TS 3.4, 3.5, 3.6 states, "the reactor shall not be operated unless the following equipment is operable and/or conditions met." Explain the reason for using the "and/or" condition in the specification, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

The intention of the "and/or" condition was to recognize that some of the items in the specification are equipment that must be operable, and that some of the items are conditions that must be met. The condition in the specification (P. 14-24 Lines 38-39) will be changed to "or" since each of the items listed is either equipment that must be operable, or a condition that must be met.

14.96 TS 3.4, 3.5, 3.6 does not appear to contain any requirements for normal ventilation during reactor operation, fuel handling, handling of radioactive material, and any operation that could cause the spread of airborne radioactivity in the confinement area. Explain why there are no requirements for normal ventilation, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

See the response to RAI 14.93. These Technical Specifications have been re-written to conform to ANSI 15.1. Section 3.5.1.1 specifies the normal operating mode ventilation components that must be operating in order to achieve normal mode confinement.

14.97 TS 3.4, 3.5, 3.6 does not appear to contain any requirements for ventilation flow rates for normal ventilation or the emergency exhaust system. Explain why there are no requirements for normal ventilation or the emergency exhaust system flow rates, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

See the response to RAI 14.93. These Technical Specifications have been re-written to conform to ANSI 15.1. The specifications that are related to ventilation flow rates are:

Specification 3.4.3.1.6:

This specification requires that the differential pressure across confinement be at least 0.5 inches of water under normal confinement / ventilation conditions. No flow rate is specified because the determination of whether or not sufficient confinement exists is based on this differential pressure.

Specification 3.4.3.2.6:

This specification requires that the differential pressure across confinement be at least 0.5 inches of water under emergency confinement / ventilation conditions. No flow rate is specified because the determination of whether or not sufficient confinement exists is based on this differential pressure.

Specification 3.5.2.2:

This specification sets a limit on the maximum emergency ventilation flow rate. The maximum flow rate is limited to 1500 cfm.

14.98 TS 3.4, 3.5, 3.6 requires the emergency cleanup exhaust system to be operable during reactor operation, but does not specify what constitutes operability of the system. Explain what constitutes operability of the emergency cleanup exhaust system (e.g., minimum required equipment, filtration requirements, etc.), and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

See the response to RAI 14.93. These Technical Specifications have been re-written to conform to ANSI 15.1. The specifications that are related to the operability of the emergency cleanup exhaust system are:

Specification 3.4.2.2:

This section specifies the confinement system components that must be operable in order for the emergency cleanup exhaust system to be operable.

Specification 3.4.3.2:

This section specifies the conditions that are required in order to achieve emergency confinement.

Specification 3.5.2:

This section specifies the ventilation system components that must be operable in order for the emergency cleanup exhaust system to be operable.

Specification 3.5.2.3:

This section specifies the emergency filter efficiency that is required in order for the emergency exhaust system to be operable.

14.99 TS 3.4, 3.5, 3.6 requires that the function of the emergency generator is "to insure power systems and other designated systems." To what "power system" does this refer? What are the "other designated systems" referenced in the function statement? Explain the reason for not specifying what equipment is required to be powered by the emergency generator. Explain why there are no LCOs regarding what constitutes operability of the emergency generator (e.g., type of generator, minimum operating time, etc.), and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

See the response to RAI 14.93. These Technical Specifications have been re-written to conform to ANSI 15.1. The verbiage has been changed to make it clear that the function of the emergency power system is to insure that the confinement system will be capable of performing its intended function in the event of a facility power failure.

Specification 3.6.2 describes the confinement / ventilation system components that are required to be supplied with emergency power.

The LCO for the emergency power supply is that it is capable of supplying the emergency and dilution blowers with enough power that they are capable of operating in the event of a facility power failure. The power source is not specified in the Technical Specifications so that it will be possible to replace the generator that is currently used if need be, without modifying the RINSC Technical Specifications.

14.100 The bases for TS 3.4, 3.5, 3.6 appear to only contain bases for operation of the emergency exhaust system. Provide bases for normal operation of the confinement and the requirements for emergency power.

Fifth Response Submitted November 26, 2010

The emergency power system is unrelated to the normal operation of the confinement system. Under normal conditions, the differential pressure that controls the confinement air pathway is generated by the confinement exhaust blower. This blower is not supplied with emergency power.

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. Confinement is only maintained if the emergency confinement system is turned on.

Loss of facility power also causes the lighting in confinement to shut off. Except for minimal emergency lighting that switches on when a power failure is detected, there is no other lighting in confinement during facility power outages. Consequently, there is no reasonable opportunity to continue any of the activities that would have the potential to cause an airborne release of radioactivity during a facility power outage. If it were absolutely necessary for operations that require confinement to continue, confinement would have to be provided via the emergency system.

14.101 ANSI/ANS-15.1 recommends technical specifications include the minimum number, type, and location of required environmental radiation monitors. Section 11.1.7 of the SAR discusses environmental monitoring at the RINSC. Explain the reason for not including such requirements, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

As noted in the comment, ANSI 15.1 recommends that the technical specifications include the minimum number, type and location of required environmental radiation monitors. In our view, the key phrase in the ANSI standard is its referral to "required environmental monitors" presupposing that the safety analysis has determined the need for such monitors. Our safety analysis did not establish a specific need for such monitors. In addition, it is unclear as to which technical specification is referred to by this comment. Section 11.1.7 is not a technical specification but merely describes a monitoring program using integrating dosimeters that has been in existence for over twenty years. The results from that monitoring are included in our annual report to the NRC to show compliance with basic 10 CFR 20 requirements. Technical Specification 6.8.4.e. already requires "a description of any environmental surveys performed outside the facility." Since this set of comments does not include administrative controls, e.g., annual reporting requirements, it is unclear as to where the suggested technical specification would go. RINSC has an operating history of over forty years that suggests that the current environmental monitoring system is sufficient, is documented and shared with the NRC annually and that no additional technical specification is needed.

14.102 The "Applicability" section of TS 3.7.1 mentions fuel movement and handling of radioactive materials in the reactor building, but the specification only specifies requirements for reactor operation. Explain why there are no requirements for radiation monitoring systems during fuel movement and handling of radioactive materials in the reactor building, and revise the proposed TS as appropriate.

Seventh Response Submitted December 14, 2010

The facility radiation monitoring system is described in SAR Section 7.2.15 and summarized in Table 3.2 of the technical specifications. The radiation monitoring system is powered and "on" all of the time. Thus, it is unnecessary to have separate requirements for fuel movement or handling radioactive materials.

14.103 TS 3.7.1.1 states, "The particulate activity monitor and the gaseous activity monitor for the facility exhaust stack shall be operating." TS 3.2.1, Table 3.2, item 5 only requires one building air gaseous exhaust (stack) monitor, and does not require a separate particulate monitor. Explain this apparent inconsistency, and revise the proposed TS as appropriate. (See RAI 14.78)

Fourth Response Submitted September 8, 2010

See the answer to RAI question 14.78.

14.104 TS 3.7.1.1 states, "The particulate activity monitor and the gaseous activity monitor for the facility exhaust stack shall be operating." This statement does not specify when the monitors are required to be operating. Explain why the TS does not specify when the monitors are required to be operating, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

This comment appears to be taken out of context. Specification 3.7.1.1 reads: "When the reactor is operating, gaseous and particulate sampling of the stack effluent shall be monitored by a stack monitor with a readout in the control room. The particulate activity monitor and the gaseous activity monitor for the facility exhaust stack shall be operating. If either unit is out of service for more than one shift (6 hours), either the reactor shall be shut down or the unit shall be replaced by one of comparable monitoring capability." It is clear that the monitor is required during normal operation of the reactor. If, for some reason, either monitor is out of service for more than six hours, we either shut the reactor down or replace the defective monitor with a comparable one.

14.105 TS 3.7.1.1 specifies that the reactor may be operated for up to 6 hours without either a particulate activity monitor or a gaseous activity monitor. Explain the basis for operating the reactor for 6 hours without particulate effluent activity detection capability. Explain the basis for operating the reactor for 6 hours without gaseous effluent activity detection capability.

Second Response Submitted August 6, 2010

The Basis for operating the reactor for up to 6 hours without a gaseous activity monitor is covered in the answer to RAI Question 14.80.

14.106 TS 3.7.1.2 allows the reactor to be operating for up to 6 hours without the continuous air monitoring unit required by TS 3.2.1, Table 3.2, item 11. TS 3.2.1 states that the reactor shall not be made critical unless the unit is operating. Explain this apparent inconsistency between TS 3.7.1.2 and TS 3.2.1, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

The initial idea behind this apparent inconsistency was that the reactor would not be started unless the stack gaseous and particulate air monitoring systems were both working at the time of the reactor checkout. If one of the systems failed during operation, it could be replaced with another instrument for up to 6 hours. This allows the staff six hours to work on getting the failed system back in operation before having to suspend reactor operations.

