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

STATE OF RHODE ISLAND AND PROVIDENCE PLANTATIONS

RHODE ISLAND ATOMIC ENERGY COMMISSION
Rhode Island Nuclear Science Center
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Narragansett, RI 02882-1165

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

November 26, 2010

Re: Letter dated 13 April 2010
Docket No. 50-193

Dear Mr. Kennedy:

Enclosure one is attached in reply to your Request for Additional Information (RAI) dated April 13, 2010 regarding license renewal for the Rhode Island Nuclear Science Center Reactor (RINSC). The enclosure contains the fifth set of answers to the questions specified in your letter. The RINSC staff is continuing to work on the RAI questions that remain outstanding.

Very truly yours,

Michael J. Davis

Michael J. Davis. Assistant Director
Rhode Island Nuclear Science Center

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

Executed on: 11/26/10 By: Michael J. Davis

Docket No. 50-193

Enclosures: 1. Fifth Response to Request for Additional Information Letter Dated April 13, 2010

A035
A020
NRR

RAI Fifth Response

4.15 Section 4.6 references the computer code PLTEMP as the code used to determine many of the thermal-hydraulic characteristics of the reactor core. Provide a discussion of the use of this code including models of the RINSC core, applicability of the code to the thermal hydraulic conditions in the RINSC core, validation and benchmarking of the code, and code uncertainty. Provide a copy of Reference 4.6.

Revised: The revision is shown in green.

In Section 4.6 the original PLTEMP code[1] was used to determine many of the thermal-hydraulic characteristics of the reactor core. This code has been superseded by the PLTEMP/ANL V4.0 code.[2] The analysis of steady-state forced-convection operation has been redone using the newer code and is attached as a new Section 4.6 of the SAR, "4.6 Steady-State Thermal-Hydraulic Analysis" (Reference BB).

Although both the old and the newer code have models that can be used to obtain the flow distribution in a reactor core, a more direct, transparent, and tractable approach was taken in the new analysis. A hydraulics model of the RINSC core was developed based on engineering fundamentals. The equations used in the analysis are provided. Their application is explained in detail. Key intermediate results for a reactor flow rate of 1580 gpm are given in tables so that one can verify that the analysis was performed correctly. The hydraulics analysis yielded the flow rate for each fueled channel in the reactor as a function of total reactor flow rate.

In the next phase of the analysis, the individual channel flow rates obtained in the previous phase were used to perform a thermal analysis of the core. The limiting channel was identified as an internal channel in element D6 and next to the highest power fuel plate. The new PLTEMP code was used to perform thermal analysis of this limiting channel. The highest power fuel plate bounded by two channels, each with half of the flow area of single channel, was modeled. This model is simple and easy to check. For the channel flow rate corresponding to a reactor flow rate of 1580 gpm the PLTEMP results for the powers at which the onset of nucleate boiling and the onset of flow instability were predicted to occur were verified by a hand calculation. Key intermediate results used in the verification are given in tables so that one can verify that the results are correct.

In the new analysis, key results such as coolant flow rates, bulk coolant and clad surface temperatures, the conditions that produce onset of nucleate boiling and the conditions that cause flow instability are either performed by a hand calculation or are verified by one. Thus, PLTEMP code validation, benchmarking, and code uncertainty are less relevant because results that are

essential to demonstrate the safety of the RINSC core have been hand calculated or hand checked. Moreover, the PLTEMP/ANL code used in the new analysis is based on an evolutionary sequence of "PLTEMP" codes in use at ANL for the past 26 years.[3-9] Validation of PLTEMP/ANL has followed standard practice in any code development task, where comparisons are made with other codes, with measurements, and with hand calculations where possible. Many examples of validation are given in the PLTEMP/ANL V4.0 manual[2], and in References 3-9.

As a copy of the original PLTEMP code,[1] reference 4.6 in the 2004 SAR is provided.

Also provided is Reference 4-6:

K. Mishima, K. Kanda and T. Shibata, "Thermal-hydraulic Analysis for Core Conversion to Use of Low-Enrichment-uranium Fuels in KUR," KURRI-TR-258 (1984).

