RHODE ISLAND ATOMIC ENERGY COMMISSION
RESEARCH REACTOR
LICENSE NO. R-95
DOCKET NO. 50-193

RESPONSES TO NRC STAFF REQUEST FOR ADDITIONAL INFORMATION FOR LICENSE RENEWAL REVIEW

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

Bioassays may be required for anyone handling or using unsealed radioactive sources. Bioassays are required for adults likely to receive an annual intake in excess of 10 percent of the applicable annual intake limits.[1] Bioassays will also be required for minors and declared pregnant women likely to receive an annual intake in excess of 1 percent of the annual intake limits for adults.

The annual limit on intake (ALI) is the activity of an intake of radioactive material that if taken alone would irradiate a person,[2] to the limit set for each year of occupational exposure.

The dose equivalents are recorded annually on a clear, legible record containing all of the information required by NRC Form 5.The accuracy and precision of our dosimetry service is independently tested by the National Institute of Standards and Technology, in accordance with American National Standards Institute ANSI N13.11-2009. Our dosimetry service is fully accredited in all testing categories for Ionizing Radiation Dosimetry by NVLAP (United States Department of Commerce, NIST, National Voluntary Laboratory Accreditation Program) for satisfactory compliance with criteria established under Title 15, Part 285, Code of Federal Regulations.

Found in Rhode Island Department of Health's Rules and Regulations For The Control of Radiation, Part A, Appendix B and 10 CFR 20, Appendix B.

[1] Represented by Reference Man

13.1 Section 13.1 lists nine credible accidents for research reactors based on the guidance in NUREG-1537, but only provides analyses for seven types of accidents. Provide analyses of the omitted accidents, or provide justification for not analyzing the omitted accidents.

Fourth Response Submitted September 8, 2010

The two credible accidents from the list of nine credible accidents shown in NUREG-1537 are accidents involving mishandling or malfunction of fuel and experiment malfunction. The mishandling or malfunction of fuel is an initiating event for the Maximum Hypothetical Accident (MHA), which was analyzed. The major accident involving experiments is related to an unanticipated reactivity insertion. A new analysis for a rapid insertion of the maximum reactivity worth of all experiments of 0.6% $\Delta k/k$ (TS 3.1.3) is provided in RAI 13.7.

13.2 The analysis of the maximum hypothetical accident (MHA) does not include radiation doses to personnel inside the reactor building. Provide an analysis of radiation doses to the personnel inside the reactor building. Discuss all assumptions used in the analysis, including justification for the use of the assumptions.

Third Response Submitted August 18, 2010

The answer to this question is covered in the basis document entitled "Radiological Assessment Attachment".

13.3 Table 13-3, column 2 gives the release rate of iodine isotopes from the reactor stack during the MHA. Explain how the release rate was calculated, including all assumptions
regarding confinement building volume and emergency exhaust system flow rate. Explain how the analysis is consistent with the requirements in the TS. Provide an example calculation. Explain whether the same method and assumptions used for the iodine release rate analysis was used for the whole body gamma dose analysis. If not, explain the method and assumptions used for the whole body gamma dose analysis and provide an example calculation. (See RAI 14.97)

Third Response Submitted August 18, 2010

The answer to this question is covered in the basis document entitled "Radiological Assessment Attachment".

13.4 The footnote of Table 13-5 indicates an assumed reduction of 10% of radioiodine by the reactor pool. Page 13-4 indicates a release of 1% of radioiodine from the reactor pool. Explain this apparent inconsistency.

Third Response Submitted August 18, 2010

The answer to this question is covered in the basis document entitled "Radiological Assessment Attachment".

13.5 The footnote of Table 13-5 indicates a 50% reduction of noble gases. Explain the reason for the reduction in noble gases.

Third Response Submitted August 18, 2010

The answer to this question is covered in the basis document entitled "Radiological Assessment Attachment".

13.6 Section 13.2.2 of the SAR references Figure 13.1, but the figure does not appear in the SAR. Provide a copy of this figure.

Seventh Response Submitted December 14, 2010

ANL generated a new analysis for a 0.6 % dk/k reactivity insertion accident since total experiment worth is limited to this much excess reactivity. This section of the SAR will be re-written based on this analysis. See the response to RAI question 13.7. Figure 13.1 of the SAR references a graph of the core power and peak cladding temperature with time. This figure was not re-generated for the new analysis.

13.7 Section 13.2.2 of the SAR states that ANL performed a PARET analysis of reactivity insertions, but there is no reference provided for the PARET analysis. Provide a copy of the referenced calculation, including initial conditions and assumptions used in the analysis. If available, provide a copy of the PARET input deck.

Fourth Response Submitted September 8, 2010

For these RAI, four reactivity insertion transients were re-analyzed for forced convection cooling mode, and one reactivity insertion was analyzed for natural convection mode using the PARET/ANL Version 7.5 code.

PARET/ANL Version 7.5 solves the point-kinetics equations for reactor power versus time, while computing thermal-hydraulic conditions in one or more fuel channels. Feedback from the thermal-hydraulic solution is continuously fed back into the point-kinetics equations. The reactor is modeled using two hydraulic channels: a hot channel, and an average channel. The hot channel represents conditions in the coolant channel between fuel plates that is most limiting. This channel typically has fuel plates having the highest power density adjacent to it. The average channel represents all other coolant channels adjacent to fuel plates. Reactivity feedback from fuel heatup (Doppler Effect), water heatup, and water density change, are accounted for using feedback coefficients derived from the neutronics models. The input to the PARET model consists of several categories of information:

1. geometry of the channels (fuel meat, clad, coolant)
2. delayed neutron kinetics data
3. reactivity insertion definition
4. control system response
5. initial operating conditions of power and flow
6. solution options such as time step sizes and edit selection.

The neutronics codes used to generate input for the PARET models were:

WIMS/ANL for multi-group neutron cross sections; REBUS-PC (which includes DIF3D as the neutronics solver) for power density information, and VARI3D (which also includes DIF3D as the neutronics solver for real and adjoint flux), to provide the reactor kinetics delayed neutron fractions, decay constants, and prompt neutron lifetime. Data on reactor power distribution is provided in Reference 2, and data on the reactor kinetics parameters and reactivity feedback coefficients is provided in RAI 4.10.

The forced convection transients are assumed to take place under the following assumptions [Technical Specifications, Revised Section 2.1.1 in RAI 14.36, "Safety Limits in the Forced Convection Mode]:

Measured Parameter	Limiting Trip Value	Safety Limit
P	2.3 MW	2.4 MW
m	1740 gpm	1580 gpm
H	23' 9.1"	23' 6.5"
T_o	123 F	125 F

The natural convection transients are assumed to take place under the following assumptions [Technical Specifications, Revised Section 2.1.2 in RAI 14.52, "Safety Limits in the Natural Convection Mode]":

Measured Parameter	Limiting Trip Value	Safety Limit
P	125 kW	200 kW
H	23' 9.1"	23' 6.5"
T_o	128 F	130 F

The period trip at 4 seconds is assumed to fail. The power trip is functional. The time delay for control blades to begin to move after a trip is assumed to be 100 ms. The time to full insertion is the maximum allowed of 1.0 second [TS 3.2.3].

Case 1: Rapid Insertion of 0.6% Δk/k Reactivity From Very Low Power

The reactor is initially operating at 10 watts, 123 °F coolant inlet temperature, and 1740 gpm. There is a water head of 23' 9.1" above the top of the fuel meat, which provides a pressure of 1.715 x 10^5 Pa. Then 0.6% Δk/k reactivity, the total reactivity worth of all experiments [TS 3.1.3], is inserted as a very short ramp of 0.1 second duration, starting at 0.0 seconds. The reactor power rises rapidly. The power trip at 2.3 MW is actuated at 10.179 s. Since no actual negative reactivity from the control system occurs for 100 ms after the trip, the reactor power continues to rise from the trip level of 2.3 MW to a maximum of 2.423 MW at 10.279 s and the control blades are inserted. The reactor power drops rapidly to shutdown conditions.

Peak temperatures for fuel meat centerline, clad surface, and coolant are: 79.8 °C; 79.1 °C; and 63.6 °C, respectively, at 10.30 s. These peak fuel and clad surface temperatures are far below the maximum temperature of 530 °C for LEU silicide fuel that the NRC finds acceptable as fuel and clad temperature limits not to be exceeded under any conditions of operation (See NUREG-1537, Part I, Appendix 14.1 and NUREG-1313). The peak coolant temperature is well below the saturation temperature of 115.4 °C.

Case 2: Slow Insertion of 0.02 % Δk/k /Second Reactivity From Very Low Power

The reactor is initially operating at 10 watts, 123 °F coolant inlet temperature, and 1740 gpm. There is a water head of 23' 9.1" above the top of the fuel meat, which provides a pressure of 1.715 x 10^5 Pa. Then a long, slow ramp reactivity insertion begins at a ramp rate of 0.02 % Δk/k / s (TS 3.2.4), continuing for 100 s. Power rises slowly. The power trip at 2.3 MW is actuated at 32.198 s. Since no actual negative reactivity from the control system occurs for 100 ms after the trip, the reactor power continues to rise from the trip level of 2.3 MW to a maximum of 2.509 MW at 32.298 s. The reactor power drops rapidly to shutdown conditions.

Peak temperatures for fuel meat centerline, and clad surface are: 79.1 °C; 78.9 °C. The peak coolant temperature of 62.8 °C is reached at 32.40 s. These peak fuel and clad surface temperatures are far below the maximum temperature of 530 °C for LEU silicide fuel that the NRC finds acceptable as fuel and clad temperature limits not to be exceeded under any conditions of operation (See NUREG-1537, Part I, Appendix 14.1 and NUREG-1313). The peak coolant temperature is well below the saturation temperature of 115.4 °C.

The safety limit on power of 2.4 MW is exceeded (the power briefly reaches 2.509 MW). However, the safety limit on power does not apply to transients. In this case, the fuel meat and cladding reach peak temperatures of about 79 C, far below the maximum allowed temperature of 530 °C

Case 3: Slow Insertion of 0.02 % Δk/k / Second Reactivity From 1.8 MW Power

The reactor is initially operating at 1.8 MW, 123 °F coolant inlet temperature, and 1740 gpm. There is a water head of 23' 9.1" above the top of the fuel meat, which provides a pressure of 1.715 x 10^5 Pa. The coolant inlet temperature for which an outlet temperature of 123 °F is reached was iteratively determined to be 113.6 °F (45.34 °C). Starting from this initial condition, a long, slow ramp reactivity insertion begins at a ramp rate of 0.02 % Δk/k / s [TS 3.2.4], continuing for 100 s. Power rises slowly. The power trip at 2.3 MW is actuated at 6.774 s. Since no actual negative reactivity from the control system occurs for

100 ms after the trip, the reactor power continues to rise from the trip level of 2.3 MW to a maximum of 2.313 MW at 6.874 s. The reactor power drops rapidly to shutdown conditions.

Peak temperatures for fuel meat centerline, and clad surface are: 76.7 °C; 75.9 °C; and 59.6 °C. The peak coolant temperature of 59.6 °C is reached at 6.90 s. These peak fuel and clad surface temperatures are far below the maximum temperature of 530 °C for LEU silicide fuel that the NRC finds acceptable as fuel and clad temperature limits not to be exceeded under any conditions of operation (See NUREG-1537, Part I, Appendix 14.1 and NUREG-1313). The peak coolant temperature is well below the saturation temperature of 115.4 °C.

Case 4: Slow Insertion of 0.02 % Δk/k /second Reactivity From 2.2 MW Power

The reactor is initially operating at 2.2 MW, 123 °F coolant inlet temperature, and 1740 gpm. The coolant inlet temperature for which an outlet temperature of 123 °F is reached was iteratively determined to be 111.5 °F (44.19 °C). There is a water head of 23' 9.1" above the top of the fuel meat, which provides a pressure of 1.715×10^5 Pa. Starting from this initial condition, a long slow ramp reactivity insertion begins at a ramp rate of 0.02 % Δk/k / s [TS 3.2.4], continuing for 100 s. Power rises slowly. The power trip at 2.3 MW is actuated at 2.498 s. Since no actual negative reactivity from the control system occurs for 100 ms after the trip, the reactor power continues to rise from the trip level of 2.3 MW to a maximum of 2.308 MW at 2.598 s. The reactor power drops rapidly to shutdown conditions.

Peak temperatures for fuel meat centerline, and clad surface are: 75.9 °C; 75.1 °C; and 58.5 °C. The peak coolant temperature of 58.5 °C is reached at 2.600 s. These peak fuel and clad surface temperatures are far below the maximum temperature of 530 °C for LEU silicide fuel that the NRC finds acceptable as fuel and clad temperature limits not to be exceeded under any conditions of operation (See NUREG-1537, Part I, Appendix 14.1 and NUREG-1313). The peak coolant temperature is well below the saturation temperature of 115.4 °C.

Case 5: Rapid Insertion of 0.6% Δk/k Reactivity From 100 kW Under Natural Convection Cooling

The reactor was brought up to 100 kW under natural convection conditions with a maximum outlet temperature of 128 °F. There is a water head of 23' 9.1" above the top of the fuel meat, which provides a pressure of 1.719×10^5 Pa. Power and flow are allowed to stabilize out to 360 s, at which time the power is 100 kW. Then a very short reactivity ramp of 0.6% Δk/k is inserted over 0.1 s, starting at 360.00 s. The power trip at 125 kW is actuated at 360.036 s. The reactor power continues to rise from the trip level of 115 kW to a maximum of 404 kW at 360.140 s. The reactor power drops rapidly to shutdown conditions.

Peak temperatures for fuel meat centerline (65.7 °C) and clad surface (65.7 °C) occur at 360.18 s, whereas the peak coolant temperature (62.2 °C) occurs at 59.4 s during the rise to power. These peak fuel centerline and clad surface temperatures are far below the maximum temperature of 530 °C for LEU silicide fuel that the NRC finds acceptable as fuel and clad temperature limits not to be exceeded under any conditions of operation (See NUREG-1537, Part I, Appendix 14.1 and NUREG-1313). The peak coolant temperature is well below the saturation temperature of 115.4 °C.

The safety limit on power of 200 kW is exceeded (the power briefly reaches 404 kW). However, the safety limit on power does not apply to transients. In this case, the fuel centerline and clad surface reach peak temperatures of about 66 °C, far below the maximum allowed temperature of 530 °C.

Reference

1. A. P. Olson, A USERS GUIDE TO THE PARET/ANL V7.5 CODE, May 1, 2010, GTRI-Conversion Program, Nuclear Engineering Division, Argonne National Laboratory Internal Memorandum, May 1, 2010.

2. Memo dated 3 September 2010 from Earl E. Feldman to James E. Matos entitled "Steady State Thermal Hydraulic Analysis for Forced Convective Flow in the Rhode Island Nuclear Science Center (RINSC) Reactor"

13.8 Section 13.2.2 states that a 200 millisecond delay was used as a conservative assumption for the time for the control blades to begin to insert following a scram. However, TS 3.2.3 specifies that the full control blade insertion time is 1 second, and does not specify a maximum control blade insertion delay time. Explain this apparent inconsistency between the SAR and the proposed TS.

First Response Submitted June 10, 2010

This question confuses the fact that the 200 msec cited is a very conservative estimate of the delay from the time that the scram signal is initiated, to the time that the blades begin to drop, while the 1 sec insertion time represents the total amount of time that it takes from the initiation of the scram signal, to full insertion of the control blade.