14.107 The footnote to TS 3.7.1.3 states that the reactor may be operated in a steady-state power mode if an area radiation monitor or the reactor bridge radiation monitor is replaced with a portable gamma-sensitive monitor with its own alarm. How long can the reactor be operated with a portable monitor performing the function of an area radiation monitor, and why? How does the portable instrument notify the reactor operator of changing radiation conditions? Given that TS 3.2.1, Table 3.2 only requires one operating radiation monitor on the experimental level, explain how a single portable monitor provides adequate detection capability to monitor radiation conditions on the entire experimental level.

Fifth Response Submitted November 26, 2010

The practice has been to limit reactor operation to one shift when a portable monitor is performing the function of an area radiation monitor on the experimental level. This allows time to repair or replace the defective radiation monitor without immediately shutting down the reactor. It is apparent that the comment fails to consider the earlier discussion in Technical Specification 3.2. The note (b) in Table 3.2 states that the reactor cannot be continuously operated without a minimum of one radiation monitor on the experimental level of the

reactor building and one monitor over the reactor pool operating and capable of warning personnel of high radiation areas. If one looks carefully at the asterisk, one would note that the radiation monitors are on the reactor bridge, next to the fuel safe and at the thermal column. Since the reactor bridge and fuel safe are on level 5 while the thermal column is on level 3, there is only one radiation monitor on the experimental level normally, i.e., the one by the thermal column. This configuration has existed for many years and the safety analysis did not identify a need for any additional monitoring. Based on the safety analysis in Chapter 13, the critical area radiation monitor is the one on the reactor bridge since it is the first warning of a fuel failure (MHA). This technical specification will be changed and updated when the technical issues relating to the MHA have been resolved.

14.108 The "Bases" section of TS 3.7.1 does not provide bases for the stack effluent monitors. Provide bases for the stack effluent monitors.

Fifth Response Submitted November 26, 2010

The "Bases" portion of the specification should read: "A continuing evaluation of contamination levels within the reactor building will be made to prevent airborne gaseous and particulate radioactivity from reaching the derived air concentration levels in 10 CFR 20 and to assure the safety of personnel. A continuing evaluation of gaseous and particulate activity in the facility exhaust will be made to assure that airborne effluent releases remain within 10 CFR 20 limits offsite. This is accomplished by the monitoring systems described in Table 3.2."

14.109 The "Objective" section of TS 3.7.2.a states, "To assure containment integrity is maintained during reactor operation..." Explain what "containment integrity" means. Explain how TS 3.7.2 "assures containment integrity."

Second Response Submitted August 6, 2010

The "Applicablity" section of TS 3.7.2.a has a typo (P. 14-27 Lines 2-3). It should be changed to:

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

The "Objective" section of TS 3.7.2.a (P. 14-27 Lines 13-16) should be changed to:

> 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.

14.110 TS 3.7.2.a.1 limits the concentration of radioactive materials in the effluent released from the facility exhaust stack to 10E5 times the air effluent concentration limits in 10 CFR 20. The bases state that the limit incorporates a dilution factor of 4x10E4. Given that the release concentration limit is greater than the dilution factor, explain how the dilution factor ensures that off-site concentrations of radioactive materials will be below the air effluent concentration limits in 10 CFR 20.

Fifth Response Submitted November 26, 2010

In our view, the pertinent regulation is 10 CFR 20.1101(d) which 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." Thus, we are requesting that the Technical Specification be changed to read:

Limiting Condition for Operation: The annual total effective dose equivalent to the individual member of the public likely to receive the highest dose from air effluents will not exceed 10 mrem as calculated using a generally-accepted computer program.

Surveillance Requirement: Airborne effluents shall be monitored by a continuous 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.

Records: Records of calibration, annual releases and effective dose equivalent calculations shall be maintained for at least three years.

Basis: 10 CFR 20.1101(d) states, in part, "to implement the ALARA requirements of § 20.1101 (b), and notwithstanding the requirements in § 20.1301 of this part, a constraint on air emissions of radioactive material to the environment, excluding Radon-222 and its daughters, shall be established by licensees other than those subject to § 50.34a, such that the individual member of the public likely to receive the highest dose will not be expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year from these emissions."

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

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

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.

14.111 Explain how TS 3.7.2.a.1 ensures that airborne effluents released from the RINSC will satisfy the ALARA dose constraint of 10 CFR 20.1101(d).

Fifth Response Submitted November 26, 2010

Please see our response to RAI 14.110 above. Compliance with 10CFR20.1101 dose limits for individual members of the public from gaseous effluents (10mrem/y) is currently demonstrated by calculation through the use of the COMPLY code as generally described in Section 11.1.7 of the SAR.

14.112 The "Bases" section of TS 3.7.2.a references a letter sent to the NRC in 1963. 10 CFR 50.36 requires that the proposed TS be derived from analysis included in the SAR. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

Ninth Response Submitted February 24, 2011

Response: The calculation of the accident x/Q is provided in Chapter 13, Section 13.2.1.for short-term releases. The dispersion factor given in the technical specification for normal operations was calculated from historic wind rose data provided in the referenced letter. That data has been updated and is summarized below:

Wind From	Frequency
N	0.062
NNE	0.058
NE	0.044
ENE	0.013
E	0.012
ESE	0.013
SE	0.058
SSE	0.049
S	0.058
SSW	0.084
SW	0.105
WSW	0.064
W	0.068
WNW	0.095
NW	0.104
NNW	0.068

It should be noted that the wind pattern is heavily influenced by Narragansett Bay.

In our atmospheric dispersion model, we determined the radionuclide concentrations at ground-level receptors beneath an elevated, buoyant plume of dispersing airborne effluents using two major steps: First, we calculated the height to which the plume rises at a given downwind distance from the plume source. The calculated plume rise was then added to the height of the plume's source point to obtain the so-called "effective stack height", also known as the plume centerline height or simply the emission height. The stack at the Rhode Island Nuclear Science Center is 35 meters high. The effective stack height is determined by the buoyancy of the airborne effluent resulting from the effluent's temperature relative to the temperature of the immediate atmosphere. The ground-level radionuclide concentration beneath the plume at a given downwind distance was then predicted using the Gaussian dispersion equation. It should be noted that our airborne effluents are lighter than the surrounding air because they are generally at a higher temperature than the ambient air into which they are discharged. The dilution factor given in the specification was based on a dispersion factor ($X/Q = 10^{-5}$ sec/m^3). However, please change the specification to read: "The annual total dose equivalent to the maximally exposed individual from radioactive materials discharged to the atmosphere shall not exceed 10 millirems using a generally accepted atmospheric dispersion model."

14.113 The first sentence of TS 3.7.2.b states "The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined before release." This statement appears to be a surveillance requirement and redundant to the requirement specified in TS 4.7.b.2. Explain the reason for including this requirement as an LCO, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

Please replace the technical specification with the following:

> **Limiting Condition for Operation**: Releases from the liquid waste retention tank shall meet requirements in 10 CFR 20.2003.
>
> **Surveillance Requirement**: "The liquid waste retention tank discharge shall be batch sampled and the gross activity per unit volume determined before release."
>
> **Records**: Records of releases shall be maintained for at least three years.
>
> **Basis**: 10 CFR 20.2003 permits discharges to the sanitary sewer provided that conditions in 10 CFR 20.2003 (a) are met.

14.114 TS 3.7.2.b states, "All off-site releases shall be directed into the municipal sewer system." The bases state that liquid wastes can be removed from the site by a commercial licensed organization. Explain this apparent inconsistency between the specification and the bases, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

Please see the response to RAI 14.113.

14.115 TS 3.7.2.b does not contain requirements for the concentration of radioactivity in liquid wastes that can be discharged from the RINSC site. Explain why TS 3.7.2.b does not contain any such requirements, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

Please see the response to RAI 14.113.

14.116 ANSI/ANS-15.1 recommends that the technical specifications specify that experiments will be designed such that they do not contribute to the failure of other experiments or reactor systems and components important to safety. Explain the reason that the proposed TS do not contain any such requirement for experiments, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Technical Specification section 3.8 will be re-written so that it more closely conforms to ANSI 15.1. The reactivity limits on experiments are covered in the re-written section of Technical Specifications 3.1.3 and 3.1.4. See the answer to RAI question 14.137. See the answer to RAI 13.7 for the transient analysis associated with a step reactivity insertion of the maximum worth of an experiment.

Technical Specification 3.8 should be revised to say:

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.

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.

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.

14.117 TS 3.8.3 states, "Fissionable materials shall have total iodine and strontium inventory less than that allowed by the facility by-product license." What facility by-product license does this specification reference? What inventory limits does that by-product license specify? Why are iodine and strontium the only elements of concern for experiments involving fissionable materials? Provide an analysis of the consequences of the failure of an experiment involving fissionable materials that shows the consequences are bounded by the analysis of the MHA presented in Chapter 13 of the SAR. Discuss all assumptions used in the analysis, including justification for the use of the assumptions.