References:

1. Kaichiro Mishima, Keiji Kanda, and Toshikazu Shibata, "Thermal-Hydraulic Analysis for Core Conversion to the Use oaf Low-Enriched Uranium Fuels in the KUR," *Proceedings of the 1984 International Meeting on Reduced Enrichment for Research and Test Reactors*, Argonne National Laboratory, October 15-18, 1984.
2. Arne P. Olson and Kalimullah, "A Users Guide to the PLTEMP/ANL V4.0 Code," Global Threat Reduction Initiative (GTRI) – Conversion Program, Nuclear Engineering Division, Argonne National Laboratory, March 24, 2010.
3. K. Mishima, K. Kanda and T. Shibata, "Thermal-Hydraulic Analysis for Core Conversion to the Use of Low-enrichment Uranium Fuels in the KUR," KURRI-TR-258, Research Reactor Institute, Kyoto University, Dec. 7, 1984.
4. K. Mishima, K. Kanda and T. Shibata, "Thermal-Hydraulic Analysis for Core Conversion to the Use of Low-Enriched Uranium Fuels in the KUR," ANL/RERTR/TM-6, CONF-8410173, p. 375, 1984.
5. W. L. Woodruff and K. Mishima, "Neutronics and Thermal-Hydraulics Analysis of KUHFR," ANL/RERTR/TM-3, CONF-801144, p. 579, 1980.
6. W. L. Woodruff, "Some Neutronics and Thermal-hydraulics Codes for Reactor Analysis Using Personal Computers," Proc. Int. Mtg. on Reduced Enrichment for Research and Test Reactors, Newport, RI, Sept. 23-27, 1990, CONF-9009108 (ANL/RERTR/TM-18), Argonne National Laboratory (1993).
7. W. L. Woodruff, J. R. Deen and C. Papastergiou, "Transient Analyses and Thermal- hydraulic Safety Margins for the Greek Research Reactor (GRR1)," Proc. Int. Mtg. on Reduced Enrichment for Research and Test Reactors, Williamsburg, VA, Sept. 19-23, 1994, CONF-9409107 (ANL/RERTR/TM-20), Argonne National Laboratory (1997).

8. W. L. Woodruff, "A Kinetics and Thermal-hydraulics Capability for the Analysis of Research Reactors," Nucl. Technol., Volume 64, 196 (1983).
9. W. L. Woodruff and R. S. Smith, "A Users Guide for the ANL Version of the PARET Code, PARET/ANL (2001 Rev.)," ANL/RERTR/TM-16, Mar. 2001.

4.17 Provide the values of coolant temperature and coolant height used in the analysis of Section 4.6.2. Provide justification for the use of these values.

Revised: The revision is shown in green.

Section 4.6 in the 2004 SAR has been superseded by a new Section 4.6 and is attached as "4.6 Steady-State Thermal-Hydraulic Analysis" (Reference BB). In the new analysis the values of inlet coolant temperature and coolant height used in the analysis are based on the Technical Specifications Safety Limit values for reactor power (2.4 MW), reactor flow (1580 gpm), outlet temperature (125° F), and pool water depth (23.54 feet above the active core). These are slightly more limiting than was necessary since Limiting Safety System Setting values would have been sufficient.

The added pressure at the top of the active core due to the weight of the water at a depth of 23.54 feet is 989.8 kg/m^3 × 9.80665 m/s2 × 23.54 ft × 0.0254 × 12 m/ft = 69645 Pa = 0.696 bar. Narragansett, Rhode Island is at 20 feet above sea level, where atmospheric pressure is 1.013 bar. Thus, the absolute pressure at the top of the active core is 1.013 + 0.696 bar = 1.709 bar. A pressure of 1.7 bar was used in the analysis. The enthalpy at the core exit was obtained for this pressure and the 125° F via the NIST Steam Tables. The power to flow ratio of 2.4 MW / 1580 gpm yielded the enthalpy rise from core inlet to core outlet. The outlet enthalpy minus the enthalpy rise yielded the inlet enthalpy. The inlet enthalpy and 1.7 bar pressure yielded the inlet temperature of 114.5° F. This was rounded up to 115°F and used as the inlet temperature for the new Section 4.6 thermal analyses. The density of water at 115°F and 1.7 bar is 989.8 kg/m^3, which is the value use above to determine the added pressure due to the depth of the water.

Thus, the direct answer to the question is 115° F and 23.54 feet of water above the active core. The justification is that these values are consistent with the Technical Specifications Safety Limit values, which bound the Limiting Safety System Setting values.

4.19 Section 4.6.2. Describe the methods used to determine the values of $T_{surface}$ and Tonb found in Table 4-9 and Table 4-10. Include all assumptions and correlations used in the calculations and provide justification for their use given the thermal-hydraulic characteristics of the coolant channels.

Revised: The revision is shown in green.

Section 4.6 in the 2004 SAR has been superseded by a new Section 4.6 and is attached as "4.6 Steady-State Thermal-Hydraulic Analysis" (Reference BB). In

the new analysis the values of the clad surface temperature ($T_{surface}$) and the values of the surface temperature that cause onset of nucleate boiling (T_{onb}) are calculated by the PLTEMP/ANL V4.0 code.[1] A copy of the manual of the code is provided. The key heat transfer correlations used in the analysis of onset of nucleate boiling are the Bergles and Rohenow correlation, which is used to determine the amount of superheat to reach onset of nucleate boiling at the clad surface, and the Dittus-Boelter correlation, which is use to determine the Nusselt number. These two correlations and a full description of the models, assumptions, and correlations of the PLTEMP/ANL code can be found in Reference 1.