In an effort to be consistent, all of the new analysis will be done with the assumption that the delay time between the initiation of the scram signal, and the time that the control blades begin to drop is 100 msec. This is considered to be conservative

13.9 Section 13.2.2 of the SAR states that during a reactivity insertion, the onset of nucleate boiling is approached, but does not occur. Provide quantitative details regarding the approach to nucleate boiling that show that the safety limits are not exceeded. (See RAI 14.62)

Fourth Response Submitted September 8, 2010

See the analysis of reactivity insertions provided in RAI 13.7

13.10 Section 13.2.3 presents the LOCA analysis for a break in a beam port. Provide justification that this beam port failure is the limiting initiating event for a LOCA.

First Response Submitted June 10, 2010

The only open penetrations into the pool are the rabbit, through port, and beam port tubes. The rabbit tubes enter through the pool wall at an elevation that is close to the top of the pool. Consequently, shearing open a rabbit tube will not lead to significant draining of the pool. Dropping something into the reactor pool, and shearing the through port is not considered to be a credible accident scenario because it runs underneath the thermal column extension. As a result, the beam ports are used for the LOCA analysis, and the assumption is made that the beam port extension for one of the largest beam ports is sheared off. In all of these cases, the likelihood of dropping anything into the pool that causes this kind of damage is very low because there is a steel plate bridge over the top of the core.

13.11 The calculation of pool drain time in Section 13.2.3 makes assumptions about the design of and administrative controls for use of the beam ports and through-port. Propose TS requirements for the design and operation of the beam ports and through-port that are consistent with the assumptions made in the analysis of a LOCA, or provide justification for not including such TS requirements.

Tenth Response Submitted July 15, 2011

The Addendum LOCA analysis shows that as long as the area between each individual experimental port and confinement is no greater than 1.48 in^2, then there is sufficient pool drain time to allow for decay power to reach the point at which the fuel cladding cannot be compromised. However, this assumes that the water level will not drop below the elevation of the bottom of the eight inch beam ports. The elevation of the bottom of the through port is below the elevation of the bottom of the eight inch beam port, and an analysis for a LOCA in which the fuel is completely un submerged has not been performed. The answer to RAI question 10.2 shows that administrative controls on the use of the through port will prevent conditions from occurring that could lead to a LOCA that has not been analyzed. Therefore, the administrative controls will be set conservatively to say that:

1. Each beam port shall have no more than an area of 1.25 in^2 open to confinement during reactor operation.

2. When the reactor is in operation, the drain valve to the through port shall be closed.

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

4. When the through port is not being monitored for a leak condition, the ends of the port shall be closed.

The bases for these specifications will be that:

Specification 1:

The LOCA analysis shows that as long as the pool level does not drain through an area greater than 1.48 in^2 to confinement, then there will be sufficient time for decay power to drop to a point which will not damage the fuel cladding, provided that the pool level does not drop below the elevation of the bottom of the eight inch beam ports. It also shows that if any single port has a catastrophic failure, the un-damaged ports do not become pool drain pathways. Consequently, limiting the areas of each experimental port that is open to confinement to 1.25 in^2 is conservative.

Specification 2:

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

Specification 3:

If the end(s) of the through port that will be used for access have gate valves mounted to them, then in the event of a leak, the port can be easily isolated so that the leak is stopped.

Specification 4:

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

13.12 Line 32 of page 13-10 states that a coolant height of "139.4 feet (normal water level of pool)" was used as the initial coolant height in the LOCA analysis. Explain why this coolant height is consistent with the limiting safety system setting for coolant height given by TS 2.2.1 and the set point for the pool water level safety channel required by TS 3.2.1. (See RAI 14.72)

Fifth Response Submitted November 26, 2010

Because the normal water level of the pool is greater than the limiting safety system setting for coolant height given by TS 2.2.1 and the set point for the pool water level safety channel required by TS 3.2.1, it should not be used for the initial coolant height in the LOCA analysis. Section 13.2.3, Loss of Coolant Accident (LOCA), has been completely replaced and the replacement is attached. In the new analysis the coolant level at which the scram occurs is 23.54 feet above the top of the active core, which in the new analysis is taken to be the top of the fuel meat. This level is the minimum pool level that is permitted by the Safety Limits while operating at any force convection power level, TS 2.1.1.

13.13 Page 13-10. Provide definitions for the terms h1, h2, and C in the equation on line 43.

First Response Submitted June 10, 2010

C = coefficient of discharge, which is dependent on the type of orifice through which the water is draining. We assume the orifice is a standard sharp-edged orifice, which has a discharge coefficient of 0.61. The reference for this is Mark's Mechanical Engineer's Handbook, Theodore Baumeister (editor), McGraw-Hill, New York, 1958. P. 3-62.

h1 = upper elevation of the water level

h2 = drain elevation

Fifth Response Submitted November 26, 2010

Revised 08/26/10

h1 is the initial water level. h2 is the final water level, which is located at the bottom of the failed beam port. C is the orifice coefficient for the assumed ½-inch diameter hole through which water flowing through the failed beam port exits the pool. However, section 13.2.3, Loss of Coolant Accident (LOCA), has been completely replaced and the replacement is attached. In the new analysis of the draining of the pool, section 13.2.3.2, "Drain Time," the variables are defined and the model is derived from first principles. In the new analysis, h1, h2, and C are replaced by h_1, h_f, and C_d, respectively.

13.14 Page 13-13. The boundary condition of 1,200 degrees F used in the calculation is not consistent with the cladding blistering temperature of 986 degrees F, which is the criterion for fuel damage found in the literature for U_3Si_2 fuel. Provide an analysis using the fuel blistering temperature, or provide a discussion of why the boundary condition of 1,200 degrees F is conservative.

Fifth Response Submitted November 26, 2010

Section 13.2.3, Loss of Coolant Accident (LOCA), has been completely replaced and the replacement is attached. In the new analysis in section 13.2.3.14, 986° F (530° C) is identified as the temperature limit for the fuel plates during the LOCA. In section 13.2.3.13 of the new analysis, a maximum fuel temperature of 486° C, which is less than the temperature limit, is predicted during the postulated LOCA.

13.15 Page 13-13. Please provide a conclusion for the analysis ending on line 18 of this page.

Fifth Response Submitted November 26, 2010

The conclusion for the analysis ending on line 18 of this page, as is indicated symbolically on lines 13 through 18 of page 13-13, is that 0.013 Btu/sec would be conducted vertically downward to the submerged portion of the fuel plate through the fuel meat part of the fuel plate if the maximum fuel temperature (at x = 2.0') is 1200° F and the temperature of the fuel plate at the water surface (at x = 0.7') is 212°F. However, section 13.2.3, Loss of Coolant Accident (LOCA), has been completely replaced and the replacement is attached. The analysis ending on line 18 of Page 13-13 of the 2004 RINSC safety analysis report belongs to the fuel assembly heat transfer model of that report. A new heat transfer model is provided in the new analysis in section 13.2.3.4, "Development of Heat Transfer Model".

13.16 Page 13-13. The analysis assumes that the decay power spatial distribution can be approximated by a sinusoidal curve. Provide justification for this assumption.

Fifth Response Submitted November 26, 2010

The power distribution in the reactor along the length of the fuel tends to have a shape that can be approximated by a sinusoid with the peak value near the middle of the length. In such situations, often a chopped cosine is typically used. However, section 13.2.3, Loss of Coolant Accident (LOCA), has been completely replaced and the replacement is attached. The new analysis assumes that the axial distribution of the decay power spatial distribution is uniform. The new analysis explains why a uniform distribution bounds the worst case. Specifically, the new section 13.2.3.4 states:

> An issue is the axial distribution of the power along the length of the fuel plate. Typically, the distribution will tend to be in the shape of a symmetric chopped cosine shape whose peak is near the center of the 23.25-inch fuel meat length. Since the center of the fuel meat length is 4 inches above the water level, within the exposed length of the fuel plate the power is skewed toward the bottom. The more that power is skewed toward the bottom, the lower the peak solid temperature will be. The reason for this is that heat generated lower in the exposed portion of the fuel plate length has a shorter conduction path to the surface of the water than does heat generated higher in the exposed portion. Moreover, as is obvious, lower fuel temperatures will also be produced if a greater portion of the heat produced over the entire fuel plate is generated below the surface of the water. Figure 4.6-5 (of the currently replaced section 4.6 of the RINSC reactor safety analysis report, "4.6 Steady-State Thermal-Hydraulic Analysis" Reference BB), provides the axial power shape for the fuel meat length of the highest power plate, which is the one next to the beryllium reflector in assembly D6. In order to keep the model simple and avoid the issue of the precise axial power shape, we will use a bounding approach and assume that the heat generation rate in the fuel plate is uniformly distributed over the entire length of the fuel meat. The water level is at 0.672, i.e., 15.625 inches/23.25 inches, in the Figure 4.6-5. Numerical integration of the axial power in the figure shows that 56.6 % of the power is generated in the exposed portion of the fuel meat length and the remaining 43.4% is generated in the submerged portion. This is to be compared with the uniform distribution, in which 67.2% of the power is generated in the exposed part of the fuel meat length and the remaining 32.8% is generated in the submerged portion.

13.17 The calculation of "Heat Conduction to the Water in Core Box from the Non-Fuel Aluminum in the Element" appears incomplete. Provide the remainder of the calculation, a discussion of the results of calculation, all assumptions made in the calculation and justification for those assumptions, and any conclusions based on the calculation.

Fifth Response Submitted November 26, 2010

The missing text from "Heat Conduction to the Water in Core Box from the Non-Fuel Aluminum in the Element" is provided in Figure 13.17-1 in the box below. It was copied from Appendix D of the *Safety Analysis Report for the Low Enriched Fuel Conversion of the Rhode Island Nuclear Science Center Research Reactor*, Revision 1, December, 1992. However, Section 13.2.3, Loss of Coolant Accident (LOCA), of the 2004 RINSC safety analysis report has been completely replaced and the replacement is attached. Although the new analysis does not have a "Heat Conduction to the Water in Core Box from the Non-Fuel Aluminum in the Element" section, it is complete, the results of the calculations are discussed and all assumptions made in the calculations and justifications for those assumptions, and any conclusions based on the calculations are provided. See, for example, section 13.2.3.4, "Development of Heat Transfer Model," of the attached replacement.

Figure 13.17-1 Missing Text

$$Q = \frac{131 \times 0009 \times (1200-212)}{l = (2-.7) = 1.3'} = 89.604 \text{ Btu/hr} \times 1/3600$$

$$Q = .02489 \text{ Btu/sec}$$

total heat conducted = fuel + aluminum
= .013 + .02489 = .03789 Btu/sec

From the original SAR it was assumed that about 30% of the heat was used in steam formation

therefore .3 x .03789 = .011367 Btu/sec

and the total heat removal = .03789 + .011367
= .049 Btu/sec

Since the heat generation of.0397 Btu/sec is less than that required to reach the plate melting point (.049 Btu/sec), it is assumed that the fuel does not reach the melting point.

13.18 Section 13.2.4 mentions a low flow alarm and a low flow trip that are inconsistent with the requirements of the TS. Provide analyses of loss-of-coolant-flow accidents that are consistent with the requirements of the TS, or propose TS requirements that are consistent with the current analyses.

Fourth Response Submitted September 8, 2010

See the response to Question 13.19.

13.19 Section 13.2.4.1 states that the peak clad temperature during a loss-of-flow-accident induced by a loss of electrical power is 103 degrees C. Provide an analysis that supports this statement. Justify all assumptions made in the analysis.

Fourth Response Submitted September 8, 2010

RELAP5 cases were run to analyze loss-of-coolant-flow accidents in the RINSC reactor. These cases were consistent with the Limiting Safety System Settings in the Forced Convection Mode. Cases were run both with and without proper opening of the natural circulation gate valves during the transient.

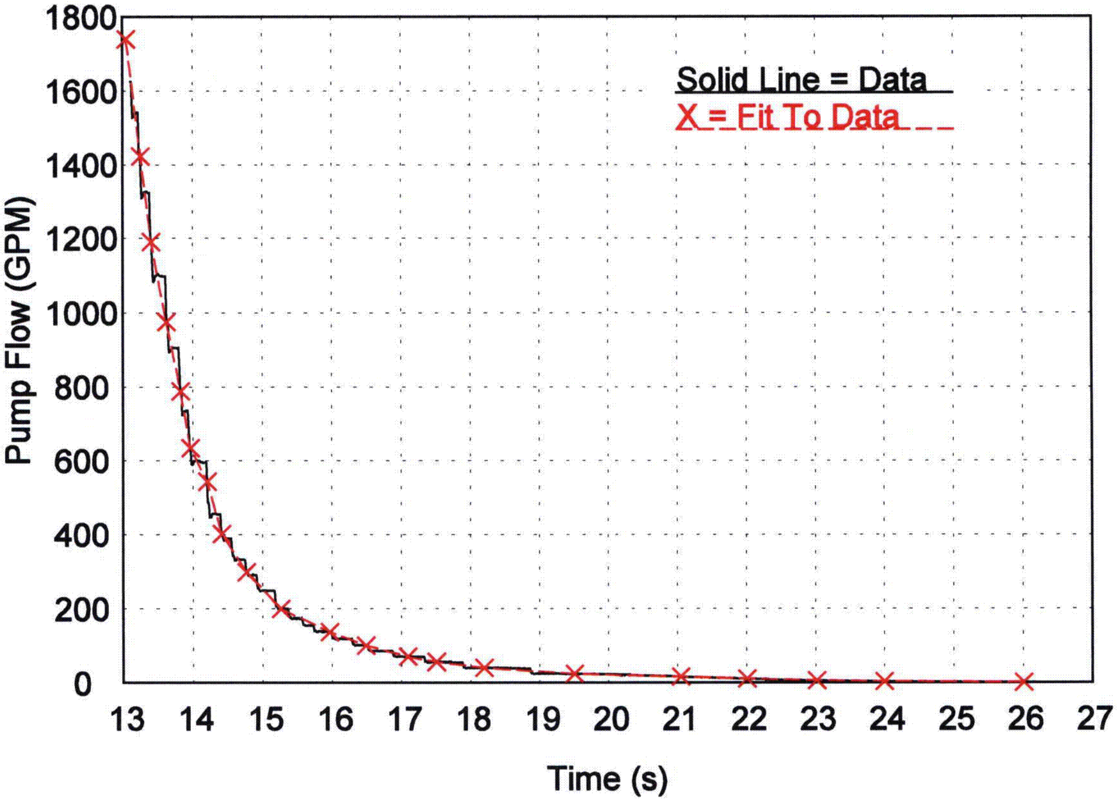
The RELAP5 model included the peak fuel channel, representing the highest power stripe in the highest power plate and its coolant. This model also included the average fuel channel, representing the rest of the fuel plates and their coolant. Bypass flow around the graphite reflectors, through the beryllium reflectors, and inside the control blade shrouds was represented in the model. Also bypass flow through the gamma shield was represented. The model included the pump, the heat exchanger and the associated piping. The model also included the coolant ducts between the piping and the core box, as well as the natural circulation gate valves in the duct walls. The pool was also represented.

The initial steady-state conditions for the calculations were set at the hottest conditions that might not trip a scram. These conditions were taken from the limiting trip values in the Limiting Safety System Settings in the Forced Convection Mode. These conditions are listed in the table below.

Initial Steady-State Conditions for the LOF Cases

Parameter	Value
Reactor power	2.3 MW
Total pump flow	1740 GPM
Height of water above the top of the core	23 ft. 9.1 in.
Outlet temperature	123 F

The transient was initiated by a loss of power to the pump. RINSC personnel have measured the pump flow after a pump trip. The measured results are shown in the figure below. Also shown is a smoothed fit to the data. The smoothed fit was used to specify the pump flow in the RELAP5 calculations.



Measured Pump Coastdown Curve

Key parameters of the transient calculation are given in the table below.