Eighth Response Submitted January 24, 2011

Please change proposed technical specification 3.8.3 to read: "Each experiment containing fissionable materials shall be limited to a maximum reactivity worth of 0.60 % ΔK/K if secured or 0.08 % ΔK/K if moveable. The total reactivity of all experiments shall not exceed 0.60 % ΔK/K." The basis for the proposed technical specification can be found in Chapter 10 of the SAR. SAR Section 10.3, "Experiment Review," states that the reactivity worth of any single

independent experiment or combination of connected experiments that can be added to the core simultaneously cannot exceed 0.60% ΔK/K and the calculated reactivity worth of any single independent experiment not rigidly fixed in place or the combination of connected or related experiments added to the core simultaneously cannot exceed 0.08% ΔK/K. Positive reactivity is the result of the insertion of either fissile materials or reflector materials into the core.

To address the questions posed in the RAI, when the SAR was written, the facility had an Agreement State broad-scope radioactive materials license and that is the license to which the quoted statement refers. When the facility was governed by both licenses, the broad-scope license allowed inventory was limiting. Among the numerous radionuclides formed when fissile material is fissioned in an experiment, the limitations on strontium and iodine were the most restrictive of the inventory limits in the broad-scope license. During the long delay between when the SAR was submitted and the NRC completed its review and issued this RAI, the broad-scope license was dropped. It should be noted that our current technical specifications contain the same restriction and we were asked by the NRC not to submit any requests for amendments to our license while we were awaiting review of our SAR. Thus, since the limiting broad-scope license inventory items no longer exist, the questions posed in the first portion of this RAI are essentially moot at this point.

It is our contention that the limitations on the reactivity worth of an experiment essentially assures that the consequences of failure of that experiment will remain within the dose equivalent consequences of the fuel element failure. It should be noted that the RINSC is currently licensed to increase the core fuel elements from fourteen to seventeen. Each additional fuel element provides approximately 275 grams of uranium-235 far exceeding the reactivity of any single experiment or combination of experiments containing fissionable material. Additionally, there are technical specification limits on core excess reactivity and core shutdown margin that must be met taking the experiment into account. The inventory of radioactivity in the core is dependent on core power level and the RINSC is limited to 2 MW. The MHA assumes the failure of a fuel element containing the fission products from far more fissionable material than any single experiment or combination of experiments. Thus, the MHA bounds the radiological consequences of the failure of an experiment containing fissionable materials.

14.118 TS 3.8.5 states, "experiments shall be designed against failure from internal and external heating at the true values associated with the LSSS for reactor power level and other process variables." ANSI/ANS-15.1 recommends that experiments also be able to withstand reactor transients. Section 4.6.4 of the SAR states that a rising power transient could result in a maximum reactor power of 2.78 MW, which is greater than the LSSS value of 2.30 MW. Explain how TS 3.8.5 ensures that experiments will be designed to withstand reactor transients, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Technical Specification 3.8 has been re-written in order to make it conform more closely to ANSI 15.1. There is no specific discussion about internal or external heating. Refer to the answer provided for RAI question 14.116.

14.119 The requirements of TS 3.8.10 imply that accidents involving experiments could result in occupantional and public radiation doses up to the regulatory limits. These doses would be greater than the consequences of a fuel failure accident analyzed in Section 13.2.1 of the SAR. Explain why the SAR considers the fuel failure accident to be the MHA if the failure of an experiment could have greater consequences. Provide an analysis of the occupational and public dose consequences of the worst-case failure of an experiment that is consistent with the requirements of the proposed TS. Discuss all assumptions used in the analysis, including justification for the use of the assumptions.

Eighth Response Submitted January 24, 2011

Once more the RAI shows the inherent confusion in the guidance provided by the NRC in NUREG-1537, Part 1. The NUREG uses ANSI/ANS-15.1-1990 as its recommended (required) guide. In keeping with ANSI/ANS-15.1-1990, Section 3.8.3 (1), credible failure of any experiment shall not result in releases or exposures in excess of established limits nor in excess of annual limits established in 10 CFR 20. Proposed technical specification 6.5.9, "Operating Procedures," states, in part: "Experiment review on a case-by-case basis assuring that section 3.8.3(2) of ANSI/ANS 15.1 is satisfied." Experiments are reviewed by the Nuclear and Radiation Safety Committee prior to initiation (see proposed technical specification 6.4.2.b).

14.120 TS 3.8.10 contains requirements related to occupational and public radiation doses resulting from experiments. The specification states, "Experimental materials... which could off-gas... under: (1) normal operating conditions of the experiment... shall be limited in activity such that: if 100% of the gaseous activity or radioactive aerosols produced escaped to... the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the occupational limits for maximum permissible concentration." Explain the reason for allowing normal operation of experiments to result in off-site concentrations of radioactive materials up to the regulatory limits. Does this requirement pertain to the sum of all experiments or individual experiments? Explain how this requirement ensures compliance with the ALARA dose constraint of 10 CFR 20.1101(d). (See RAI 14.123)

Fifth Response Submitted November 26, 2010

Please change TS 3.8.10 to read:

"The radiation dose received by any individual member of the public shall not exceed 100mrem (1mSv) in any calendar year as a result of all experiments conducted at the facility.

The radiation dose in unrestricted areas shall not exceed 2mrem (0.02mSv) in any one hour from any single experiment or set of experiments.

Annual air emissions of radioactive materials from routine operations and all experiments conducted shall not result in doses greater than 10mrem (0.1mSv) total effective dose equivalent (TEDE)"

14.121 TS 3.8.10 specifies requirements related to failure of an experiment encapsulation. Explain what specific types of encapsulation are covered by TS 3.8.10, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

As originally submitted, specification 3.8.1 indicates that sample materials that are going to be irradiated must be contained in containers or encapsulation materials that do not react with water, and do not induce corrosion of core and core structural materials. The type of encapsulation is not relevant, as long as it is able to contain the sample material, and as long as the material used for the containers will not react with any of the core, core structure, or coolant materials.

As originally submitted, specification 3.8.10 describes actions to be taken in the event of the failure of a sample container.

Technical Specification 3.8 has been re-written in order to make it conform more closely to ANSI 15.1. There is no discussion about types of encapsulation. Refer to the answer provided for RAI question 14.116.

14.122 It appears that the first sentence of the second paragraph of TS 3.8.10 explicitly excludes "fuel materials" from the requirements of TS 3.8.10. Clarify whether "fuel materials" is synonymous with "fissionable materials" as used in TS 3.8.3. If the requirements of TS 3.8.10 exclude fissionable materials, explain the reason for not including similar requirements for experiments that contain "fuel materials," and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Technical Specification 3.8 has been re-written in order to make it conform more closely to ANSI 15.1. The radiological release of all experiments is now covered in Specification 3.8.2.1. Refer to the answer provided for RAI question 14.116.

14.123 The second paragraph of TS 3.8.10 states, "if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the occupational limits for maximum permissible concentration." Revise the proposed TS to use current 10 CFR Part 20 terminology (e.g., Annual Limit on Intake or Derived Air Concentration). Explain why occupational concentration limits are used as limits for the release of radioactive material to the atmosphere, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

Please see the response to RAI 14.120.

14.124 The third paragraph of TS 3.8.10 contains assumptions used to calculate releases of radioactive material from experiment malfunctions. These assumptions do not appear to be derived from analyses in the SAR and the bases for TS 3.8 state that the specification is "self explanatory." Provide discussions and/or analyses that explain the assumptions required by TS 3.8.10. Revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

Please see the response to RAI 14.120.

14.125 The third paragraph of TS 3.8.10 states, "Limits for maximum permissible concentrations are specified in the appropriate section of 10 CFR 20." Revise the proposed TS to use current 10 CFR Part 20 terminology and to be more specific about the section of 10 CFR Part 20 that applies to TS 3.8.10.

Seventh Response Submitted December 14, 2010

Please change the wording in TS 3.8.10 to read: "Limits for derived air concentrations for occupational exposure may be found in 10 CFR Part 20, Appendix B, Table 1, Column 3 and limits for derived air concentrations for airborne effluent releases may be found in 10 CFR Part 20, Appendix B, Table 2, Column 1."

14.126 The bases for TS 3.8 state that several of the specifications are "self explanatory." In accordance with 10 CFR 50.36, provide bases for all of the specifications in TS 3.8.

Eighth Response Submitted January 24, 2011

Technical Specification 3.8 has been re-written in order to make it conform more closely to ANSI 15.1. Refer to the answer provided for RAI question 14.116.

14.127 TS 3.9.a.1 sets a limit of 1x10E22 neutrons per square centimeter on the accumulated flux for the beryllium reflectors. The SAR does not appear to contain an analysis that supports the flux limit. Provide an analysis of the flux limit for the beryllium reflectors.

Second Response Submitted August 6, 2010

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 X 10^{22} nvt, and suggest that "a value of 1 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 X 10^{22} nvt is even more conservative than what this analysis considers because it is not limited to fast neutron flux. See the reference entitled "Be N Fluence".

14.128 The bases for TS 3.9.a reference a version of the SAR that is different than the version of the SAR submitted with the license renewal application. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

Eighth Response Submitted January 24, 2011

TS 3.9a references "Part A Section VIII" of the SAR. This section is from a previous SAR. TS 3.9a should reference Section 4.2.3, 'Neutron Moderators and Reflectors'. The TS shall be changed to reference the current SAR. (See response for RAI 14.1)

14.129 TS 3.9.b appears to be a surveillance requirement and not a LCO on the physical condition of the fuel. Explain why the TS do not specify an LCO on the physical condition of the fuel, and revise the proposed TS as appropriate.

Ninth Response Submitted February 24, 2011

As submitted Technical Specification 3.9.b requires that the fuel elements be inspected for physical defects and reactor core box fit. This is a surveillance requirement rather than an LCO. Consequently, this specification has been moved to become Technical Specification 4.9.b. See the answer to RAI question 14.167.

14.130 TS 4.0 specifies that some surveillance requirements may be deferred during periods of reactor shutdown. As recommended in ANSI/ANS-15.1, allowed deferral of a surveillance requirement should be specified as part of the surveillance requirement. Each surveillance requirement that may be deferred during reactor shutdown must specify whether the surveillance must be completed prior to reactor operation. Each allowed deferral must be supported by a basis

statement that explains the reason deferral is warranted during reactor shutdown. Revise the proposed TS as appropriate.

Ninth Response Submitted February 24, 2011

This question is best addressed after the set of RINSC surveillance items have been determined. The rationale and basis for deferring any given surveillance item, and the determination of how long the item may be deferred depends on what the complete set of surveillance items include, and therefore what other related operability checks, calibrations, and inspections are being performed during an outage.

14.131 TS 4.1.1 requires measurement of shim blade insertion rates. Explain the reason for not requiring measurement of shim blade withdrawal rates, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

TS 4.2.8 requires that reactivity insertion rates be determined annually and whenever a new core is configured. In order to make this determination, shim safety blade withdrawal rates must be determined. No requirement is specified for measuring the withdrawal rates, because it would be redundant. They are determined as part of the reactivity insertion rate measurement.

14.132 TS 4.1.1 does not require surveillance of the shim safety blades following maintenance or replacement. Explain the reason for not requiring surveillance of the shim safety blades following maintenance or replacement, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Technical Specifications 4.1 and 4.2 have been re-written in order to make them conform more closely to ANSI 15.1. Refer to the answer provided for RAI question 14.138 for the revised version of Technical Specification 4.1, and refer to the answer provided for 14.141 for the revised version of Technical Specification 4.2. Specification 4.1.1.2 provides the surveillance requirements for shim safety blade reactivity worths. Specification 4.2.2 provides the surveillance requirements for shim safety blade reactivity insertion rates. In both cases, a surveillance requirement has been added to cover activities that could have an effect on these parameters.

14.133 TS 4.1.1.b references the startup core and three other analyzed cores. Explain the reason for referencing the startup core, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

There is no reason for specific core configurations to be referenced in this section of the specification. Technical Specification 4.1 has been re-written in order to make it conform more closely to ANSI 15.1. Refer to the answer provided for RAI question 14.138. The surveillance requirements have been written so that they do not refer to specific core configurations.

14.134 TS 4.1.1.b implies that there are only three allowed core configurations for the RINSC reactor. The proposed TS do not contain an LCO restricting the configuration of the RINSC core to three configurations. Explain the reason for only requiring surveillance of the shim safety blades when switching to one of the three referenced core configurations, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Technical Specification 4.1 has been re-written in order to make it conform more closely to ANSI 15.1. Refer to the answer provided for RAI question 14.138. The surveillance requirements have been written so that they do not refer to specific core configurations.

14.135 TS 4.1.1.b references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed TS will become part of the TS and license. Explain why it is necessary to make the referenced portion of the SAR a requirement in the proposed TS.

Eighth Response Submitted January 24, 2011

Technical Specifications 4.1 and 4.2 have been re-written in order to make them conform more closely to ANSI 15.1. Refer to the answer provided for RAI question 14.138 for the revised version of Technical Specification 4.1, and refer to the answer provided for 14.141 for the revised version of Technical Specification 4.2. Specification 4.1.1.2 provides the surveillance requirements for shim safety blade reactivity worths. Specification 4.2.2 provides the surveillance requirements for shim safety blade reactivity insertion rates. References to the SAR have been removed.

14.136 TS 4.1.2 requires inspection of the shim safety blades to detect swelling. The bases for TS 4.1.2 state that inspection will detect swelling and cracking. Explain this apparent inconsistency between the specification and the bases, and revise the proposed TS as appropriate.

Ninth Response Submitted February 24, 2011

The purpose of inspecting the control blades is to ensure that they do not swell. Swelling could cause the blade insertion rate to increase to an extent that scram drop times could be impacted. In an effort to make the basis consistent with the

specification, the paragraph that describes the basis for this (P. 14-32 Lines 43-46) will be modified to say:

> Shim safety blade inspections have the potential to be the single largest source of radiation exposure to the facility personnel. In order to minimize personnel radiation exposure and provide an inspection frequency that will detect early evidence of swelling, an annual inspection interval was selected for Specification 4.1.2.

14.137 TS 4.1.3 requires measurement of an experiment's reactivity worth prior to the "initial use" of the experiment. The bases for TS 4.1.3 state, "The specified surveillance relating to the reactivity worth of experiments will assure that the reactor is not operated for extended periods before determining the reactivity worth of experiments." The specification and bases imply that the reactor can be operated without determining the reactivity worth of experiments. Explain how TS 4.1.3 ensures that the experiment reactivity requirements of TS 3.1 are met, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

In order to determine the reactivity worth of an experiment, one may either:

A. Estimated it based on previous experience due to the similarity of material, quantity of material, position in the core, etc. with other experiments for which reactivity worth has been determined.
B. Measure it by determining the critical control rod heights with, and without the experiment loaded, and calculate the reactivity difference to determine the reactivity effect of the experiment.

For experiments that are not similar to any previously performed experiments, option B is the only way to determine the reactivity effect.

The reactor is defined in TS 1.17 to be operating "whenever it is not secured or shutdown". Consequently, it is not possible to measure the reactivity effect without the reactor being in operation. The basis for this specification acknowledges this fact.

The answer to RAI question 14.60 that was sent with RINSC's second submission of answers should be updated to say:

3.1.3 The total 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:

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

3.1.4 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:

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

14.138 The bases for TS 4.1.3 state that the specification "provides assurance that experiment reactivity worths do not increase beyond the established limits due to core configuration changes." The specification does not appear to require any surveillance of experiment reactivity worths following core configuration changes. Explain the apparent inconsistency between the specification and the bases, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Technical Specifications 3.1 and 4.1 have been re-written in order to make them conform more closely to ANSI 15.1. Some of the specifications that had been in section 3.1 have been moved. The table that follows provides a summary of how things have been changed. The new Technical Specification 4.1.1.3.2 addresses the issue regarding experiment worth changes due to core configuration changes.

Original Location	**Specification**	**New Location**
3.1.1	Shutdown Margin Reactivity	3.1.1.1.1
3.1.2	Core Excess Reactivity	3.1.1.1.2
3.1.3	Total Experiment Reactivity Worth	3.1.1.3.1
3.1.4	Individual Experiment Reactivity Worth	3.1.1.3.2
3.1.5	Criticality During Fuel Loading	3.1.1.1.4
3.1.6	Regulating Rod Reactivity Worth	3.1.1.2.1
3.1.7	Flooded Experiment	3.8.2.4
3.1.8	Negative Temperature Coefficient	3.1.1.1.3
	Temperature Coefficient Surveillance	4.1.1.1.3
3.1.9	FC Mode Operation Core Grid Filled	3.1.2.1
3.1.10	FC Mode Operation Coolant Gate Stored	3.1.2.2

The basis section of Specifications 3.1.1.3.1 and 3.1.1.3.2 refer to analyses performed for reactivity insertions. The determination of a period due to a 0.08 % dK/K reactivity insertion is part of the answer to RAI question 14.61.

The basis section of Specification 4.1.1.1.3 refers to an analysis that estimates the temperature coefficient. This analysis is part of the answer to RAI questions 4.12 and 4.13.

The new versions of Technical Specification 3.1 and 4.1 are:

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

3.1.1.3.1 The total 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:

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:

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.

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

Specification 3.1.1.1.3 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.

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.

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.

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.

Specification:

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.

Basis:

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.

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

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

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.

14.139 ANSI/ANS-15.1 recommends annual thermal power verification. Explain the reason that the proposed TS do not contain any such requirement, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

RINSC Technical Specification 4.2 has been revised. See the answer to RAI question 14.141. Specification 4.2.7.5 requires that the power level channels be calibrated annually. This calibration is done by thermal power verification.