Key Parameters for the Loss-of-Flow Transients

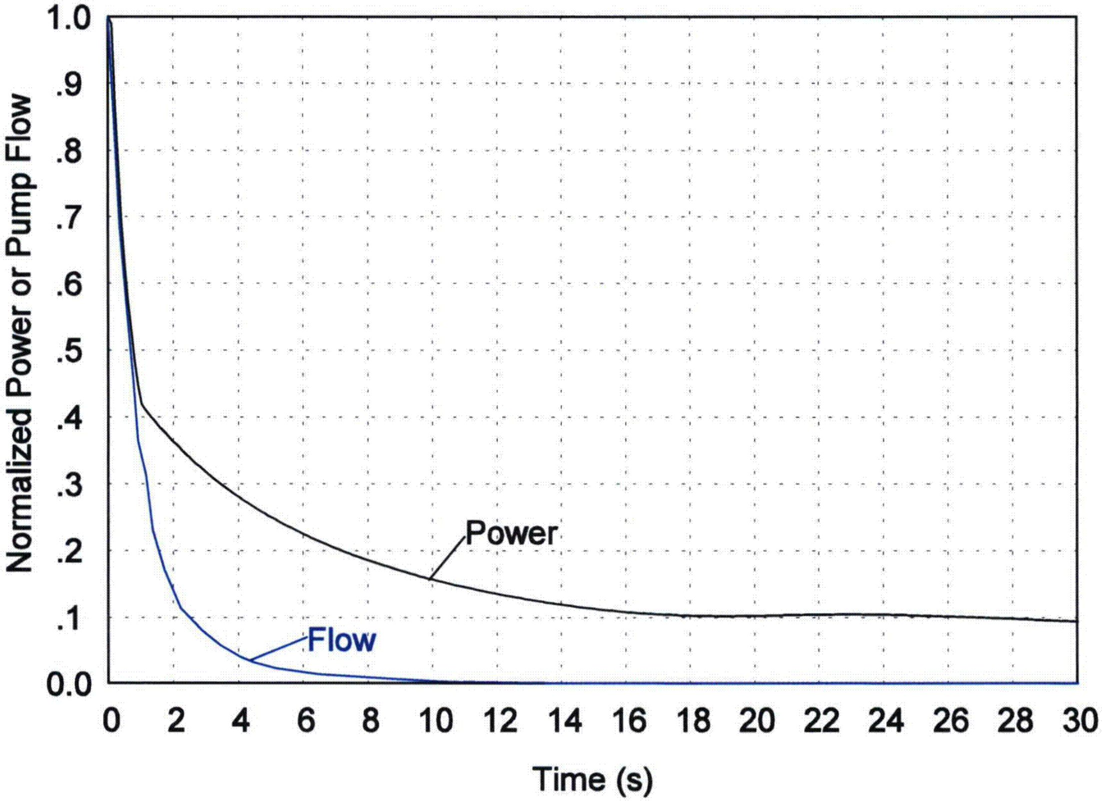
Parameter	Value	Justification
Pump coast-down	As shown in figure above	Measured
Time when natural circulation gate valves open	9 seconds after start of pump coast-down	Measured
Scram reactivity	1 % $\Delta k/k$	Limiting Conditions for Operation
Scram reactivity insertion time	1 second	Limiting Conditions for Operation

For the base case, the one in which the natural circulation gate valves open properly, the timing of transient events is shown in the table below.

Loss-of-Flow Transient Timing, Gate Valves Open Properly

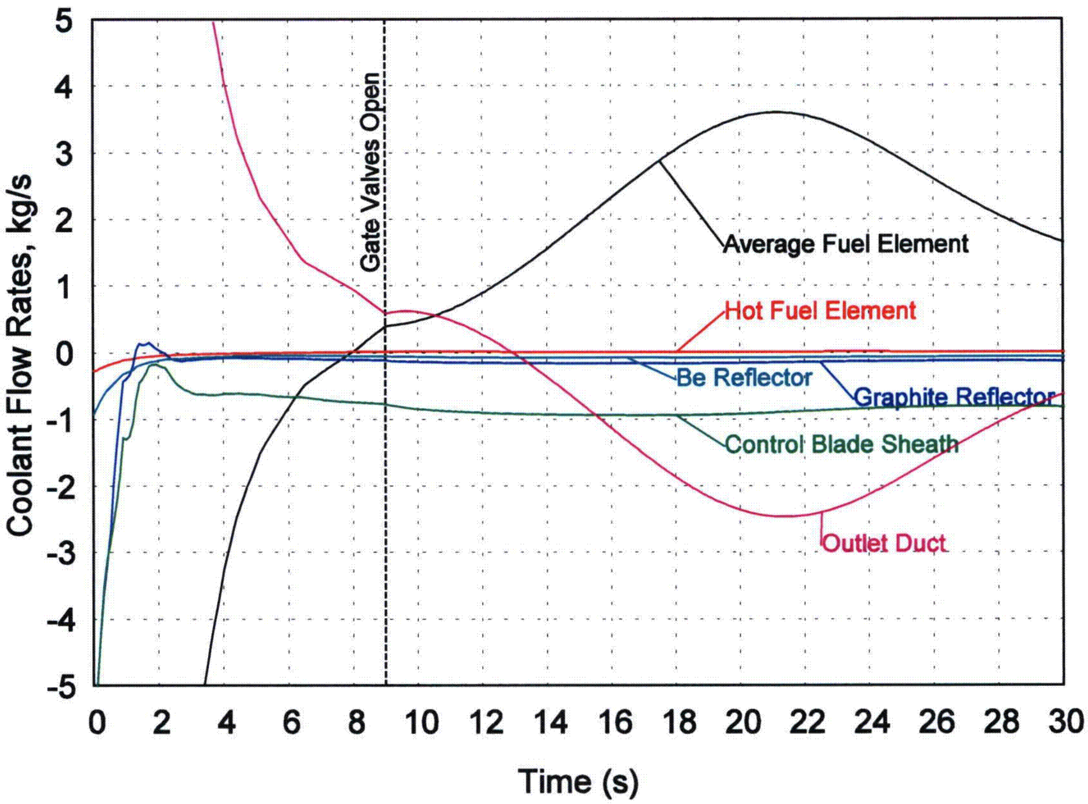
Time, s	Event
0.0	Pump trips
0.0	Low flow scram tripped immediately
0.1	Control blades start moving
1.0	Control blades fully inserted
5.31	Flow reversal in peak fuel channel
8.01	Flow reversal in average fuel channel
9.0	Natural circulation gate valves open
9.41	Clad surface temperature peaks at 115.62 C in middle of core, peak fuel temperature = 115.73 C, coolant saturation temperature = 115.90 C at this point.

The normalized power and pump flow for the base case are shown in the figure below. For the first second, while the control blades are inserting, the power and flow fall at about the same rate. After that the power falls slower, and the pump flow drops to zero.



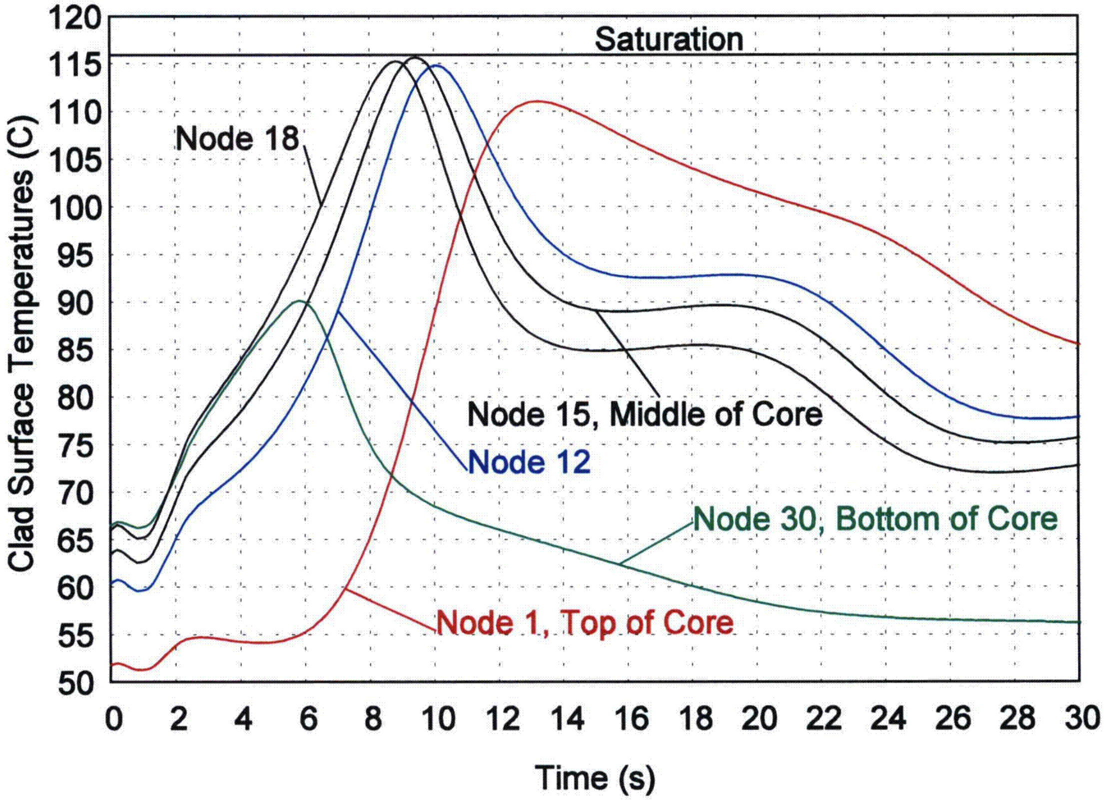
Power and Pump Flow for the RINSC LOF

Flow rates through the fuel channels, the reflectors, the control rod sheaths and the outlet duct are shown in the figure below. In this figure positive flow rates are upward, and negative flow rates are downward. As indicated in the table above, flow reversal occurs in the average fuel channel at about 8 seconds. During normal forced flow operation both flow around the graphite reflectors and flow through the control rod sheaths contribute significantly to bypass flow, but after the pump coasts down the flow through the control blade sheaths dominates the bypass flow. During forced flow the flow through the outlet duct is upward. During normal natural circulation operation the flow in the outlet duct is downward. During the loss-of-flow transient the flow in the outlet duct is upward until the gate valves open. It then takes a few seconds for the flow to reverse in the outlet duct.



Flow Rates for the RINSC LOF

Clad surface temperatures at a number of nodes in the peak fuel channel are shown in the figure below. The highest clad temperature occurs at node 15 which is at the middle of the core. The peak clad surface temperature almost reaches the saturation temperature, where sub-cooled boiling would start. As indicated in the transient timing table above, the peak fuel centerline temperature is a fraction of a degree hotter than the peak clad surface temperature.



Clad Surface Temperatures in the Peak Fuel Channel

The peak coolant temperatures in this transient are close to the onset of sub-cooled boiling. If the initial power level were higher and the onset of sub-cooled boiling occurred, then the coolant heat transfer coefficient would go up and would limit the rise in fuel and clad temperatures.

The failure of the natural circulation gate valves to open during a loss-of-flow transient would be an extremely unlikely event, but the loss-of-flow transient was repeated with failure of the natural circulation gate valves to open. In the case in which the gate valves do not open the peak clad surface temperature was a fraction of a degree lower (115.61 C vs 115.62 C) and occurred at the same time (9.41 seconds).

The smallness and the sign of the difference in peak clad surface temperature due to the failure of the gate valves to open can be explained by two factors. First, in both cases the peak temperature occurred only 0.41 second after the gate valves should open. The flow rate figure above indicates that the change in the outlet duct flow rate in the first 0.41 second after the gate valves open is small, so the change in peak clad temperature should be small. Second, close examination of the flow rate figure near 9 seconds indicates that if the gate valves open the outlet duct upward flow rate immediately after the gate valves open is slightly higher than what would be obtained by extrapolating the flow rate from before the gate valves open. This leads to slightly more coolant being sucked out of the outlet plenum and slightly less up-flow through the fuel channels, causing slightly higher temperatures in the fuel channels if the gate valves do open.

The question of why the opening of the gate valves initially leads to slightly more up-flow in the outlet duct requires consideration of the factors that determine the flow rate in the outlet duct. Before the gate valves open the flow rate in the outlet duct is caused by the pump sucking coolant upward through the duct at a rate determined by the pump coast-down. When the gate valves first open at 9 seconds the coolant in the outlet duct is still somewhat hotter than the pool temperature. The natural circulation head in the outlet duct leads to natural circulation flow up the duct and partly out the gate valve. The natural circulation up-flow in the duct is partially reduced by the up-flow through the fuel channels. The fuel channel flow sucks coolant out of the outlet plenum and lowers the pressure at the bottom of the duct. The natural circulation duct flow happens to be initially somewhat higher than the pump flow at that time in the coast-down. As time progresses after the gate valves open, the coolant temperatures in the outlet duct decrease and the natural circulation up-flow through the fuel channels increases, leading to flow reversal and down-flow in the outlet duct.

In conclusion, a loss-of-coolant-flow transient in the RINSC reactor would result in peak fuel and clad temperatures that are hundreds of degrees below the temperatures at which damage to the fuel plates would occur.

13.20 Section 13.2.5 provides an analysis of a startup accident, but does not specify assumptions for coolant flow or coolant height. Explain the assumptions used in the analysis for coolant flow and coolant height. Explain how the analysis treated power peaking factors.

Fifth Response Submitted November 26, 2010

The forced convection transients are assumed to take place under the following assumptions [Technical Specifications, Revised Section 2.1.1 in RAI 14.36, "Safety Limits in the Forced Convection Mode]:

Measured Parameter	Limiting Trip Value	Safety Limit
P	2.3 MW	2.4 MW
m	1740 gpm	1580 gpm
H	23' 9.1"	23' 6.5"
T_o	123 F	125 F

The natural convection transients are assumed to take place under the following assumptions [Technical Specifications, Revised Section 2.1.2 in RAI 14.52, "Safety Limits in the Natural Convection Mode]":

Measured Parameter	Limiting Trip Value	Safety Limit
P	125 kW	200 kW
H	23' 9.1"	23' 6.5"
T_o	128 F	130 F

In each case, the Limiting Trip Values are used in the analysis because they are the permitted operating conditions that produce the most extreme fuel and clad temperatures. The Startup Accident has been analyzed under both forced-flow and natural-convection flow conditions. See RAI 13.7, Case 1 and Case 5.

Treatment of Power Peaking Factors:

Within a given fuel plate, there is a variation of power density across the width of the plate and along the length of the plate in the axial direction. Each of the 308 fuel plates in the core is in a different neutron flux environment. The power density is fully 3-dimensional.

In the original neutronics analysis, the DIF3D code edited an estimate of the local peak power density with each fuel volume, based on neutron flux and current gradients. The purpose of this method was to enable the computational model to be small enough to be tractable at that time. The present analysis takes advantage of advances in computational capabilities of the latest version of DIF3D, running on much faster computers with much more memory than existed before.

The new neutronics models captured the 3D effects by a two-step process. In the first step, which is equivalent to the method used in the original analysis, a 3D core model was defined in X-Y-Z geometry, with Z being the axial dimension. There were 17 axial nodes along the length of the fuel meat. In the X-Y plane, this model represented each fuel assembly as three components: two side plates, and a homogenized region consisting of all of the fuel meat, clad, and coolant water associated with the 22 fuel plates inside the envelope of the fuel assembly. Water between the fuel assemblies was separately modeled. The WIMS/ANL code was used to obtain multi-group neutron cross sections for the various compositions in the reactor. Then the DIF3D multi-group diffusion theory neutronics code calculated the neutron flux and power distribution in the core. From this result, the power per fuel assembly was obtained. Fuel assemblies D6 and E6 were found to be the two most-limiting assemblies.