14.140 ANSI/ANS-15.1 recommends annual surveillance of required interlocks. Explain the reason that the proposed TS do not contain any such requirements, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

RINSC Technical Specification 4.2 has been revised. See the answer to RAI question 14.141. Interlock surveillance is covered in Specification 4.2.6.

14.141 TS 4.2 specifies surveillance requirements for the safety system and safety-related instrumentation required by TS 3.2.1. However, the proposed TS do not specify surveillance requirements for many of the items required by TS 3.2.1, Table 3.1 and Table 3.2. In accordance with 10 CFR 50.36(c)(3), propose surveillance requirements for the safety system and safety related instrumentation required by TS 3.2.1.

Eighth Response Submitted January 24, 2011

Specification 4.2 has been revised to more closely reflect what ANSI 15.1 suggests should be covered by this specification. The following table shows how the locations of the original specifications have changed.

Specification	**Original Location**	**ANSI Standard**	**New Location**
Channel Test of Neutron Flux Level Safeties	4.2.1	4.2.5	4.2.4.2
Channel Test of Period Safety	4.2.1	4.2.5	4.2.4.3
Channel Calibration of Channels in Table 3.1	4.2.2	4.2.5	4.2.7
Radiation Monitors in Table 3.2 Operable	4.2.3		4.2.3
Rod Drop Time	4.2.4	4.2.4	4.2.1.1
Rod Drop Time	4.2.5	4.2.4	4.2.1.2
Shutdown Margin	4.2.6	4.1.2	4.1.1.1.1
Excess Reactivity	4.2.7	4.1.1	4.1.1.1.2

Reactivity Insertion Rate	4.2.8	4.2.2	4.2.2
Control Rod Reactivity Worth	4.1.1	4.2.1	4.1.1.2
Power Calibration		4.2.8	4.2.7.5

The following is the new proposed Specification:

4.2 Reactor Control and Safety System

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

4.2.2 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

4.2.3 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 or de-energized:

4.2.3.1 The experimental level area radiation monitor
4.2.3.2 The pool top area radiation monitor
4.2.3.3 The gaseous effluent air monitor
4.2.3.4 The particulate air monitor

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 after the channel has been repaired or de-energized:

4.2.4.1 Control room manual scram button
4.2.4.2 Power level channels
4.2.4.3 Period channel
4.2.4.4 Rod control communication watchdog scram

4.2.5 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 or de-energized for which reactor power level will be greater than 100 kW:

4.2.5.1 All of the reactor safety and safety related instrumentation listed in 4.2.4
4.2.5.2 Primary coolant flow rate scram

4.2.6 The following reactor safety and safety related instrumentation alarms, scrams, and interlocks shall be tested annually:

4.2.6.1 The following detector HV failure scrams:

4.6.2.1.1 Power level channels
4.6.2.1.2 Period channel

4.2.6.2 The following shim safety withdrawal interlocks:

4.2.6.2.1 Start-up count rate
4.2.6.2.2 Test / Select switch position

4.2.6.3 The following servo control interlocks:

4.2.6.3.1 Regulating blade not full out
4.2.6.3.2 Period less than 30 seconds

4.2.6.4 The following coolant system channel temperature alarms and scrams:

4.2.6.4.1 Primary inlet temperature alarm
4.2.6.4.2 Primary outlet temperature alarm
4.2.6.4.3 Primary outlet temperature scram
4.2.6.4.4 Pool temperature alarm
4.2.6.4.5 Pool temperature scram

4.2.6.5 The following coolant system channel flow scrams:

4.2.6.5.1 Primary flow scram
4.2.6.5.2 Inlet and outlet coolant gates open scrams
4.2.6.5.3 No flow thermal column scram

4.2.6.6 Low pool level scram

4.2.6.7 The following bridge scrams:

4.2.6.7.1 Bridge manual scram
4.2.6.7.2 Bridge movement scram
4.2.6.7.3 Bridge low power position scram

4.2.6.8 Seismic scram

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

4.2.7.1 The experimental level area radiation monitor
4.2.7.2 The pool top area radiation monitor
4.2.7.3 The gaseous effluent air monitor
4.2.7.4 The particulate air monitor
4.2.7.5 Power level channels
4.2.7.6 Primary flow channel
4.2.7.7 Primary inlet temperature channel
4.2.7.8 Primary outlet temperature channel
4.2.7.9 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 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 4.2.4 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 4.2.5 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 4.2.6 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 4.2.4 and 4.2.5. The annual requirement is consistent with the historical facility frequency.

Specification 4.2.7 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.

14.142 TS 4.2.1.a requires channel tests of nuclear instrumentation "prior to each reactor startup following a period when the reactor was secured." Given that the TS do not require the reactor to be secured on a periodic basis, explain the reason for not requiring periodic (e.g., quarterly) surveillance of the nuclear instrumentation, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

While we do not want to limit our ability to operate the reactor for extended runs over multiple days, the current typical operating schedule at RINSC is one shift per day. Our desire is to set this surveillance such that these channel checks are performed once prior to the initial start-up of the day, so that if there are multiple start-ups for the day, additional channel checks are not required.

In the event that there were a multi-day operation, it is not considered likely that RINSC could operate for a quarter of a year without re-fuelling. RINSC reached its equilibrium core in October of 2008. Based on operating data, we expect to have to refuel after 1550 MWH of operation. Therefore, if we were start with a fresh core, and operate 24 hours per day, 7 days per week, we would reach our

limit of 1550 MWH of operation in 2.1 months. Consequently, quarterly surveillance in lieu of prior to initial start-up of the day is redundant.

If the wording is changed to make sure that pre-start checkouts are performed prior to the initial start-up each day that the reactor is started up from the shutdown condition, rather than after it has been secured, these conditions can be met.

RINSC Technical Specification 4.2 has been revised. See the answer to RAI question 14.141. The new proposed specification regarding the operability of the Neutron Flux Level Safety and Period Safety Channels is:

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 after the channel has been repaired or de-energized:

4.2.4.2 Power level channels
4.2.4.3 Period channel

14.143 TS 4.2.2 states, "A channel calibration of the safety channels listed in Table 3.1, which can be calibrated, shall be performed annually." Revise the proposed TS to explicitly state which channels listed in Table 3.1 will be calibrated annually.

Eighth Response Submitted January 24, 2011

Table 3.1 was updated as part of the answer to RAI question 7.1. RINSC Technical Specification 4.2 has been revised. See the answer to RAI question 14.141. The new proposed specification regarding channel calibrations is:

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

4.2.7.1 The experimental level area radiation monitor
4.2.7.2 The pool top area radiation monitor
4.2.7.3 The gaseous effluent air monitor
4.2.7.4 The particulate air monitor
4.2.7.5 Power level channels
4.2.7.6 Primary flow channel
4.2.7.7 Primary inlet temperature channel
4.2.7.8 Primary outlet temperature channel
4.2.7.9 Pool temperature channel

14.144 TS 4.2.3 appears to be an LCO and not a surveillance requirement. Explain the reason for including TS 4.2.3 in the surveillance requirements, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

The LCO regarding the required radiation monitoring instrumentation is covered in the new proposed Specifications 3.2.1.3 and 3.2.1.4 which were submitted as part of the answer to RAI question 14.87. The corresponding surveillance requirements are part of the revised Specification 4.2 which is part of the answer to RAI question 14.141.

14.145 TS 4.2.6 does not require surveillance of the shutdown margin following changes in control blades. Explain the reason for not requiring surveillance of the shutdown margin following control blade changes, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

Technical Specification 4.2.6 will be changed to say:

> The shutdown margin shall be determined in accordance with operating procedures:
>
> > Annually,
> >
> > When a new core is configured,
> >
> > Following control blade changes.

14.146 TS 4.2.6 references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed TS will become part of the TS and license. Explain why it is necessary to make the referenced portion of the SAR a requirement in the proposed TS.

Eighth Response Submitted January 24, 2011

Technical Specification 4.2 has been re-written as part of the answer to RAI 14.141. The reference to the SAR has been removed.

14.147 TS 4.2.7 does not require surveillance of the excess reactivity following changes in control blades. Explain the reason for not requiring surveillance of the excess reactivity following control blade changes, and revise the proposed IS as appropriate.

Second Response Submitted August 6, 2010

Technical Specification 4.2.7 will be changed to say:

> The excess reactivity shall be determined in accordance with operating procedures:
>
> > Annually,
> >
> > When a new core is configured,
> >
> > Following control blade changes.

14.148 TS 4.2.7 references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed IS will become part of the IS and license. Explain why it is necessary to make the referenced portion of the SAR a requirement in the proposed TS.

Eighth Response Submitted January 24, 2011

Technical Specification 4.2 has been re-written as part of the answer to RAI 14.141. The reference to the SAR has been removed.

14.149 TS 4.2.8 does not require surveillance of the reactivity insertion rate following changes in control blades. Explain the reason for not requiring surveillance of the reactivity insertion rate following control blade changes, and revise the proposed IS as appropriate.

Second Response Submitted August 6, 2010

Technical Specification 4.2.8 will be changed to say:

> The excess reactivity shall be determined in accordance with operating procedures:
>
> > Annually,
> >
> > When a new core is configured,
> >
> > Following control blade changes.

14.150 TS 4.2.8 references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed IS will become part of the IS and license. Explain why it is necessary to make the referenced portion of the SAR a requirement in the proposed TS.

Eighth Response Submitted January 24, 2011

Technical Specification 4.2 has been re-written as part of the answer to RAI 14.141. The reference to the SAR has been removed.

14.151 The "Bases" section of TS 4.2 does not contain bases for TS 4.2.6, 4.2.7, or 4.2.8. Provide bases for these specifications.

Eighth Response Submitted January 24, 2011

RINSC Technical Specification 4.2 has been revised. See the answer to RAI question 14.141.

14.152 The bases for TS 4.2.3 states, "Radiation monitors are checked for proper operation in Specification 4.2.3. Calibration and setpoint verification involve..." However, TS 4.2.3 appears to be an LCO and does not specify surveillance requirements (e.g., channel tests, channel checks, or channel calibrations). Explain this apparent inconsistency between the specification and the bases for IS 4.2.3, and revise the proposed IS as appropriate.

Eighth Response Submitted January 24, 2011

RINSC Technical Specification 4.2 has been revised. See the answer to RAI question 14.141.

14.153 The second paragraph of the bases for TS 4.3.a appears to be a description of the pool level detection system, not the bases for the proposed surveillance requirements. In accordance with 10 CFR 50.36, provide bases that explain the reasons for the requirements of TS 4.3.a.4, 4.3.a.5, and 4.3.a.6.

Tenth Response Submitted July 15, 2011

Technical Specifications 3.3 and 4.3 have been re-written in order to make them more consistent with ANSI 15.1. The following table provides a summary of how the specifications have changed:

Original Location	Specification	New Location
3.3.a.1	Primary pH	Note 1
3.3.a.2	Primary Conductivity	3.3.1.1
3.3.a.3	Primary Radiological Analysis	3.3.1.2
3.3.b.1	Secondary pH	Note 2
3.3.b.2	Secondary Radiological Analysis	3.3.2
4.3.a.1	Primary pH Surveillance	Note 1
4.3.a.2	Primary Conductivity Surveillance	4.3.1.1
4.3.a.3	Primary Radiological Analysis Surveillance	4.3.1.2
4.3.a.4	Pool Level Scram Test	4.2.6.6

4.3.a.5	Pool Inspection – Primary System Inspection	4.3.1.4
4.3.a.6	Pool Level Verification	4.3.1.3
4.3.b.1	Secondary pH Surveillance	Note 2
4.3.b.2	Secondary Radiological Analysis Surveillance	4.3.2.1
N/A	Secondary System Inspection	4.3.2.2

Note 1 ANSI 15.1 recommends that either pH or Conductivity be monitored. pH and conductivity are related, so monitoring them both is redundant. CO_2 dissolves into the pure pool water, which pushes the equilibrium pH down to an average of approximately 5.6. Consequently, conductivity is a better measure of water quality. RINSC proposes to use a conductivity measurement rather than pH to monitor water quality.

Note 2 The purpose of measuring pH and conductivity is to reduce activation products in the coolant, and to minimize corrosion. Activation products are not an issue on the secondary side of the cooling system because the coolant is not exposed to a neutron flux. Corrosion on the secondary side of the cooling system is no longer an issue because the aluminum piping has been replaced with PVC piping, and the cooling towers are made of non-corrosive materials as well. RINSC proposes to remove this surveillance.

3.3 Coolant Systems

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.

4.3 Coolant Systems

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.

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

4.3.2 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 primary 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 primary 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.

14.154 The "Bases" section of TS 4.3.b appears to be a description of how secondary coolant chemistry is controlled and how secondary coolant radioactivity is monitored, not the bases for the proposed surveillance requirements. In accordance with 10 CFR 50.36, provide bases that explain the reasons for the requirements of TS 4.3.b.1 and 4.3.b.2.

Tenth Response Submitted July 15, 2011

Technical Specifications 3.3 and 4.3 have been re-written in order to make them more consistent with ANSI 15.1. See the answer to RAI question 14.153.

14.155 ANSI/ANS-15.1 recommends surveillance of required ventilation filters. Explain the reason that the proposed TS do not contain any such requirements, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

ANSI/ANS standards are recommendations, not requirements. ANSI/ANS 15.1 Section 4.5.2 recommends that filter efficiency measurements be made annually to biennially, or following major maintenance. The only required ventilation filter at RINSC is the Emergency Exhaust Filter, which is tested annually as required by RINSC TS 4.4, 4.5, 4.6 Specification 4.

14.156 The first sentence of Specification 1 of TS 4.4, 4.5, 4.6 appears to be a surveillance requirement. The rest of Specification 1 appears to be a combination of a description of system operation and LCOs for the confinement and emergency exhaust systems (e.g., maximum emergency cleanup system flow rate, minimum differential pressure, etc.). Revise Specification 1 to include only surveillance requirements and relocate any LCOs to the appropriate sections of the proposed TS.

Fifth Response Submitted November 26, 2010

Technical Specifications 4.4, 4.5, 4.6 have been broken apart in an attempt to make this section of the Technical Specifications follow the format outlined in ANSI 15.1. These specifications will be written as follows:

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:

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:

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

    1. Quarterly
    2. After any maintenance that could affect the operability of the system

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:

    1. The evacuation horn sounds
    2. The following dampers close:

        1. Confinement Air Intake Damper
        2. Confinement Air Exhaust Damper

    3. The negative differential pressure inside confinement with respect to the outside is at least 0.5 inches of water.
    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:

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.

Specification:

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

    1. Quarterly
    2. After any maintenance that could affect the operability of the system

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:

    1. The following blowers are de-energized:

        1. Confinement Exhaust Blower
        2. Rabbit System Blower
        3. Off Gas System Blower

    2. The following blowers are energized:

        1. Emergency Exhaust Blower
        2. Dilution Blower

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

    1. Annually
    2. After any maintenance that could affect the operability of the system

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

    1. Biennially
    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:

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

2. A functional test of the emergency power system under load shall be performed:

    1. Biennially
    2. Following emergency system load changes

Bases:

Periodic tests of the emergency power system 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.

14.157 Specification 2.a of TS 4.4, 4.5, 4.6 requires inspection of "building ventilation blowers and dampers (including solenoid valves, pressure switches, piping, etc.)" Revise the proposed TS to explicitly state each piece of equipment that must be inspected.

Fifth Response Submitted November 26, 2010

See the response to RAI question 14.156. These technical specifications have been completely re-written to conform to ANSI 15.1.

Specification 4.5.1.1 requires that the ventilation components for normal operation be verified to be operation each day of reactor operation.

Specification 4.5.2 indicates the surveillance requirements for verifying the operability and functionality of the emergency operation mode ventilation system components.

14.158 Specification 2.b of TS 4.4, 4.5, 4.6 requires inspection of personnel access and reactor room overhead doors. Explain why the specification does not require inspection of the truck door, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

The truck door IS the overhead door.

14.159 Specification 3 of TS 4.4, 4.5, 4.6 does not contain enough detail regarding the testing frequency of the emergency generator. Revise the proposed TS to include the maximum surveillance interval for testing the emergency generator. Describe the tests that comprise the emergency generator testing, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

See the response to RAI question 14.156. These technical specifications have been completely re-written to conform to ANSI 15.1.

Specification 4.6.1 requires that the starting capability of the emergency power system be verified quarterly.

Specification 4.6.2 requires that the capability of the emergency power system to operate under load be tested biennially, and whenever the emergency system load changes.

14.160 ANSI/ANS-15.1 recommends technical specifications include surveillance requirements for radiation monitoring at site boundary and environmental monitoring. Section 11.1.7 of the SAR discusses environmental monitoring at the RINSC. Explain the reason for not including such surveillance requirements, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

ANSI/ANS-15.1 recommends but does not require a technical specification including surveillance requirements for radiation monitoring at the site boundary and environmental monitoring. The section entitled "Environmental Effects of Facility Operation" of Appendix 12.1 to NUREG 1537 Part 2, indicates that "Yearly doses to unrestricted areas will be at or below established guidelines in 10CFR Part 20." It is our contention that as long as we can demonstrate that our annual doses to individuals in unrestricted areas meet those criteria, we do not need a technical specification governing radiation monitoring at our site boundary or additional environmental monitoring. Over forty years of operating experience support that claim.

14.161 TS 4.7.a.1 requires annual calibration of the particulate air monitors. The LCOs specified in TS 3.2.1, Table 3.2 do not appear to contain a requirement for particulate air monitors. Explain this apparent inconsistency, and revise the proposed TS as appropriate. (See RAI 14.103)

Fourth Response Submitted September 8, 2010

See the answer to RAI question 7.4, which has been revised to require "a minimum of one radiation monitor that is capable of warning personnel of high radiation levels in the confinement particulate effluent".