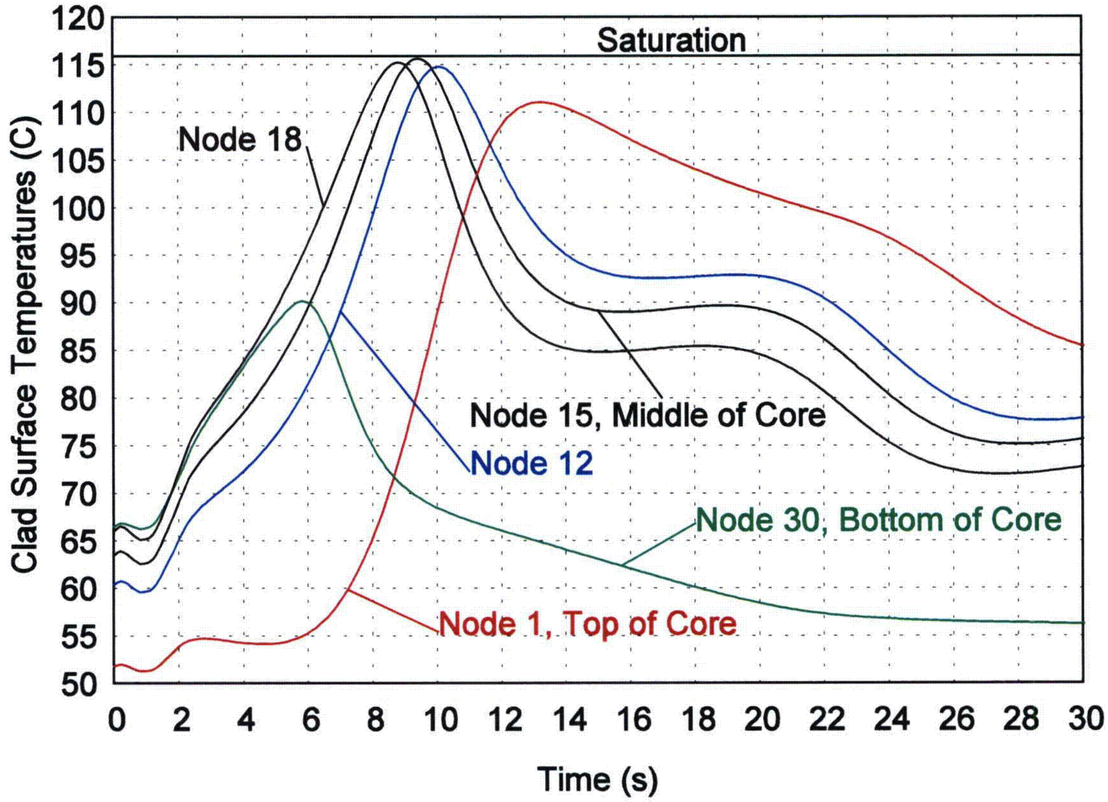
In the second step, the spatial mesh was refined in the X-Y plane through D6 and E6 in order to capture additional spatial detail regarding power distribution with the fuel assembly. The spatial mesh spanning D6 and E6 was increased in order to provide a mesh interval for every plate, and one for each of nine stripes across the width of the plate. In this way, it was possible to determine the precise location and value of the peak power density, rather than using an estimate based on neutron flux and current. Assembly D6, plate 1 (the plate closest to core center), was found to have the largest power density.

The peaking factors for the location with the highest power density can be found as follows. The neutronics results yield a 3D array of power density values by mesh interval. The ratio of any value in this (X,Y,Z) array to the average core power density is the local power peaking factor. The relative power density profile along the hottest stripe of the hottest fuel plate (plate 1) is plotted in Fig. 4.6-5. Figure 4.6-6 shows the variation of relative power density across the width of the plate. Stripe 9 has the largest power density. When this local power peaking factor is divided by the axial power peaking factor from Fig. 4.6-5, one obtains the radial power peaking factor. See Reference BB.

13.21 Figures 13-1, 13-2, 13-3, and 13-4 were not included in the license renewal application. Provide copies of these figures.

Fifth Response Submitted November 26, 2010

A revised Figure 13-1 is provided based on the new transient analyses that were done for these RAI. See the answer to RAI Question 13.19.



Clad Surface Temperatures in the Peak Fuel Channel

Beam Port

Figure 13-2

Beam Port Vent and Drain Connection

[redacted]

Figure 13-3

Figure 13-4 (Schematic Diagram for the Postulated Loss of Coolant Calculations), referenced on page 13-10, line 1, has been deemed unnecessary and out-dated and has thus been removed from the SAR. Chapter 13 will be corrected accordingly.

Section 1.0, "Definitions"

14.1 The proposed TS contain numerous references to a version of the SAR that is different than the version of the SAR submitted with the license renewal application (e.g., TS 4.2.6 references "SAR (Part A, Section V)"). Such references are included in TS 4.1.1.b, 4.2.6, 4.2.7, 4.2.8, 5.3, 5.5, and in the bases for TS 2.1.1, 2.2.1, 3.1, 3.2, 3.9.a, 4.9.a, and 4.9.b. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

Seventh Response Submitted December 14, 2010

The Rhode Island Nuclear Science Center Technical Specifications contain numerous referencing errors due to discrepancies between document versions. Below is a table of the sections in which these errors can be found, their page numbers as can be found in Chapter 14 of the SAR, the current reference, and the corrected referenced section. The Corrected SAR Reference column indicates where the information should have been located. In some cases, these sections of the SAR have been revised. Any information that has been moved or omitted will be addressed in the future.

TS Section	Page	Reference	Corrected SAR Reference
2.1.1	14-12	Part B	Sections 4.6-4.8
2.2.1	14-14	Part B	Section 13.2.3
2.2.1	14-14	Part A, Section XI	Section 13.2.5
2.2.1	14-14	Part A, Section IX and Part B Section X and Appendix D	Section 13.2.3
3.1	14-17	Part A, Section V	Section 4.5
3.2	14-18	Section XI	Section 4.5
3.9a	14-30	Part A Section VIII	Section 4.2.3
4.1.1b	14-32	Part A, Section V	Section 4.5
4.2.6	14-33	Part A, Section V	Section 4.5
4.2.7	14-34	Part A, Section V	Section 4.5
4.2.8	14-34	Part A, Section V	Section 4.5
4.9a	14-39	Part A, Section VIII	Section 4.2.3
4.9b	14-40	Part A, Section VI	Section 4.5
5.3	14-41	Figure 4, Revision 1, Section V, Dec. 1992	Figure 4-1, Chapter 4
5.5	14-42	Part A, Section XII	Section 9.2.3

14.2 The "Specification" section of several proposed TS contain references to portions of the SAR. Any portion of the SAR referenced in the "Specification" section of a proposed TS will become part of the TS and license. Unless it is intended that portions of the SAR become requirements of the TS and license, revise the "Specification" sections of the proposed TS to eliminate references to the SAR.

Seventh Response Submitted December 14, 2010

This is a general comment. Specifications that contain references to portions of the SAR will be revised accordingly on a case by case basis.

14.3 The proposed TS do not appear to use the term "certified operator" defined by TS 1.1. Explain the reason for including this definition, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The reference to "certified" operator will be removed. In its place, "Operator" will be defined as: "An individual authorized by the U. S. Nuclear Regulatory Commission to carry out the responsibilities associated with the position requiring the certification".

14.4 The proposed TS do not appear to use the term "class A operator" found in the definition of TS 1.1.1. Explain the reason for including this term in the definition, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The reference to "Class A Operator" will be removed because it is not used anywhere in the document.

14.5 TS 1.1.1 does not specify that a senior reactor operator is also a reactor operator. If it is intended that a senior reactor operator can also function as a reactor operator, revise TS 1.1.1 accordingly.

First Response Submitted June 10, 2010

The definition of "Senior Reactor Operator" will be changed to "An individual licensed under 10 CFR Part 55 to manipulate the controls of the RINSC reactor and to direct the licensed activities of reactor operators".

14.6 The proposed TS do not appear to use the term "class B operator" found in the definition of TS 1.1.2. Explain the reason for including this term in the definition, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The reference to "Class B Operator" will be removed because it is not used anywhere in the document.

14.7 The wording of TS 1.1.2 is non-specific in that it defines a reactor operator as, "an individual who is licensed to operate the controls of a reactor." Explain the reason for not making this definition specific to the RINSC reactor, and revise the definition as appropriate.

First Response Submitted June 10, 2010

The definition of Reactor Operator will be changed to "An individual licensed under 10 CFR Part 55 to manipulate the controls of the RINSC reactor".

14.8 TS 1.4 contains two references which are more than 40 years old. Revise the definition to include valid and up-to-date references.

First Response Submitted June 10, 2010

Updated references will be used in the definition. The new definition will be:

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

14.9 The proposed TS contain definitions of two distinct types of channels (i.e., instrumentation channel (TS 1.5) and measured channel (TS 1.8)). The definitions of these channel types are very similar, but include different lists of components that comprise each type of channel. Explain the physical and operational characteristics that differentiate these two channel types. Explain the reason for not consolidating the definitions of the two channel types into a single definition of "channel" that is consistent with the guidance in ANSI/ANS-15.1.

First Response Submitted June 10, 2010

The definition of "Measured Channel" will be removed because it is redundant. The definition of "Channel" will be consistent with ANSI/ANS 15.1:

"1.5 Channel

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

1.5.1 Channel Test

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

1.5.2 Channel Check

Channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.5.3 Channel Calibration

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

14.10 The proposed TS do not appear to use the term "instrumentation channel" defined by TS 1.5. Revise the proposed TS to use consistent terminology.

First Response Submitted June 10, 2010

The definition of "Instrumentation Channel" will be removed because it is redundant. See RAI question 14.9.

14.11 TS 1.5 contains subsections that define "channel test," "channel check," and "channel calibration." As formatted in the proposed TS, it is unclear whether these definitions apply only to instrumentation channels, or whether they apply to other types of channels defined in the proposed TS (i.e., measured channel (TS 1.8)). If it is intended that these definitions apply to all types of channels, revise the proposed TS accordingly.

First Response Submitted June 10, 2010

See RAI questions 14.9 and 14.10.

14.12 The proposed TS do not appear to use the term "measured channel" defined by TS 1.8. The proposed TS use the terms "measuring channel" (TS 1.28) and "safety channel" (TS 1.28 and TS 3.2, Table 3.1). Revise the proposed TS to use consistent terminology.

First Response Submitted June 10, 2010

TS 1.8 "Measured Channel" definition has been removed. See RAI question 14.9.

TS 1.28 is now TS 1.24 and has been revised to say "A safety channel is a channel in the reactor safety system."

TS 3.2 Table 3.1 The term "Safety Channel" is still valid in this section of TS.

14.13 TS 1.9 uses the term "measuring channel" in the definition of measured value. This term is not defined in the proposed TS (TS 1.8 defines "measured channel"). Revise the proposed TS to use consistent terminology.

First Response Submitted June 10, 2010

The term "Measured Channel" has been deleted. See RAI question 14.9. Reference to "Measuring Channel" has also been removed.

14.14 TS 1.14 does not specify a reference core condition at which the excess reactivity is measured. Explain the reason for not specifying a reference core condition, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

TS 1.14 is a generic definition of excess reactivity, rather than a reference to a specific RINSC core value. This is due to the fact that the amount of excess reactivity within the core depends on core condition (temperature, poisons, etc…). This definition will not be changed.

14.15 TS 1.15 defines reactivity limits as "limits imposed on the reactor core excess reactivity." Contrary to this definition, TS 3.1, "Reactivity Limits," contains many limits on reactivity that are not related to the reactor core excess reactivity (e.g., TS 3.1.6). Explain this apparent inconsistency, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

This definition has been revised to encompass all of the limits placed on reactivity:

"Reactivity limits are those limits placed on the reactivity worths of reactor configurations, components, and experiments".

14.16 TS 1.15 appears redundant with TS 1.14, except that TS 1.15 specifies that the excess reactivity is referenced to a reference core condition. Consider consolidating TS 1.14 and TS 1.15.

First Response Submitted June 10, 2010

The definition in TS 1.14 provides a generic description of how "Reactivity Excess" is defined. The definition in TS 1.15 provides a general description of the types of reactivity limits that are imposed on core configurations, core components, and experiments. The reference to a "Reference Core" has been removed. See RAI question 14.15.

14.17 Clarify whether the word "equipment" used in TS 1.16 should be replaced with the word "experiment." If so, revise TS 1.16 as appropriate. If not, explain how alterations in equipment position or configuration could affect the reactivity worth of an experiment.

First Response Submitted June 10, 2010

The word "equipment" has been replaced with the word "experiment". The revised definition is:

"The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration".

14.18 The first sentence of TS 1.17 states, "the reactor is operating whenever it is not secured or shut down." The term "reactor secured" is not defined in the TS. TS 1.19 defines the term "reactor secure." Explain this apparent inconsistency, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

"Reactor Secured" is defined as:

"The reactor is secured when the following conditions are met:

a. The reactor is shutdown.
b. The master switch is in the off position and the key is removed from the lock.
c. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.
d. No experiments are being moved or serviced."

14.19 The formatting of TS 1.19 is confusing in that it contains a subsection that defines "subcritical." The formatting implies that the reactor is secure whenever it is subcritical, which is inconsistent with the guidance in ANSI/ANS-15.1. Additionally, the definition of subcritical given by TS 1.19.1 is inconsistent with the use of the term in other definitions (e.g., TS 1.20). Revise TS 1.19 to be consistent with the guidance in ANSI/ANS-15.1, or propose separate definitions for "reactor secured" and "subcritical" that are consistent with the other proposed TS.

First Response Submitted June 10, 2010

See the revised definition of "Reactor Secured" in RAI question 14.18.

The definition for "Subcritical" has been given its own TS heading. The definition has been revised to say:

"There is insufficient fissile material or moderator present in the reactor, control rods or adjacent experiments, to attain criticality under optimum available conditions of moderation and reflection".

14.20 TS 1.24 states, "a removable experiment... can reasonably be anticipated to be moved one or more times during the life of the reactor." Clarify whether the anticipated movement of a removable experiment would be intentional movement of the experiment or could be unintentional movement of the experiment. Similar to TS 1.3.2, describe the restraining forces required for removable experiments. Explain the differences between removable experiments and secured experiments. Explain the differences between removable experiments and movable experiments.

First Response Submitted June 10, 2010

"Removable Experiment" has been deleted from the TS. All RINSC experiments are categorized as either "fixed" experiments or "moveable" experiments. It is assumed that all experiments could be removed from the core, experimental facilities, or facility.

14.21 Explain why the definition of removable experiment given in TS 1.24 is separate from the definitions of other types of experiments contained in TS 1.3, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

See RAI question 14.20.

14.22 The definition of research reactor given by TS 1.26 is non-specific to the RINSC reactor. Explain the reason for including this definition, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The definition of "Research Reactor" has been deleted because it is not used.

14.23 The proposed TS do not appear to use the term "rundown" defined by TS 1.27. Explain the reason for including this definition, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The definition of "Rundown" has been deleted because it is not used.

14.24 TS 1.27 defines a "rundown" as the automatic insertion of the shim safety blades. Explain how a rundown is different from automatic insertion of the shim safety blades caused by a scram, and revise the definition accordingly. Clarify whether there are any provisions for a manually-initiated rundown.

First Response Submitted June 10, 2010

See RAI question 14.23.

14.25 TS 1.28 states that a safety channel is a "measuring channel," but the term "measuring channel is not defined in the proposed TS (TS 1.8 defines "measured channel"). Revise the proposed TS to use consistent terminology.

First Response Submitted June 10, 2010

The definition of "Measuring Channel" has been deleted because it is not used. See RAI question 14.9.

14.26 The definition of "scram time" given in TS 1.30 uses the phrase "specified control blade movement." Explain what "specified control blade movement" means, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The definition of "Scram Time" has been modified to be:

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

14.27 The definition of "shim safety blade" given in TS 1.31 uses the phrase "function of a safety blade." Explain the meaning of this phrase as it applies to the RINSC reactor and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The definition of "Shim Safety Blade" has been modified to be:

"A shim safety blade is a control blade fabricated from a neutron absorbing material which is used to compensate for fuel burn-up, temperature, and poison effects. A shim safety blade is magnetically coupled to its drive unit allowing it to fully insert into the core due to gravity when the magnet is de-energized."

14.28 The definition of "shutdown margin" given in TS 1.33 is inconsistent with the requirements of TS 3.1.1 in that the definition does not specify the position of the regulating blade. Also, TS 1.33 uses the phrase "most reactive position," while TS 3.1.1 uses the phrase "fully withdrawn" to describe the positions of control blades. Explain these apparent inconsistencies in the TS, and revise the proposed TS as appropriate. (See RAI 14.56)

First Response Submitted June 10, 2010

The definition of "Shutdown Margin" will be changed to:

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

14.29 The proposed TS do not appear to use the term "static reactivity worth" defined by TS 1.35. Explain the reason for including this definition in the proposed TS, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

This definition has been deleted because it is not used.

14.30 TS 1.37 appears to be a description of allowed deferral of surveillance activities and not a definition of surveillance activities. ANSI/ANS-15.1 recommends that allowed deferral of surveillance activities be included in each TS requiring a surveillance activity. Explain why TS 1.37 is included in the definitions section of the proposed TS, and revise the proposed TS as appropriate. (See RAI 14.130)

First Response Submitted June 10, 2010

The definition of "Surveillance Activities" has been changed to:

"Regularly scheduled activities that verify the integrity and operability of facility infrastructure and equipment, and that ensure the safe operation of the reactor".

14.31 TS 1.38 states that maximum surveillance intervals "are to provide operational flexibility and to reduce frequency." The guidance in ANSI/ANS-15.1 states that maximum surveillance intervals "are to provide operational flexibility only and are not to be used to reduce frequency." Explain this apparent inconsistency, and revise the proposed TS as appropriate.

First Response Submitted June 10, 2010

The definition has been changed to:

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

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

14.32 The "Applicability" section of TS 2.1.1 states that the specification applies to steady state operation. Explain the reason that the safety limits (SLs) apply only to steady-state operation. If the SLs also apply to reactor transients, revise TS 2.1.1 as appropriate. If the SLs do not apply to transients, proposed SLs in accordance with 10 CFR 50.36(c)(1)(i)(A) that apply to all reactor operations allowed by the proposed TS and all credible accidents. (See RAI 4.24)

Second Response Submitted August 6, 2010

According to 10 CFR 50.36(c),(1) Safety limits, limiting safety system settings, and limiting control settings.

> (i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down.

The SLs of Section 2.1 and the associated LSSSs of Section 2.2 are established at conservative levels that effectively preclude any damage to the fuel cladding, the primary barrier to release of radioactivity, under normal and credible abnormal conditions. As supported by the thermal-hydraulic analysis in the SAR, these settings in conjunction with the Section 3.1 LCO on maximum excess reactivity also preclude damage to the fuel cladding under credible reactivity transients.

There is no CFR requirement to specifically address transients, but the SL must address all operations, not just steady state operation. TS 2.1.1 will be revised as follows.

2.1.1 Safety Limits in the Forced Convection Mode

Applicability:

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

Reactor Thermal Power, P
Reactor Coolant Flow through the Core, m
Reactor Coolant Outlet Temperature, T_O
Height of Water above the Top of the Core, H

Objective:

To assure that the integrity of the fuel clad is maintained.

Specifications:

1. The true value of reactor power (P) shall not exceed 2.4 MW.
2. The true value of reactor coolant flow (m) shall not be less than 1580 gpm.
3. The true value of the reactor coolant outlet temperature (T_O) shall not exceed 125 ^{O}F.
4. The true value of water height above the active core (H) shall not be less than 23 ft 6.5 in. while the reactor is operating at any power level.

Bases:

The basis for forced convection safety limits is to ensure that the calculated maximum cladding temperature in the hot channel of the core will not be exceeded. Thermal hydraulic analyses show that if the safety limits are not exceeded the integrity of the fuel cladding will be maintained.

14.33 The bases for TS 2.1.1 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.

Fourth Response Submitted September 8, 2010

See the revised TS 2.1.1 in RAI 14.32.

14.34 Item 2 of the "Objective" section of TS 2.1.2 states, "To assure consistency with other defined safety system parameters." Explain the meaning of this statement, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

The objective of the natural convection mode safety limits is to assure that the integrity of the fuel cladding is maintained. The second item listed as an objective (P.14-13 Line14) will be removed.

14.35 TS 2.1.2.1 specifies a SL of 217 kW for the true value of the reactor power during operation in the natural convection mode. The SL is based on preventing nucleate boiling in the hot channel. Table 4-17 of the SAR shows a negative margin to incipient boiling at 209.1 kW for the hot channel, which implies that incipient boiling occurs at a power level less than 209.1 kW. Explain this apparent inconsistency between the SL and Table 4-17, and revise the proposed TS as appropriate. (See RAI 4.29)

Fifth Response Submitted November 26, 2010

See the response to RAI 4.29.

14.36 The "Applicability" section of TS 2.2.1 reads, "LEU Fuel Temperature – Forced Convection Mode." However, the "Specification" section of TS 2.2.1 gives limits for reactor thermal power, primary coolant flow through the core, height of water above the top of the core, and reactor coolant outlet temperature, and not fuel temperature. Explain this apparent inconsistency between the "Applicability" and "Specification" sections of TS 2.2.1, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

All references to fuel temperature have been removed. Section 2.2.1 is revised as follows.

2.2.1 Limiting Safety System Settings in the Forced Convection Mode

Applicability:

These LSSSs apply to the setpoints for the safety channels monitoring reactor power, primary coolant flow, pool level and core outlet temperature.

Objective:

To assure that the integrity of the fuel cladding is maintained in the forced convection mode.

Specifications:

The limiting safety system settings for reactor thermal power (P), primary coolant flow through the core (m), height of water above the top of the core (H), and reactor coolant outlet temperature (T_O) shall be as follows:

Measured Parameter	LSSS
P	2.1 MW
m	1800 gpm
H	23 ft 9.6 in
T_O	120 ^OF

Bases:

These specifications were set to prevent coolant temperatures from approaching the value at which damage to fuel cladding could occur (see NUREG-1313, "Safety Evaluation Report related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Nonpower Reactors"). Flow and temperature limits were chosen to ensure that the integrity of the cladding is maintained even under transient conditions. The uncertainty in the flow measurement is ± 3%. The uncertainty in the temperature measurement is ± 2%. The uncertainty in the measured power level is ± 10% (see RAI 4.20 response). The uncertainty in the measurement of the pool height is estimated to be 0.5 in. At the limits of the uncertainty bands, there are still margins of 0.1 MW, 160 gpm, 2 ^OF and 2.6 in. to the SL values for power, flow, temperature and pool height, respectively. The following table summarizes the bases for the LSSS settings.

Measured Parameter	LSSS Value	Measurement Uncertainty	Limiting Trip Value	Safety Limit	Safety Margin
P	2.1 MW	± 10% (± 0.2 MW*)	2.3 MW	2.4 MW	0.1 MW
m	1800 gpm	± 3% (± 60 gpm*)	1740 gpm	1580 gpm	160 gpm
H	23 ft 9.6 in.	± 0.5 in.	23 ft 9.1 in.	23 ft 6.5 in.	2.6 in.
T_O	120 ^OF	± 2% (3 ^OF*)	123 ^OF	125 ^OF	2^OF
*Uncertainties in measured values (± 0.2 MW, ± 60 gpm, 2 ^OF) are based on the nominal operating values of 2 MW, 1950 gpm, and 90 ^OF to 115 ^OF for the power, flow and outlet temperature, respectively.					

14.37 The "Objective" section of TS 2.2.1 appears to be both an applicability statement and an objective statement. Explain why the applicability statement is in the "Objective" section of TS 2.2.1, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.38 The "Objective" section of TS 2.2.1 contains the statement, "to assure that the maximum fuel temperature permitted is such that no damage to the fuel cladding will result in the forced convection mode." This statement appears to be inconsistent with the requirement of 10 CFR 50.36(c)(1)(ii)(A) that, "where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Additionally, TS 1.7 states that limiting safety system settings (LSSS) will be "chosen so that automatic protective action will correct an abnormal situation before a safety limit is exceeded," which appears to be inconsistent with the objective to limit fuel temperature. Explain these apparent inconsistencies, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.39 TS 2.2.1 gives LSSS for reactor thermal power, primary coolant flow through the core, height of water above the top of the core, and reactor coolant outlet temperature. TS 2.1.1 establishes SLs for these variables. 10 CFR 50.36(c)(1)(ii)(A) requires that, "where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Explain how the LSSS satisfy the requirement of 10 CFR 50.36. Include analyses, with fully justified assumptions, that show the LSSS prevent exceeding a SL for all operations allowed by the proposed TS and all credible accidents. Per 10 CFR 50.36(a)(1), these analyses shall be summarized and/or referenced in the bases for the LSSS. (See RAI 14.32)

Eighth Response Submitted January 24, 2011

See rewritten TS 2.2.1 in response to 14.36. A summary of the new transient analyses is included in the Bases.

14.40 The bases for TS 2.2.1 reference fuel temperature and fuel cladding temperature as though these parameters were the parameters for which the SLs were established. TS 2.1.1 does not establish SLs on fuel temperature or fuel cladding temperature. TS 2.1.1 establishes SLs on reactor thermal power, reactor coolant flow through the core, reactor coolant outlet temperature, and height of water above the top of the core. Explain how the bases support each LSSS, and revise the proposed TS as appropriate. (See RAI 14.39)

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.41 The bases for TS 2.2.1 make multiple references to fuel temperature and fuel cladding temperature limits. If the intention is to have these limits be SLs for the RINSC reactor, revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.42 The bases for TS 2.2.1 state, "flow and temperature limits were chosen to prevent incipient boiling even if transient power rises to the 2 MW trip limit of 2.4 MW." However, the LSSS for reactor power specified by TS 2.2.1 is 2.3 MW. Explain this apparent inconsistency between the bases and the specification.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.43 The bases for TS 2.2.1 state, "flow and temperature limits were chosen to prevent incipient boiling even if transient power rises to the 2 MW trip limit of 2.4 MW." However, Section 4.6.4 of the SAR states that during a rising power transient, the calculated fuel surface temperature would be above the onset of nucleate boiling temperature. Explain this inconsistency between the bases of TS 2.2.1 and the analysis in the SAR.

Eighth Response Submitted January 24, 2011

See rewritten TS 2.2.1 in response to 14.36. The revised thermal hydraulic analysis for the SAR is consistent with the revised TS.

14.44 The bases for TS 2.2.1 include uncertainties associated with some of the LSSS parameters, but exclude reactor power and coolant height. Discuss the uncertainties associated with these parameters and explain how the uncertainties were incorporated into the analyses supporting the LSSS. (See RAI 4.20)

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36. Supporting analyses use the limiting values provided in the table included in the Bases statement.

14.45 The bases for TS 2.2.1 state, "the LSSS for the pool level is set for a scram upon a 2 inch drop in water level." TS 2.2.1 specifies a LSSS of 23.7 feet, which is a true value, and not a magnitude of decrease in pool level. Explain this apparent inconsistency, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.46 The bases for TS 2.2.1 state, "the safety limit settings chosen provide acceptable safety margins to the maximum fuel cladding temperature." Explain the meaning of the phrase "safety limit settings." Provide quantitative values for the safety margins referred to as "acceptable safety margins," and explain the reasons they are considered acceptable.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.47 The bases for TS 2.2.1 state, "the LSSS for the pool level results in a higher number since the pool level scrams upon a 2 inch drop in water level." Explain what "higher number" means in this context.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36.

14.48 The bases for TS 2.2.1 contain the reference, "Report on the Determination of Hot Spot Factors for the RINSC Research Reactor, August 1989." Provide a copy of this reference.

See rewritten TS 2.2.1 in response to 14.36. Reference is no longer used for determining the flow and temperature measurement uncertainties, and is not included. See RAI question 4.20 for flow and temperature measurement uncertainty analysis.

14.49 The bases for TS 2.2.1 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.

Second Response Submitted August 6, 2010

See rewritten TS 2.2.1 in response to 14.36. References to a previous SAR have been removed.

14.50 The bases for TS 2.2.2 state, "the SAR has determined that up to 217 kW can be removed by natural convection." However, Table 4-17 of the SAR shows a negative margin to incipient boiling at 209.1 kW, which implies that incipient boiling occurs at a power level less than 209.1 kW. Explain this apparent inconsistency between the bases and Table 4-17 of the SAR. (See RAI 14.35)

Fifth Response Submitted November 26, 2010

As indicated in the response to RAI 14.52, the bases for TS 2.2.2 has been revised and no longer states, "the SAR has determined that up to 217 kW can be removed by natural convection." Moreover, the analysis of the thermal behavior of the LEU core during steady-state operation in the natural-convection mode, which was in section 4.6.5 of the 2004 RINSC Reactor SAR and included Table 4-17, has been completely redone and is replaced by section 4.7 of Reference AA.

As stated in the response to RAI 4.28, Reference AA refers to the completely redone analysis of the thermal behavior of the LEU core in the *forced*-convection mode, which is provided in Reference BB.

In the analysis of Reference AA, onset of nucleate boiling is predicted to occur at 369 kW with all uncertainties included. Thus, there is no inconsistency between the bases for TS 2.2.2 and the power at which onset of nucleate boiling is predicted to occur in the analysis of section 4.7 of Reference AA.

AA. Argonne National Laboratory intra-laboratory memo, Earl E. Feldman and M. Kalimullah to James E. Matos, "Steady State Thermal-Hydraulic Analysis for Natural-Convective Flow in the Rhode Island Nuclear Science Center (RINSC) Reactor," November 8, 2010.

BB. Argonne National Laboratory intra-laboratory memo, Earl E. Feldman to James E. Matos, "Steady State Thermal-Hydraulic Analysis for Forced-Convective Flow in the Rhode Island Nuclear Science Center (RINSC) Reactor," September 3, 2010.

14.51 The bases for TS 2.2.2 state, "with a 15% overpower trip, 115 kW will be the LSSS." This seems to be an arbitrary value with no supporting analysis or justification. Provide an analysis, with fully justified assumptions, that demonstrates the LSSS on reactor power will prevent a SL from being exceeded for all operations allowed by the proposed TS and all credible accidents.

Eighth Response Submitted January 24, 2011

See rewritten TS 2.2.2 in response to 14.52. A summary of the revised thermal/transient analyses is included in the response to this RAI.

14.52 The bases for TS 2.2.2 state, "the pool level scram (2 inch drop) is the same as the forced convection mode." TS 2.2.2 specifies a LSSS of 23.7 feet, which is a true value, and not a magnitude of decrease in pool level. Explain this apparent inconsistency, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

The pool level scram will be in reference to pool level elevation. In the new safety analysis, the pool level safety limit has been set at 23 ft 6.5 inches above the top of the reactor core. The LSSS has been set at 23 ft 9.6 inches above the top of the core. The safety margin that is provided by this is shown in the bases section.