14.162 TS 4.7.a.3 requires a daily channel check of the "main floor monitor." The TS do not appear to contain an LCO for a "main floor monitor." Revise TS 4.7.a.3 to use terminology for radiation monitors consistent with the terminology for radiation monitors required by TS 3.2.1, Table 3.2 or TS 3.7.1, or propose an LCO for a "main floor monitor."

Fourth Response Submitted September 8, 2010

See the answer to RAI question 7.4, which shows the revision to RINSC Technical Specification Table 3.2. As this revision is written, the required air monitoring instrumentation includes "a minimum of one radiation monitor that is capable of warning personnel of high radiation levels in the confinement particulate effluent". The Stack Particulate Monitor may serve to meet this requirement. Consequently, the Main Floor Air Monitor may not be required to be operational during reactor operation.

Revise TS 4.7.a.3 to say:

1. A channel check of the particulate air monitor shall be performed for each day of operation, or once for each operation that lasts for multiple days.
2. The particulate air monitor shall be calibrated annually.
3. A channel check of the gaseous air monitor shall be performed for each day of operation, or once for each operation that lasts for multiple days.
4. The gaseous monitor shall be calibrated annually.

14.163 The bases for TS 4.8 state, "Review of the experiments... assures that the insertion of experiments will not negate the consideration implicit in the Safety Limits." Explain what "consideration implicit in the Safety Limits" means in terms of experiments.

Eighth Response Submitted January 24, 2011

This statement was intended to mean that the safety review of experiments will ensure that the installation of experiments will not put the reactor in a condition that makes reaching a safety limit credible.

Technical Specification 4.8 will be re-written to say:

4.8 Experiments

Applicability:

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

Objective:

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

14.164 Since the application for license renewal was submitted, TS 4.9 was amended by Amendment No. 29 to Facility Operating License No. R-95, dated December 28, 2004. Clarify whether the amended TS 4.9 that is currently in the license should replace proposed TS 4.9 contained in the application for license renewal.

Fourth Response Submitted September 8, 2010

The amended version of this specification should be used. See the answer to RAI question 14.167.

14.165 The bases for TS 4.9.a reference a version of the SAR that is different than the version of the SAR submitted with the license renewal application. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

Eighth Response Submitted January 24, 2011

TS 4.9a references "Part A Section VIII" of the SAR. This section is from a previous SAR. TS 4.9a should reference Section 4.2.3, 'Neutron Moderators and Reflectors'. The TS shall be changed to reference the current SAR. (See response for RAI 14.1)

14.166 The bases for TS 4.9.b state, "The fission density limit for this reactor cannot be exceeded." The proposed TS do not appear to contain a fission density limit for the fuel. Explain the reason for not including a fission density limit for the fuel. (See RAI 14.55)

Fifth Response Submitted November 26, 2010

NUREG-1313 (p. 7) states that LEU silicide fuel with up to 4.8 g U/cm^3 was irradiation tested in the 30 MW ORR reactor to burnups up to 98% of the contained ^{235}U. No indication of unusual conditions were observed on the fuel plates that were tested. Consequently, LEU silicide fuel has no burnup limit and no fission density limit. Also see the response to RAI 4.2.

14.167 The bases for TS 4.9.b state, "Burnup calculations are made quarterly (4.9.1)." To what does "(4.9.1)" refer? Explain the reason that the burnup calculations are not a required surveillance, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

Burn-up calculation data is not used to assess the physical condition of the fuel. The fuel is qualified to 98% burn-up (See the answer to RAI 14.55). Consequently, this statement should be removed.

Change this section to say:

4.9.a Beryllium Reflectors

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:

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

   a. Less than or equal to 1 E 22 neutrons / cm^2, and
   b. The fluence shall be determined annually.

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:

   a. The surveillance each year shall include at least one fifth of the beryllium reflectors,
   b. 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
   c. If damage is discovered, then the surveillance shall be expanded to include all of the beryllium reflectors prior to use, and annually thereafter.

Bases:

The neutron fluence 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 X 10^{22} nvt, and they suggest that "a

value of 1 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 X 10^{22} nvt is even more conservative than what this analysis considers because the RINSC limit is not limited to fast neutron flux.

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.b LEU 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:

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:

1. The surveillance each year shall include at least one fifth of the fuel elements,
2. The surveillance each year shall include fuel elements that represent a cross section with respect to burn-up,
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. 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.

14.168 The bases for TS 4.9.b reference a version of the SAR that is different than the version of the SAR submitted with the license renewal application. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

Eighth Response Submitted January 24, 2011

TS 4.9b references "Part A Section VI" of the SAR. This section is from a previous SAR. TS 4.9b should reference Section 4.5, 'Nuclear Design'. The TS shall be changed to reference the current SAR. (See response for RAI 14.1)

14.169 In accordance with 10 CFR 50.36(a)(1), provide bases for proposed technical specifications in Section 5, "Design Features."

Eighth Response Submitted January 24, 2011

Chapter 14, section 5 has been re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.172.

14.170 ANSI/ANS-15.1 recommends that the number and type of control blades be included in the technical specifications. Explain the reason that the regulating blade is not specified in TS 5.3. Explain the reason that the control blade materials are not specified in the proposed TS, and revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Chapter 14, section 5 has been re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.172. The number and type of control blades have been included in section 5.3.

14.171 Proposed TS 5.3 references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed TS will become part of the TS and license. Explain why it is necessary to make the referenced section of the SAR a requirement in the proposed TS.

Eighth Response Submitted January 24, 2011

Chapter 14, section 5 has been re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.172. References to the SAR have been removed.

14.172 TS 5.4 appears to contain LCOs for the emergency cleanup system (e.g., filter requirements). Explain the reason for including these LCOs as part of the design features of the reactor building, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

Chapter 14, Section 5 of the RINSC Safety Analysis Report has been re-written to conform to ANSI 15.1. The items in Section 5.4 that appeared to be LCOs have been moved to Chapter 14, Sections 3 and 4. The re-written version of Section 3 is included as part of the answer to RAI 14.93. The re-written version of Section 4 is included as part of the answer to RAI 14.156. The LCOs that have been moved from Section 5, have been moved to the following sections:

1. Section 3.4.2.2.1.1 requires that the confinement and clean up systems become activated when an evacuation button is depressed.
2. Section 3.4.3.2 covers the requirement that gas leaks between confinement and the outside shall be inward.
3. Section 3.4.3.2.6 requires that a negative differential pressure be maintained between confinement and the outside.
4. Section 3.5.1.1.2 requires that the confinement room exhaust air go through a roughing filter and an absolute filter prior to being released to the environment.

5. Section 3.5.2.1.2 describes the components of the emergency filter system that are required.
6. Sections 3.5.2.1.2 and 3.5.2.1.4 require the confinement exhaust to exit through an emergency filter system and a stack.
7. Section 3.5.2.3 requires that the charcoal filter in the emergency filter system be capable of removing 99% of the radioiodines likely to be present in the event of a fuel failure.
8. Section 3.5.2.4 requires that the exhaust absolute filter for the emergency filter system be certified by the manufacturer to be capable of removing 0.3 micron particulates.
9. Section 3.6 requires that emergency power be available to operate the clean-up system in the event of a facility power failure.
10. Section 4.4.2.2.2 requires that the dampers close when an evacuation button is depressed.
11. Section 4.4.2.2.4 requires that the confinement room vent fans and HVAC system shut off when an evacuation button is depressed.

Chapter 14, Section 5 has been re-written to be the following:

5.0 Design Features

5.1 Site and Facility Description

The Rhode Island Nuclear Science Center (RINSC) 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 facility is one of a number of buildings located on the Bay Campus of the university. The RINSC facility consists of a reactor room and an office wing with one entrance between them. The reactor room acts as the confinement space. The reactor pool is constructed on top of a military gun pad.

A more detailed description of the site and the facility is located in Chapter 1.

5.2 Reactor Coolant System

The RINSC reactor is located in a concrete pool that is lined with aluminum. For operations up to 100 kW, natural convection cooling is possible. For operations above 100 kW, forced convection cooling must be applied. In both cases, the coolant is light water provided by the local town water supply.

The primary section of the forced convection cooling system takes water from the pool outlet line, and directs it to a delay tank where its 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 primary loop consists of a pump and a heat exchanger.

For each loop, the water from the delay tank goes through a primary pump, through a primary heat exchanger, and back to the pool via the pool inlet line, where the two loops recombine. 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

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

A more detailed description of the reactor coolant system is located in Chapter 5.

5.3 Reactor Core and Fuel

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 elements 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.

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.

A more detailed description of the reactor core and fuel is located in Chapter 4, as well as a description of core parameters.

5.4 Fissionable Material Storage

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.