Revise Section 2.2.2 as follows.

2.2.2 Limiting Safety System Settings in the Natural Convection Mode

Applicability:

These LSSSs apply to the setpoints for the safety channels monitoring reactor power, pool level and pool water temperature.

Objective:

To assure that the integrity of the fuel cladding is maintained in the natural convection mode.

Specifications:

The limiting safety system settings for reactor thermal power (P), height of water above the top of the core (H), and reactor pool water temperature (T_p) shall be as follows:

Measured Parameter	LSSS
P	115 kW
H	23 ft 9.6 in.
T_p	125 ^{o}F

Bases:

These specifications were set to prevent coolant temperatures from approaching the value at which damage to fuel cladding could occur (see NUREG-1313, "Safety Evaluation Report related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Nonpower Reactors"). Power and temperature limits were chosen to ensure that the integrity of the cladding is maintained even under transient conditions. The uncertainty in the pool temperature measurements is ± 2%. The uncertainty in the measured power level under natural convection conditions is ±10% (see RAI 4.20 response). The uncertainty in the measurement of the pool height is estimated to be 0.5 in. At the limits of the uncertainty bands, there are still margins of 75 kW, 2 ^{o}F and 2.6 in. to the SL values for power, pool temperature and pool height, respectively. The following table summarizes the bases for the LSSS settings.

Measured Parameter	LSSS Value	Measurement Uncertainty	Limiting Trip Value	Safety Limit	Safety Margin
P	115 kW	± 10% (± 10 kW*)	125 kW	200 kW	75 kW
H	23 ft 9.6 in.	± 0.5 in.	23 ft 9.1 in.	23 ft 6.5 in.	2.6 in.
Tp	125 ^{o}F	± 2% (3 ^{o}F*)	128 ^{o}F	130 ^{o}F	2 ^{o}F
*Uncertainties in measured values (± 10 kW, 3 ^{o}F) are based on the nominal operating values of 100 kW and 108 ^{o}F for the power and pool temperature, respectively.					

14.53 The bases for TS 2.2.2 state, "the pool temperature 130 $^{\circ}$F safety limit, having a 3% error, results in a LSSS of 126 $^{\circ}$F." Explain the basis for the 3 percent error. Provide an analysis, with fully justified assumptions, that demonstrates the LSSS on pool temperature will prevent a SL from being exceeded for all operations allowed by the proposed TS and all credible accidents.

Fourth Response Submitted September 8, 2010

See the response to RAI Question 4.20 for the power and coolant height measurement uncertainty estimate.

See rewritten TS 2.2.2 in response to 14.52 for how they were treated in the analysis supporting the LSSS.

14.54 The bases for TS 2.2.2 do not discuss uncertainties associated with reactor power and coolant height. Explain the uncertainties associated with these variables and explain how the uncertainties were treated in the analyses supporting the LSSS.

Fourth Response Submitted September 8, 2010

See the response to RAI Question 4.20 for the power and coolant height measurement uncertainty estimate.

See rewritten TS 2.2.2 in response to 14.52 for how they were treated in the analysis supporting the LSSS.

14.55 ANSI/ANS-15.1 recommends technical specifications establish limits on fuel burnup. Explain the reason for not including such a specification, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

The type of fuel used at RINSC has been qualified to 98% burn-up. Consequently, no limit on fuel burn-up is necessary. The reference for this is NUREG 1313.

14.56 TS 3.1.1 requires the shutdown margin to be determined with the most reactive shim safety blade and the regulating blade fully withdrawn. The bases for TS 3.1.1 do not mention the position of the regulating blade. Explain this apparent inconsistency, and revise the proposed TS as appropriate. (See RAI 14.28)

Second Response Submitted August 6, 2010

The definition of "Shutdown Margin" will be changed to:

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

The basis for TS 3.1.1 (P.14-17 Line 6) will be changed to:

Specification 3.1.1 assures that the reactor can be shutdown from any operating condition and will remain subcritical after cool down and xenon decay even if the blade of the highest reactivity worth and the regulating blade are in the fully withdrawn position.

14.57 The bases for TS 3.1.1 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.

Second Response Submitted August 6, 2010

This reference has to do with predictions that were made about what the shutdown margin would be, prior to when the LEU core was configured, and the shutdown margin was measured. Since this core has been in operation for more than fifteen years, and the shutdown margin for it has been measured at least annually, this reference is no longer relevant. Consequently it will be removed.

14.58 The bases for TS 3.1.3 state that the limit on the reactivity worth of experiments prevents melting of the fuel. However, the SLs specified in TS 2.1 do not include fuel temperature. Explain how the LCO for the reactivity worth of experiments is consistent with the SLs, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

In the Bases section for LCO 3.1.3, rewrite the paragraph for Specification 3.1.3 as follows.

Specification 3.1.3 limits the reactivity worth of experiments to values of reactivity which, if introduced as positive step changes, would preclude violating any Safety Limit. Transient analysis demonstrates that this LCO on reactivity for experiments results in no challenge to fuel integrity under credible postulated transients.

14.59 TS 3.1.4 does not include explicit reactivity limits for removable experiments. Explain which reactivity limit (movable or secured) applies to removable experiments or revise TS 3.1.4 to include an explicit reactivity limit for removable experiments. (See RAI 14.20)

Second Response Submitted August 6, 2010

The reference to "Removable" experiments has been deleted. See the answer to RAI question 14.20.

14.60 TS 3.1.4 limits the reactivity worth of each movable experiment to 0.08 %Δk/k. Section 13.2.2 of the SAR appears to state that the total reactivity worth of all movable experiments is limited to 0.08 %Δk/k. Explain whether each movable experiment is limited to 0.08 %Ak/k, or whether the total reactivity worth of all movable experiments is limited to 0.08 %Ak/k. If the reactivity worth of each movable experiment is limited to 0.08 %Ak/k, explain whether multiple movable experiments could comprise the total experiment reactivity worth limit of 0.6 %Ak/k (e.g., ten movable experiments each with a reactivity worth of 0.06 %Ak/k).

Second Response Submitted August 6, 2010

TS 3.1.3 limits the total reactivity worth of all experiments in the core to 0.6% dK/K.

TS 3.1.4 limits the reactivity worth of any individual moveable experiment to be 0.08% dK/K, and any fixed experiment to be 0.6% dK/K.

An additional limit will be added to clarify that the maximum total reactivity worth of all moveable experiments in the core is 0.08%.

Rewrite these Technical Specifications as follows:

3.1.3 The total reactivity worth of experiments shall not exceed:

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:

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

14.61 The bases for TS 3.1.4 state that the individual reactivity worth of an experiment is limited to a value that will not produce a stable reactor period of less than 30 seconds. Explain whether this statement applies to all types of experiments. Provide an analysis that supports this statement, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

It is assumed that fixed experiments will not produce a reactor period because they are fixed. The total reactivity of fixed experiments is limited to 0.6 % dK/K in order to assure that if a failure occurred in which the experiment reactivity was inserted into the reactor, there would be insufficient reactivity to produce a prompt critical condition. The prompt critical condition occurs when $\rho = \beta$. For the RINSC U-235 fuelled core, $\beta = 0.0065$. Consequently, if the reactivity insertion is $\rho = 0.6$ % dK/K, it will be less than the 0.65 % dK/K necessary to cause prompt criticality. The reference for this may be found in Glasstone, Samual; Sesonske, Alexander (1980), *Nuclear Reactor Engineering* (3rd ed.), Van Nostrand Reinhold.

The reactivity of moveable experiments is limited to 0.08 %dK/K in order to assure that they will not produce a stable period less than 30 seconds, and to assure that the reactivity can be compensated for by the action of the control and safety systems without exceeding any safety limits. The following diagram from Lamarsh; John R., Baratta, Anthony J., (2001), *Introduction to Nuclear Reactor Engineering* (3rd ed.), Prentice Hall, shows that for a U-235 fueled core, with a positive reactivity insertion of 0.08 %dK/K = 0.0008 = 8 E 10^{-4}, the period would be approximately 80 seconds, which is easily compensated for by the action of the control and safety systems.

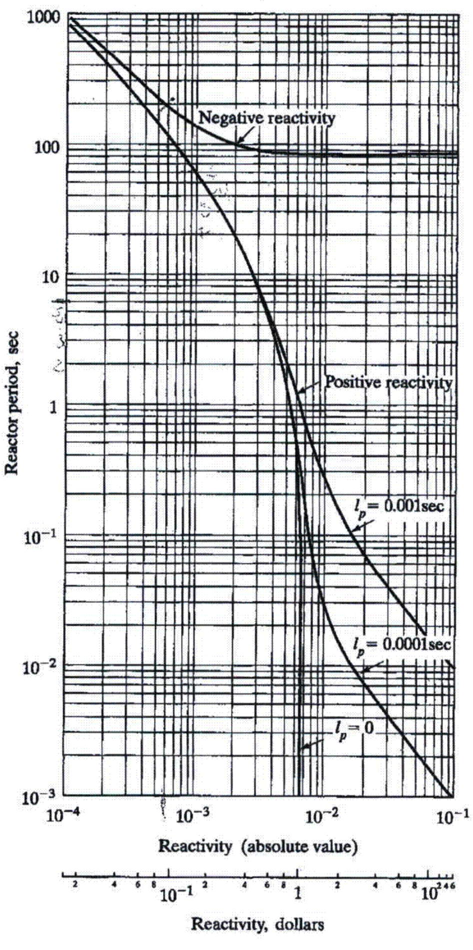


Figure 7.2 Reactor period as a function of positive and negative reactivity for a ^{235}U-fueled reactor.

This was confirmed by an analysis performed by Argonne National Laboratory. In that analysis, the initial assumptions were that:

1. Reactor Power was at 10 Watts
2. Coolant Inlet Temperature was 123 F
3. Coolant Flow Rate was 1740 gpm
4. Height of the Coolant above the Fuel Meat was 23 ft 9.1 in
5. Water Pressure at the Top of the Fuel Meat was 1.715 X 10^5 Pa

A reactivity insertion of +0.08% dK/K was added over a time span of 0.1 seconds. The following sequence of events were predicted to occur:

1. At t = 0.0 seconds the reactivity insertion begins, and reactor power begins to increase.
2. At t = 0.1 seconds the reactivity insertion ends, and reactor power continues to increase.
3. At t = 30 seconds a stable period of about 75 seconds is reached.
4. At t = 1166.6 seconds power reaches 2.3 MW, and the over power trip is actuated. At this point, feedback reactivity from Doppler, water expansion, and voids cause the period to decrease to approximately 1375 seconds, effectively making power constant at 2.3 MW.
5. The model assumed that there would be a 100 msec delay between the time that the trip was actuated, and the time that negative reactivity from the control system would begin to be inserted. Consequently, at t = 1166.7 seconds, the reactor power is still approximately 2.3 MW, but negative reactivity begins to be inserted.
6. Reactor power drops rapidly to shutdown conditions.

At t = 1167 seconds, peak temperatures are estimated to be:

1. Fuel Meat Centerline is 81.4 C
2. Clad Surface is 80.6 C
3. Coolant is 64.9 C

These temperatures are well below the maximum fuel cladding temperature limit of 530 C suggested in NUREG 1313.

14.62 The bases for TS 3.1.4 state that the control and safety systems will protect the safety limits in the case that the reactivity associated with an experiment is inserted into the reactor. Section 13.2.2 of the SAR presents an analysis of an insertion of reactivity, but does not explicitly demonstrate that the LCO is chosen such that the LSSS will prevent the SLs from being exceeded. Provide analyses, including fully justified assumptions, that show the LCO is appropriately chosen so that the LSSS will prevent exceeding the SLs. (See RAI 13.9)

Fifth Response Submitted November 26, 2010

See the analysis of a rapid insertion of 0.6 %Δk/k from very low power in RAI 13.7. A rapid insertion of 0.08 %Δk/k from very low power for the moveable experiments is bounded by the 0.6 % Δk/k insertion case.

14.63 TS 3.1.5 requires the reactor to be subcritical by at least 3.0 %Ak/k during fuel loading changes. Explain how it is determined that the reactor is subcritical by at least 3.0 %Ak/k during fuel loading changes. Explain the reason for not specifying a surveillance requirement for this LCO, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

RINSC is currently operating with its equilibrium core. The minimum shutdown reactivity for this core occurs just after re-fueling operations, in which four irradiated fuel elements are replaced with four fresh fuel elements. This operation was performed in October 2008. The data for the new core configuration with the fresh fuel indicated that the shutdown reactivity was –7.07% dK/K (See the reference entitled "Core Change Summary from RINSC Core LEU #3 to LEU #4"). As operation of the reactor continues, the shutdown reactivity will become more subcritical as fuel burn-up occurs.

TS 3.1.3 limits the total worth of all experiments to 0.6% dK/K. Therefore, if re-fuelling has just occurred, and an experiment worth +0.6% dK/K has been added, the shutdown reactivity would be approximately:

-7% dK/K + 0.6% dK/K = -6.4% dK/K

Consequently, it is not anticipated that the reactor will ever be subcritical by less than 3% dK/K during fuel loading operations.

Add the following surveillance item:

4.1.1.4 Prior to fuel loading changes, core reactivity shall be verified to be shutdown by a minimum of 3 %dK/K by using existing core data, or by making new core reactivity measurements.

14.64 TS 3.1.6 limits the reactivity worth of the regulating blade. The proposed TS do not appear to specify surveillance requirements for the reactivity worth of the regulating blade. Explain the reason for not specifying a surveillance requirement for the reactivity worth of the regulating blade, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

TS 4.1.1 will be modified to say (P14-32 Line 26):

Shim safety blade and regulating rod reactivities and insertion rates will be measured:

a. Annually
b. Whenever the core configuration is changed to an uncharacterized core

The reference to a previous SAR will be removed. This reference has to do with predictions that were made about core characteristics prior to when the LEU core was configured and tested.

14.65 TS 3.1.7 states, "Experiments which could increase reactivity by flooding, shall not remain in or adjacent to the core unless the shutdown margin required in Specification 3.1.1 would be satisfied after flooding." Explain why experiments that could reduce the shutdown margin below 1.0 %Ak/k by flooding would ever be allowed in or adjacent to the core, and revise the proposed TS as appropriate. (See RAI 4.14)

Seventh Response Submitted December 14, 2010

Technical Specification 1.16 takes into consideration credible malfunction in the definition of the reactivity worth of experiments. See the answer to RAI question 14.17. Technical Specification 3.1.7 makes clear that flooding is a credible malfunction.

As discussed in the answer to RAI question 14.137, 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 3.1.7 requires that the experiment be removed immediately.

14.66 TS 3.1.8 states "surveillance will be conducted at initial startup and change in fuel type." Explain the reason that this surveillance requirement is included in the LCO, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

This LCO has to do with the fact that the temperature coefficient must be negative. The statement "surveillance will be conducted at initial startup and change in fuel type" was meant to indicate that the temperature coefficient would be verified to be negative at initial start-up, and if there was a change in fuel type.

This was verified during the initial startup with the LEU fuel. Any change in fuel type would require a change in the license. Consequently, this surveillance is no longer necessary. As a result, it will be removed.