A more detailed description of the fuel storage racks is located in Chapter 9.

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:

1. Fissionable material that is not in use or not in an approved shipping container shall be in storage.
2. Fissionable material storage facilities shall have keff <= 0.9, for all conditions of moderation and reflection using light water.

Bases:

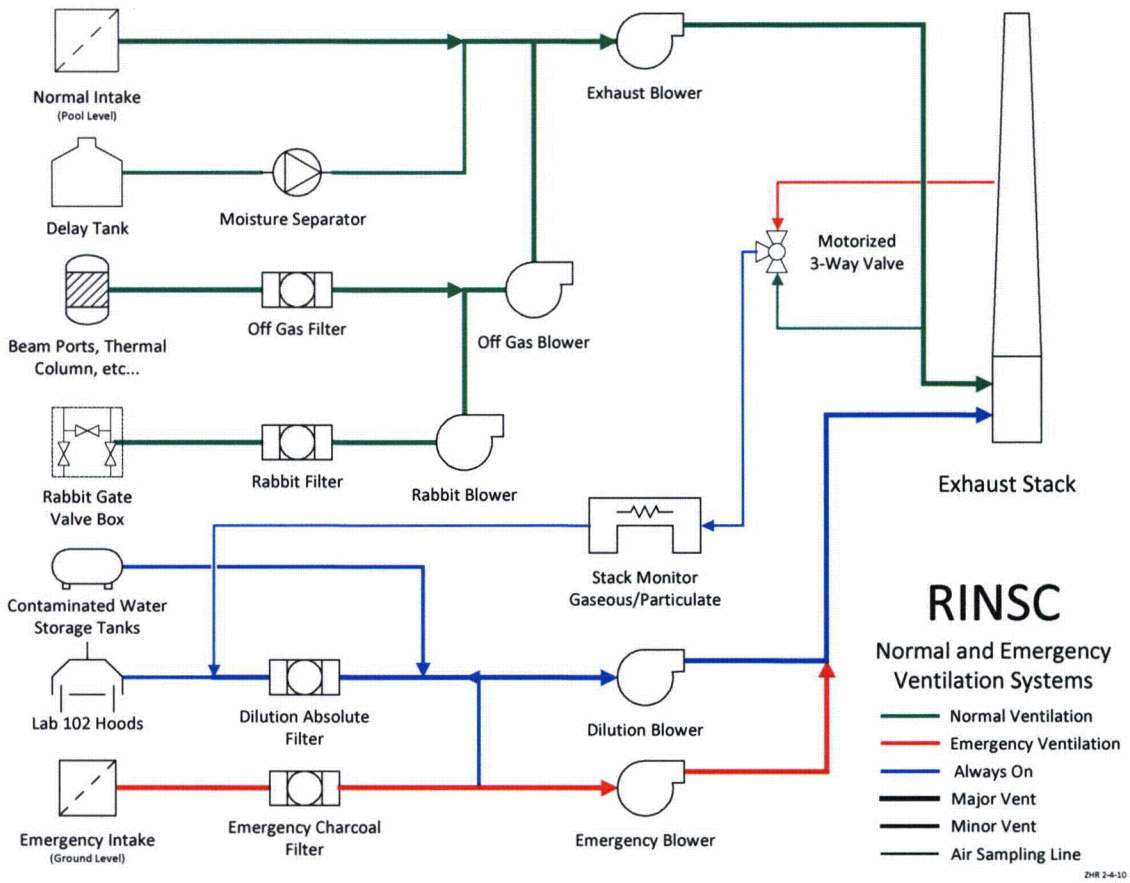
These specifications conform to ANSI 15.1. They ensure that fissionable material that is not in use will remain in a configuration that cannot achieve criticality.

14.173 TS 5.4 states, "The reactor building exhaust blower operates in conjunction with additional exhaust blower(s) which provide dilution air from non-reactor building sources." Clarify whether this statement applies to normal ventilation, emergency cleanup system operation, or both.

Fourth Response Submitted September 8, 2010

This statement applies to both, normal and emergency operation. Under normal operation, air exits confinement through the normal exhaust intake, goes through the exhaust blower, and enters the base of the stack. The experimental gas

systems (off gas and rabbit) tie into the normal exhaust system so that air from these systems go through the exhaust blower and enter the base of the stack. This part of the system is shown in green on the following diagram:



Dilution air is provided by suction on laboratory fume hoods. This air enters the system through the fume hoods, goes through the dilution blower, and enters the base of the stack, where it mixes with confinement air. This system is always on, regardless of whether the confinement exhaust system is in normal operating mode, or emergency operating mode. This part of the system is shown in blue.

Under emergency conditions, the normal exhaust system is shut off, dilution air continues to be provided, and air exits confinement through the emergency exhaust intake. The air goes through the emergency charcoal filter, the emergency blower, and enters the base of the stack, where it mixes with dilution air. This part of the system is shown in red.

14.174 TS 5.4 mentions dilution air from non-reactor building sources. Explain the reason for not establishing a quantitative LCO for dilution air, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

The new analysis for the maximum hypothetical accident in Chapter 13 does not take credit for dilution air. Consequently, no LCO is necessary. See the basis document entitled "Fuel Damage Radiological Assessment".

14.175 TS 5.4 mentions exhaust air from the reactor building. Explain the reason for not establishing a quantitative LCO for exhaust air, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

The confinement system is dependent on a negative differential pressure of 0.5 inches of water across confinement. This differential pressure is achieved by the confinement exhaust blower. As long as the blower has a sufficient flow rate to maintain the 0.5 inch differential pressure, the blower exhaust flow rate is adequate for achieving its intended function. The blower flow rate is not measured. The underlying LCO parameter that is measured is the differential pressure. Consequently, the LCO is the differential pressure rather than the exhaust blower flow rate.

14.176 Proposed TS 5.5 references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed TS will become part of the TS and license. Explain why it is necessary to make the referenced section of the SAR a requirement in the proposed TS.

Eighth Response Submitted January 24, 2011

Chapter 14, section 5 has been re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.172. References to the SAR have been removed.

The following RAI relates to financial qualifications.

1. Exhibit 3, "Summary of Decommissioning Calculations," of the supplement to the application dated January 19, 2010, provided a 20-year SAFSTOR scenario for the Rhode Island Nuclear Science Center (RINSC), with three columns listed for "Assumed Decom escalation factor," with a 5 percent escalation factor for "Management Maintenance & Supervision," "Assumed Discount Rate Factor," with a 2.85 percent discount rate factor, and "Assumed Discount Rate Factor," with a 0 percent (no discounting) discount rate factor. As discussed during the phone conversation of March 3, 2010, the NRC staff requires the following supplemental information to the Rhode Island Atomic Energy Commission Request for Additional Information response dated January 19, 2010:

(a) Clarify what the costs in "Management Maintenance & Supervision" represent and whether the annual costs associated with SAFSTOR are accounted for in the analysis. Also, identify the basis for the use of the $800,000 base cost in year 1.

First Response Submitted June 10, 2010

The costs in "Management Maintenance & Supervision" are the annual costs associated with SAFSTOR. In the first 3 years, it is estimated that the equivalent of the entire RINSC staff will be required to initialize a regular process including a maintenance and supervision routine for the SAFSTOR program; the staff cost basis then is approximately equal to our present staff costs. Beyond this time frame a SAFTOR staff is expected to be reduced by a factor of 4.

(b) The RAI response states that "[a] figure of 25% was used beginning in year 4 and continuing through year 20," however the NRC staff notes that the figure in year 4 is not 25 percent of the figure from year 2. Clarify this inconsistency.

First Response Submitted June 10, 2010

As noted in (a) above, the figure in year 4 is 25% of that in year 3 (not year 2) and this is adjusted for assumed escalation of 5%.

(c) Explain how the $5,866,092 in year 1 of Column 2, Exhibit 3, was determined. Also, provide a numerical example showing how this number was determined.

First Response Submitted June 10, 2010

The $5,866,92 was determined by adding the following (with no discounting as it is year 1).

Labor A	RW A	Labor BC	RW BC to A	RadCon + Release Survey	Management Maintenance & Supervision
$834,324	$3,716,032	$105,336	$0	$410,400	$800,000

The sum of these is $5,866,92.

(d) Explain the method used to perform discounting in Column 2, Exhibit 3. Also, provide a numerical example showing the method used to perform discounting using the 2.85 percent discount rate factor.

First Response Submitted June 10, 2010

If the sum of the costs noted in c above occurring in year, n, is C_T, then the present value of these costs, PVC(n), at a discount factor, i, is

$$PVC(n) = C_T \times \frac{1}{(1+i)^{n-1}}$$

and the sum of these is the net present value of all costs, NPVC,

$$NPVC = \sum_{n=1}^{20} PVC(n)$$

For example, for year 8, PVC, is calculated as follows:

$$PVC(8) = \$281{,}420 \times \frac{1}{(1+0.0285)^{8-1}} = \$231{,}167$$