14.67 TS 3.1.9 specifies core configuration requirements for operation in the forced convection mode. Explain why the TS do not contain any similar core configuration requirements for operation in the natural convection mode. Explain why the proposed TS do not restrict core configurations to the three core configurations referenced in TS 4.1.b. (See RAI 14.134)

Fourth Response Submitted September 8, 2010

Technical Specification 3.1.9 requires that all of the core grid positions be filled with fuel elements, experiments or experiment baskets, or reflector elements during operation in forced convection mode. Under conditions of forced coolant flow, the cooling water will obviously follow the path of least resistance. If any grid position is open, some of the cooling water that would normally be forced between the fuel plates will instead go through the open grid position, reducing expected cooling to the fuel. In the natural convection mode, there is no driving force to preferentially redirect coolant flow through the open grid position. Coolant circulation depends on the temperature differences induced by heat transfer from the fuel to the adjacent water. Since there would be no fuel in the open grid position, there would be no heat transfer directly to the water in that position. In this mode, the open grid position actually provides a larger sink for heat generated in elements adjacent to the open channel.

Technical Specification 4.1.1.b requires that the shim safety blade reactivity worths and insertion rates are measured whenever the core is changed from the start-up core to "the three other cores as analyzed and specified in SAR Part A, Section V". In that analysis, the RINSC LEU core was initially configured with a start-up core that was reflected with graphite next to the fuel, surrounded by beryllium reflector elements:

LEU Core #1 Startup Core Configuration

[redacted]

As burn-up occurred, the core configuration was altered to a more efficient neutron reflection configuration in which some of the beryllium elements were moved next to the fuel, and the corresponding graphite elements were to the outer edge of the element grid. This was core LEU 2:

LEU Core #2 Configuration

[redacted]

As further burn-up occurred, the configuration was altered to the most efficient neutron reflection configuration in which all of the beryllium elements were moved to positions next to the fuel. This was core LEU 3:

LEU Core #3 Configuration - 2 May 03

[redacted]

From this point forward, burn-up is offset by performing a fuel element change in which four elements from the center of the core are removed, the remaining fuel is shuffled inward in such a way that only the four corners of the fuelled part of the core are vacant, and new fuel is place in the corners. This core is the equilibrium core:

LEU Core #4 Configuration - 24 October 2008

[redacted]

Thus, the "three other cores" that were analyzed other than the start-up core are LEU 2, LEU 3, and the equilibrium cores. The reactor is currently operating with the equilibrium core.

TS 4.1.1.b will be rewritten to indicate that blade worths shall be measured when any new core is installed in the reactor.

14.68 TS 3.2.1 specifies reactor safety systems and safety-related instrumentation that are required for critical reactor operation. However, the proposed TS do not contain any requirements for reactor safety systems and safety-related instrumentation that must be operable when the reactor is subcritical, but not secured. Explain why the proposed TS do not require any operable safety systems or safety-related instrumentation when the reactor is subcritical, but not secured (e.g., movement of fuel in the reactor core). Explain why the radiation monitors listed in Table 3.2 are not required during work of the types specified in TS 1.19.1.c and TS 1.19.1.d. Revise the proposed TS as appropriate.

Eighth Response Submitted January 24, 2011

Technical Specification 1.20 defines the reactor as "shutdown" when it is subcritical by at least the shutdown margin with the reactivity of all installed experiments included.

The term "Reactor Secured" was defined as part of the answer to RAI question 14.18 to be:

> "The reactor is secured when the following conditions are met:
>
> a. The reactor is shutdown.
> b. The master switch is in the off position and the key is removed from the lock.
> c. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods.
> d. No experiments are being moved or serviced."

There is no Technical Specification requirement for safety systems and safety related instrumentation that must be operable when the reactor is subcritical but not secured because it is impossible to do a pre-start checkout to verify the operability of the safety related instrumentation without the reactor being in a non-secured state. Condition b cannot be met because the master switch cannot be in the off position with the key removed in order to perform the pre-start checkout.

Radiation monitors are required for work of the types specified in TS 1.19.1.c and TS 1.19.1.d. Technical Specification 1.17 defines the reactor to be in operation whenever it is not secured or shutdown. The answer to RAI question 7.4 provides the list of the radiation monitoring instrumentation that is required to be in operation whenever the reactor is in operation (e.g., movement of fuel in the reactor core). It is possible to verify that these instruments are operable prior to taking the reactor into an unsecured state.

14.69 TS 3.2.1, Table 3.1 contains a column labeled "Function" that appears to contain both the function of each safety channel and the set point. As written, it is difficult to understand if the set points are maximum or minimum set points. For example, the "Function" column states "automatic scram at T 1600gpm" for the coolant flow rate safety channel. This implies the scram set point can be any value less than or equal to 1600 gpm. However, the LSSS for coolant flow rate is 1600 gpm, which means that any set point less than 1600 gpm would be inconsistent with the LSSS. Other examples are reactor power level, coolant outlet temperature, log N period, and pool temperature. Revise Table 3.1 to clearly state the maximum and minimum set points for the safety channels, and ensure the set points are consistent with the LSSS.

Fourth Response Submitted September 8, 2010

The table is being revised to make the set points more clear. See the revised table in RAI Question 7.1.

14.70 TS 3.2.1, Table 3.1 states that the function of the reactor power level safety channel is "automatic scram when `115% of range scale with 2.3 MW max," and this is required in both forced and natural convection operating modes. Explain how a maximum reactor power trip setting of 2.3 MW in the natural convection mode of operation is consistent with the LSSS of 115 kW specified by TS 2.2.2, and revise the proposed TS as appropriate. What is the range scale of the reactor power level safety channels? Can the scram functions be disabled by increasing the range scale? Are there scram set points at 115 kW and 2.3 MW that are independent of the channel range scale?

Fourth Response Submitted September 8, 2010

This was intended to communicate that there are a minimum of two over power trips that have scram set points that trip when power is at a maximum of 115% on any range. Consequently, if the power range on one of these instruments is set at 2 W, and power goes above 2.3 W (115% of 2 W), a scram will occur. The "2.3 MW max" was intended to communicate that the maximum available range for these instruments is the 2 MW scale. For natural convection mode cooling, coolant flow, and inlet and outlet temperature alarms and scrams are bypassed. When these are bypassed, the bypass switch sets the over power scram to 115% of 100 kW. Table 3.1 has been revised to say that the over power scram in both cooling modes will trip by 115% of licensed power, which historically has been limited to 100 kW for natural convection mode cooling, and 2 MW for forced convection mode cooling. See the answer to RAI question 7.1.

14.71 TS 3.2.1, Table 3.1 requires a bridge misalignment safety channel and a bridge movement safety channel. The "Function" column of Table 3.1 does not contain set points for these channels. Explain the reason that Table 3.1 does not specify set points for these channels, and revise the proposed TS as appropriate. (See RAI 7.3)

Fourth Response Submitted September 8, 2010

The bridge misalignment and bridge movement safety channels each utilize separate limit switches. The bridge movement switch is located adjacent to one of the gears used to move the bridge, with the lever arm of the switch on top of a gear tooth when correctly positioned, depressing the switch. As the bridge moves away from the high powered section the roller-wheel of the arm falls into the valley of a gear, releasing the switch.

The bridge misalignment switch is attached to the end of the track at the high power section. When the bridge is moved away from the high power section the switch is released and the bridge misalignment scram is triggered. This channel is not necessary or functional during natural convection cooled operation.

Since these are both limit switches which are simply used as state / change state indicators, no set points have ever been established.

14.72 TS 3.2.1, Table 3.1 requires a pool water level safety channel with a set point at 16 inches below the suspension frame base plate elevation. TS 2.2.1 gives the LSSS for pool water level as 23.7 feet. Explain why Table 3.1 and the LSSS use different frames of reference and different units for the pool water level safety channel set point. Explain how the LCO is consistent with the LSSS, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

The pool level safety limit and limiting safety system setting are now defined in terms of height above the top of the core. In the new safety analysis, the pool level safety limit has been set at 23 ft 6.5 inches above the top of the reactor core. The LSSS has been set at 23 ft 9.6 inches above the top of the core. Table 3.1 has been modified. See the answer to RAI question 7.1. TS 2.2.1 was modified to remove this inconsistency as part of the answer to RAI question 14.36.

14.73 TS 3.2.1, Table 3.1 requires three detector high voltage failure safety channels. The "Function" column of Table 3.1 states, "automatic scram if Voltage decreases 50V max." Explain what "Voltage decreases 50V max" means.

Fourth Response Submitted September 8, 2010

This was intended to communicate that there are three channels that use high voltage detectors, and that if the high voltage decreased on any of these channels by more than 50V, a scram would occur. Table 3.1 has been revised to make this more clear. See RAI answer 7.1.

14.74 TS 3.2.1, Table 3.1 requires a no flow thermal column safety channel when the reactor is operated above 100 kW in the forced convection mode. The table does not specify a set point for the safety channel and the SAR does not specify what flow rate is necessary to remove the heat generated in the graphite in the thermal column. Explain why there is no set point for the safety channel. (See RAI 10.3)

Second Response Submitted August 6, 2010

The no-flow Reactor Safety System Component/Channel entry in Table 3.1 is labeled incorrectly. The safety channel does not apply to the heat generated in the graphite, but to the heat generated in the gamma shield at the front of the thermal column. The flow refers to the gamma shield water coolant which is taken off the primary coolant circuit. The piping and instrumentation diagram (PID) on page 21 of the 1962 Safeguards Report [B. J. Tharpe, Safeguards Report for Rhode Island Open Pool Reactor, General Electric Document APED-3872, April 4, 1962] and shows the interconnection of the gamma shield cooling to the primary coolant loop. This figure is the same as Reference Drawing 762D192 in the reactor operating manual [Operation and Maintenance Manual, One-Megawatt Open Pool Reactor for Rhode Island Atomic Energy Commission, Providence, R.I., General Electric Document GEI-77793, October 1962]. The 1992 Safety Analysis Report [Safety Analysis Report for the Low Enriched Fuel Conversion of the Rhode Island Nuclear Science Center Research Reactor, Change 1 dated January 13, 1993] states that the "thermal shield is cooled by water which is currently forced around the shield using the pressure difference between the inlet and outlet primary coolant lines."

No flow rate is specified for the gamma shield because primary coolant flow rate is monitored. As long as the minimum primary flow rate is maintained, there is sufficient flow through the gamma shield. Additionally, there is a No Flow Thermal Column Flow Scram that serves as an auxiliary check that there is coolant flow through the gamma shield. The facility has a 43 year history of operating experience that shows that this coolant system is sufficient.

See also the response to RAI 10.3.

14.75 TS 3.2.1, Table 3.2, items 1 and 2 contain the acronym "FC." Define this acronym, and revise the proposed TS as appropriate.

Fourth Response Submitted September 8, 2010

FC stands for "Forced Convection" which represents the operating mode for which the inlet and outlet coolant temperature alarms are required. Tables 3.1 and 3.2 has been combined to indicate all of the safety channels and non-radiation monitoring safety related instrumentation that is required. The table has been revised to make the operating mode for which the channels and instruments are required, and the trip set points more clear. See the table in RAI Answer 7.1.

14.76 TS 3.2.1, Table 3.2 requires a log count rate blade withdrawal interlock with a set pointless than 3 counts per second. Explain why a set point less than 3 counts per second (e.g., a set point of 0 counts per second) is appropriate for this safety-related instrument, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

The purpose of this interlock is to ensure that this channel is functioning and detecting neutrons. Historically, a minimum count rate of 3 cps has been acceptable to indicate that this instrument is functional. This table will be updated to make this clear.

14.77 TS 3.2.1, Table 3.2 requires a servo control interlock with a set point of "30 sec (fullout)." What is the parameter to which the " 30 sec" set point applies? What is the component to which the "fullout" set point applies? Revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

The table will be revised to make it clear that there are two servo control interlocks in place. The first interlock prevents the operator from putting the rod control system into servo control if the Log N period is less than 30 seconds. The second interlock prevents the operator from putting the system into servo control if the regulating rod is not fully withdrawn (full out).

14.78 TS 3.2.1, Table 3.2 requires a building air gaseous exhaust (stack) monitor with a set point of "2.5 x normal particulate 2 x normal." Explain what this set point means. Clarify whether this single monitor fulfills the functions of monitoring both particulates and gaseous effluents, and revise the proposed TS as appropriate. (See RAI 14.103)

Fourth Response Submitted September 8, 2010

The building air Stack Monitor consists of two monitors housed in one unit. One monitor is a gaseous monitor, and the other is a particulate monitor. The two channels are entirely independent of each other.

All of the radiation monitors in the confinement room have set points that are in terms of "normal" radiation levels. The purpose of defining set points in terms of "normal" radiation levels is to account for the fact that the radiation levels vary in the confinement room, depending on what kinds of experiments are being performed.

14.79 TS 3.2.1, Table 3.2, item 10 requires a radiation monitor labeled "primary demineralizer (hot DI)." Explain what "hot DI" means, and revise the proposed TS as appropriate.

Second Response Submitted August 6, 2010

The primary demineralizer is the demineralizer that is used to clean up the primary pool water, as opposed to the make-up demineralizer. Since the reactor pool water has a small amount of Na-24 in it, some of the sodium accumulates in the demineralizer, making it radioactively "hot". The term “hot DI” has been used to refer to this demineralizer for the last fifty years. None of the current RINSC staff has knowledge about the origin of this term, but it is surmised that this term came about because this demineralizer has a tendency to be radioactively “hot”, and it is a demineralizer (DI).

No revision to the Technical Specifications is necessary.

14.80 TS 3.2.1, Table 3.2 contains footnote (b) which states, “The reactor shall not 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 levels.” Explain what “continuously operated” means. Explain why the radiation monitors subject to footnote (b) do not need to be operating for reactor operations that are not considered “continuous.” Explain how each radiation monitor located on the experimental level can individually provide adequate monitoring of the entire experimental level.

Second Response Submitted August 6, 2010

This question was addressed on September 22, 1995 when NRC approved Amendment Number 20 to the R-95 License. A copy of this amendment has been enclosed.

In the NRC Safety Evaluation supporting that amendment, “Continuous” operation was defined as operation for more than one 6 hour shift. The justification provided for allowing operation up to one 6 hour shift, was that:

> The purpose of the Stack Gaseous and Stack Particulate Monitors is to provide an alarm function to inform operations personnel of potential radiological releases from the stack.
>
> There are alternative radiation monitors with alarms that would be able to indicate a potential radiological release.
>
> As long as there is at least one monitor over the reactor pool and one monitor on the experimental level that would ensure that radiological releases would be detected and alarmed, NRC deemed that this would acceptably meet the monitoring requirements.

14.81 TS 3.2.2 requires all shim safety blades to be operable before the reactor is made critical. Explain why the regulating blade is not required to be operable before the reactor is made critical.

Second Response Submitted August 6, 2010

This specification requires the shim safety blades to be operable in accordance with TS 4.1.1 and 4.1.2. TS 4.1.1 defines when reactivity worths and insertion rates shall be measured. As part of the answer to RAI Question 14.64, these parameters are also required to be measured for the regulating rod as well. TS 4.1.2 defines when visual inspections of the shim safety blades are required to be performed. It is not possible to do visual inspections of the regulating blade because it is housed in a shroud. Consequently, in order to include the regulating blade in this specification to the extent possible, the following additional specification will be added:

3.2.5 The regulating rod is operable in accordance with Technical Specification 4.1.1.

14.82 TS 3.2.2 references the surveillance requirements of TS 4.1.1 and TS 4.1.2. Explain why TS 3.2.2 references these surveillance requirements.

Eighth Response Submitted January 24, 2011

Technical Specification 3.2 has bee re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.87. The new proposed shim safety LCO specifications are covered in Technical Specifications 3.2.1.1 and 3.2.1.2. The corresponding surveillance requirements are covered in Technical Specifications 4.2.1 and 4.2.2, which were submitted as part of the answer to RAI question 14.141.

14.83 TS 3.2.3 references the surveillance requirements of TS 4.2.5 and TS 4.2.6 (the reference to TS 4.2.6 appears to be incorrect). Explain why TS 3.2.3 references these surveillance requirements.

Eighth Response Submitted January 24, 2011

Technical Specification 3.2 has bee re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.87. The original TS 3.2.3 referred to the LCO regarding shim safety drop times. This is now covered in Specification 3.2.1.1. The original references to TS 4.2.5 and (incorrectly) 4.2.6 had to do with the surveillance requirement for shim safety drop times. These reference have been removed, though the surveillances are included as Specifications 4.2.1.1 an 4.2.1.2 in the revised version of Technical Specification 4.2 submitted as part of the answer to RAI question 14.141.

14.84 TS 3.2.4 appears to be a reactivity limit. Explain the reason for not including TS 3.2.4 in TS 3.1.

Fourth Response Submitted September 8, 2010

TS 3.2.4 will be moved to TS 3.1.11.

14.85 TS 3.2.4 specifies a maximum reactivity insertion rate for a single control or regulating blade of 0.02 %Δk/k per second. The bases for the TS state that the reactivity insertion rate limit was determined in the SAR, but the SAR does not appear to contain an analysis of a ramp insertion of 0.02% Δk/k per second. Section 13.2.5 provides an analysis of a startup accident, but the analyzed reactivity addition rate (0.0196% Δk/k per second) appears to be less conservative than the TS limit. Explain how the SAR supports the reactivity insertion rate limit in TS 3.2.4. If the SAR does not support the TS limit, provide an analysis that supports the TS limit. Alternately, revise the proposed TS to be consistent with the analysis in the SAR.

Fourth Response Submitted September 8, 2010

Analysis of a reactivity insertion of 0.02 %Δk/k per second is provided in the response to RAI 13.7.

14.86 The bases for TS 3.2.1 state, "the period scram limits the rate of rise of the reactor power to periods which are manually controllable." Table 3.1 indicates that the Log N Period trip channel set point is 4 seconds. The SAR does not appear to contain an analysis that shows how a reactor period slightly greater than 4 seconds would be manually controllable. Explain how a reactor period slightly greater than 4 seconds is manually controllable by the reactor operator.

Eighth Response Submitted January 24, 2011

The 4 second period limit serves as an auxiliary protection to assure that the reactor fuel would not be damaged in the event that there was a power transient.

As part of the answer to RAI question 13.7, an analysis was performed for a rapid insertion of 0.6 % dK/K reactivity from very low power. Effectively in this analysis, a step insertion of 0.6 % dK/K reactivity is inserted at low power and the power increases until the true power reaches the limiting safety system setting of 2.3 MW, at which point one of the over power trips cause a scram. It is assumed that it takes 100 ms for the control blades to start dropping into the core, and that it takes 1 second for full insertion. The analysis shows that the peak fuel temperature is well below the temperature required to damage the fuel.

An insertion of 0.6 % dK/K corresponds to a period of less than 1 second. Consequently, the consequences of a power excursion due to a 4 second period is covered by this analysis.

See the answer to RAI question 14.87 for the new basis given for the 4 second period scram.

14.87 The bases for TS 3.2 only discuss the reactor power, reactor period, and coolant flow scrams required by TS 3.2.1. Provide bases for the other safety channels and safety-related instrumentation required by TS 3.2.1, Table 3.1 and Table 3.2.

Eighth Response Submitted January 24, 2011

Technical specification 3.2 has been re-written to conform more closely to ANSI 15.1. Some of the specifications that had been in section 3.2 have been moved. The following table provides a summary of how things have been changed:

Original Location	**Specification**	**New Location**
3.2.1	Minimum Safety Instrumentation	3.2.1.3
		3.2.1.4
		3.2.1.5
3.2.2	Operability of Shim Safety Blades	3.2.1.1
3.2.3	Scram Time	3.2.1.1
3.2.4	Reactivity Insertion Rate	3.2.1.2

The radiation monitoring instrumentation described in the new RINSC Technical Specification 3.2.1.3 was taken from the description given as part of the answer to RAI question 7.4. References to specific radiation monitoring instrumentation have been removed in order to allow for more flexibility in using alternative monitoring equipment. References to specific radiation alarm setpoints have been removed. RINSC has a radiation safety program, which has safety committee oversight to ensure that ALARA principles are met. Radiation levels inside the reactor room are contingent on the number, and types of experiments that are in progress. Rather than defining setpoints with the caveat that they can be adjusted higher with the approval of the approval of the facility Director or Assistant Director, setpoints will be set in a manner that ensures that the goals of the Radiation Safety Program are met. Table 3.2 will be replaced with Specifications 3.2.1.3 and 3.2.1.4.

The reactor safety and safety related instrumentation described in the new RINSC Technical Specification 3.2.1.5 was taken from the description given as part of the answer to RAI question 7.1.

The bases for 3.2.1.1 and 3.2.1.2 refer to transient analyses that were part of the answer to RAI question 13.7.

The basis for Specification 3.2.1.4 is consistent with the answer given for RAI question 14.80 regarding the justification for being able to operate for six hours without the stack gaseous or particulate monitor.

The bases for Specification 3.2.1.5 regarding the safety limits, limiting trip values, and limiting safety system settings are consistent with the answer given for RAI question 14.36, except that the cooling modes for which the pool temperature, and primary coolant flow rate channels are required have been corrected. The basis regarding the inlet temperature channel is consistent with the answer given for RAI question 4.23. The basis regarding the outlet temperature channel is consistent with the answer given for RAI question 14.36. The basis regarding the pool temperature channel refers to the basis for Specification 2.2.2, which was updated as part of the answer to RAI question 14.52.

The new versions of Technical Specification 3.2 is:

3.2 Reactor Safety System

Applicability:

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

Objective:

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

Specification:

3.2.1 The reactor shall not be made critical unless:

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

3.2.1.2 The reactivity insertion rates of individual shim safety and regulating rods does not exceed 0.02% dK/K per second.

3.2.1.3 The following area radiation monitoring instrumentation is operable:

3.2.1.3.1 A minimum of one radiation monitor that is capable of warning personnel of high radiation levels shall be at the experimental level.

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

3.2.1.3.3 If either of these detectors fail during operation, the staff shall have one hour to either repair the detector, or find an acceptable replacement without having to shut the reactor down.

3.2.1.4 The following air radiation monitoring instrumentation is operable:

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

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

3.2.1.4.3 If either of these detectors fail during operation, the staff shall have six hours to either repair the detector, or find an acceptable replacement without having to shut the reactor down.

3.2.1.5 The following reactor safety and safety related instrumentation is operable and capable of performing its intended function:

Protection	Cooling Mode	Channels Required	Function	Set Point		
Over Power	Both	2	Scram by	Power Level	Less than or Equal to	105% of Licensed Power
Low Pool Level	Both	1	Scram by	Pool Level Drop	Less than or Equal to	23 ft 9.6 in
Primary Coolant Inlet Temperature	Forced	1	Alarm by	Inlet Temp	Less than or Equal to	111 F
Primary Coolant Outlet Temperature	Forced	1	Alarm by	Outlet Temp	Less than or Equal to	117 F
	Forced	1	Scram by	Outlet Temp	Less than or Equal to	120 F
Pool Temperature	Natural	1	Scram by	Pool Temp	Less than or Equal to	125 F
Primary Coolant Flow Rate	Forced	1	Scram by	Primary Flow Rate	Less than or Equal to	1800 gpm
Rate of Change of Power	Both	1	Scram by	Period	Less than or Equal to	4 seconds
Seismic Disturbance	Both	1	Scram if	Seismic Disturbance Detected		
Bridge Low Power Position	Forced	1	Scram if	Bridge Not Seated at HP End		
Bridge Movement	Both	1	Scram if	Bridge Movement Detected		
Coolant Gates Open	Forced	1	Scram if	Inlet Gate Open		
	Forced	1	Scram if	Outlet Gate Open		
Detector HV Failure	Both	1	Scram if	Detector HV Decrease	Less than or Equal to	50 V
	Both	1	Scram if	Detector HV Decrease	Less than or Equal to	50 V
	Both	1	Scram if	Detector HV Decrease	Less than or Equal to	50 V
No Flow Thermal Column	Forced	1	Scram by	No Flow Detected		
Manual Scram	Both	1	Scram by	Button Depressed		
	Both	1	Scram by	Button Depressed		
Servo Control Interlock	Both	1	No Automatic Servo if	Regulating Blade not Full Out		
	Both	1	No Automatic Servo if	Period	Less than	30 seconds
Shim Safety Withdrawal	Both	1	No SS Withdrawal if	Count Rate	Less than	3 cps
	Both	1	No SS Withdrawal if	Test / Select SW not Off		
Rod Control Communication	Both	1	Scram if	Loss of Communication	Less than or Equal to	10 seconds

Basis:

Specification 3.2.1.1 requires that all shim safety blades be capable of being fully inserted into the reactor core within 1 second from the time that a scram condition is initiated. As part of the Safety Analysis, Argonne National Laboratory analyzed a variety of power transients in which it was assumed that the time between the initiation of a scram signal, and full insertion of all of the shim safety rods was one second. The analysis showed that if the reactor is operated within the safety limits, this time delay will not cause an over power excursion to damage the fuel.

Specification 3.2.1.2 requires that the reactivity insertion rates of individual shim safety and regulating rods do not exceed 0.02% dK/K per second. As part of the Safety Analysis, Argonne National Laboratory analyzed ramp insertions of 0.02% dK/K reactivity from a variety of initial power levels. The reactivity insertions are stopped by the over power trip. In all cases, peak fuel and cladding temperatures due to the power overshoot are well below the temperatures required to damage the fuel or cladding. Consequently, this limit ensures that an over power condition due to a reactivity insertion from raising a control rod will not damage the fuel or cladding.

Specification 3.2.1.3 identifies the area radiation monitoring instrumentation that is required to be operable when the reactor is operated. Radiation monitors that are capable of warning personnel of high radiation levels at the experimental elevation, and over the pool serve to ensure that personnel inside the reactor room are made aware when dose rates are higher than anticipated. Additionally, these monitor alarms provide an indication of a potential fuel failure. In the event of a failure of either of these monitors, the operations staff is afforded the opportunity to rely on alternative monitoring instrumentation without having to shut the reactor down. This configuration has been in use for the life of the facility, without any indication that it is insufficient.

Specification 3.2.1.4 identifies the air radiation monitoring instrumentation that is required to be operable when the reactor is operated. Radiation monitors that are capable of warning personnel of high gaseous and particulate airborne radioactive material levels ensure that personnel are made aware of potential radiological releases from the stack. In the event of a failure of either of these monitors, the operations staff is afforded the opportunity to rely on alternative monitoring instrumentation without having to shut the reactor down. This configuration has been in use for the life of the facility, without any indication that it is insufficient.

Specification 3.2.1.5 identifies the safety and safety related instrumentation that is required to be operable when the reactor is operated.

Two independent power level channels are required for both forced and natural convection cooling modes of operation, each of which must be capable of scramming the reactor by 105% licensed power. The basis section of Specification 2.2.1 shows that this ensures that the power level safety limit of 2.4 MW will not be exceeded. Having two independent power level channels ensures that at least one over power protection will be available in the event of an over power excursion.

One low pool level channel is required for both forced and natural convection cooling modes of operation. This channel ensures that the reactor will not be in operation if the pool level is below the safety limit of 23 ft 6.5 inches above the top of the core.

One primary inlet coolant temperature channel is required for forced convection cooling mode operation. This channel alerts the operator in the event that the inlet temperature reaches 111 F. The steady state thermal hydraulic analysis that was done by Argonne National Laboratory for forced convection flow predicts that the inlet temperature would be 115 F for operation at 2.4 MW, with a primary flow of 1580 gpm and an outlet temperature of 125 F.

One primary outlet temperature channel is required for forced convection cooling mode operation. This channel is capable of scramming the reactor when the temperature reaches 120 F. The basis section of Specification 2.2.1 shows that this ensures that the coolant outlet temperature safety limit of 125 F will not be exceeded.

One pool temperature channel is required for natural convection cooling mode of operation. This channel is capable of scramming the reactor when the temperature reaches 125 F. The basis section of Specification 2.2.2 shows that this ensures that the pool temperature safety limit of 130 F will not be exceeded. This channel provides the over temperature protection when the reactor is operated in the natural convection cooling mode.

One primary coolant flow rate channel is required for forced convection cooling mode operation. This channel assures that the reactor will not be operated at power levels above 100 kW with a primary coolant flow rate that is less than the safety limit of 1580 gpm. The basis section of Specification 2.2.1 shows that if this channel is set to scram at a limiting safety system setting of 1800 gpm, the safety limit will not be exceeded.

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

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

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

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

One coolant gate open scram on each coolant duct is required during forced convection cooling mode operation. These scrams ensure that coolant flow through the inlet and outlet ducts are not bypassed during forced convection cooling.

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

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

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

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

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

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

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

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

14.88 The bases for TS 3.2 do not provide bases for TS 3.2.2 and TS 3.2.3. Provide bases for these TS.

Eighth Response Submitted January 24, 2011

Technical Specification 3.2 has been re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.87 for the bases to these specifications.

14.89 The bases for TS 3.2 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

Technical Specification 3.2 has been re-written in order to make it conform more closely to ANSI 15.1. See the answer to RAI question 14.87 for the revised bases to these specifications.

14.90 TS 3.3.a.3 appears to be a surveillance requirement and not a limiting condition for operation. Explain why TS 3.3.a does not specify a limit for primary coolant water radioactivity, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

The primary radioactive contaminants of the coolant water are tritium and sodium-24 during normal operation. The presence of other radioactive materials would be an indication of either an incipient fuel leak or a problem with an experiment. Specification 3.3.a.3 should be changed to read: "Except for tritium and sodium-24, the radioactivity in the primary coolant shall be maintained at levels statistically indistinguishable from background."

14.91 The "Applicability" section of TS 3.3.b includes cycles of chloride and resistivity. TS 3.3.b does not contain any specifications related to these parameters. Explain this apparent inconsistency, and revise the proposed TS as appropriate.

Seventh Response Submitted December 14, 2010

The specification refers to secondary coolant water. There is no need for a technical specification dealing with either chlorides or resistivity in the secondary coolant water. SAR Section 5.1, "Summary Description," starting at line 7 states: "The RINSC reactor is an open pool type reactor that uses de-mineralized water for primary coolant, shielding, and reactor moderator; and city water for secondary coolant. SAR Section 5.3.2, "Secondary Coolant System Operation," states: "City water is used as secondary coolant for both loops." SAR Section 5.5.2, "Secondary Makeup Water System," states "City water supplies the makeup water to the secondary coolant system." Starting at line 32 of SAR Section 5.5.2, the description states: "Historically the blow-down interval has been set such that the pH of the secondary water has been maintained between 5.5 and 9.0, which has kept mineral buildup and corrosion to a minimum." Since city water is being used, the applicability section of the technical specification should be reworded to say: "This specification applies to limiting conditions for secondary coolant pH and radioactivity." Please remove the words, "cycles of chloride," and "resistivity" from TS 3.3.b.

14.92 TS 3.3.b.2 appears to be a surveillance requirement and not a limiting condition for operation. Explain why TS 3.3.b.2 does not specify a limit for sodium-24 in the secondary coolant, and revise the proposed TS as appropriate.

Fifth Response Submitted November 26, 2010

For the secondary coolant to have measurable levels of radioactivity, a primary-to-secondary leak must be present. The presence of sodium-24 in the secondary coolant would be but one indication of such a leak. Specification 3.3.b.2 should be changed to read: "The radioactivity in the secondary coolant shall be maintained at levels statistically indistinguishable from background."