In the analysis for onset of nucleate boiling, the PLTEMP/ANL V4.0 code was used to analyze the limiting channel in the reactor. This analysis was performed once for each of eight reactor flow rates, spanning the flow range from 1000 to 2200 gpm. Since the model represented only the limiting channel, it was easy to hand-check the PLTEMP/ANL V4.0 results. This hand-check was performed for a reactor flow rate of 1580 gpm. At this flow rate the limiting channel has a flow rate of 0.2210 kg/s. For this flow rate PLTEMP/ANL V4.0 found that the channel power is 22.80 kW, which corresponds to a reactor power of 4.72 MW. The code results also indicate the axial locations where onset of nucleate boiling was first reached. Key values of the verification are provided in the new Section 4.6 in a table, which is repeated below as Table 4.19-1. Additional specifics of the analysis can be found in the new Section 4.6, including a listing of the PLTEMP/ANL V4.0 input used for this analysis.

Table 4.19-1 – Verification of PLTEMP/ANL Onset of Nucleate Boiling Prediction at the Limiting Axial Location for a Core Flow Rate of 1580 gpm

Quantity	Hand Calculation	PLTEMP
Channel Dimensions		
Thickness, in	0.088	
Width, in	2.62	
Heated Length, in	23.25	
Heat Width, in	2.395	
Wetted Perimeter, m	0.1376	
Hydraulic Diameter, m	0.004325	
Heat Transfer Area (2 Faces), m^2	0.07185	
Channel Flow Rate, kg/s	0.2210	
Channel Power, kW	22.80	
Pressure, bar	1.7	
Inlet Temperature, C	46.11	

Saturation Temperature, C	115.15	
Cp @ 55 C, kJ/kg-C	4.1828	
At Onset of Nucleate Boiling Location		
Layer	13	13
Channel Power to Middle of Layer, kW	15.64	
Channel Power to Exit of Layer, kW	16.56	
Local Peak-to-Average Power	1.2894	
Bulk Temperature at Middle of Layer w/o Hot Chan. Fac., C	63.03	63.01
Random Hot Channel Factor on ΔT bulk	1.24	
Bulk Temperature with Hot Channel Factor, C	67.09	67.06
Bulk Temperature at Exit of Layer w/o Hot Chan. Fac., C	68.32	
Viscosity, Pa-s	4.3910E-4	
Reynolds Number	14634	
Thermal Conductivity, W/m-C	0.65817	
Cp, kJ/kg-C	4.1867	
Prandtl Number	2.793	
Nusselt Number without Hot Channel Factors	74.55	
Global Film Coefficient Hot Channel Factor	1.2	
Film Coefficient with Hot Channel Factor, W/m^2-C	9454	
Heat Flux without Hot Channel Factors, MW/m^2	0.4092	0.4091
Random Hot Channel Factor on ΔT film	1.28	
Film Temperature Rise with Hot Channel Factor, C	55.77	
Clad Surface Temperature with All Hot Channel Factors, C	122.5	122.5
Random Hot Channel Factor on Heat Flux	1.23	
Heat Flux with Hot Channel Factors, MW/m^2	0.4779	
ΔTsaturation based on Bergles and Rohsenow, C	7.41	
Surface Temperature For Onset of Nucleate Boiling, C	122.6	122.6

Reference:

Arne P. Olson and Kalimullah, "A Users Guide to the PLTEMP/ANL V4.0 Code," Global Threat Reduction Initiative (GTRI) – Conversion Program, Nuclear Engineering Division, Argonne National Laboratory, March 24, 2010.

4.21 Section 4.6.2 states that with a flow rate of 1,950 gallons per minute (gpm), the incipient boiling temperature (defined in the SAR as $_{Tonb)}$ occurs at about 2.6 MW. From Table 4-9, it appears that the incipient temperature occurs somewhere between 1,715 gpm and 1,800 gpm. Clarify this apparent discrepancy.

Revised: The revision is shown in green.

Section 4.6 in the 2004 SAR has been superseded by a new Section 4.6 and is attached as "4.6 Steady-State Thermal-Hydraulic Analysis" Reference BB. In the new analysis Table 4-9 is longer relevant. A more detailed explanation, if one is needed, follows.

Table 4.9 represents a double search. First a range of solutions for various core pressure drops are searched and interpolated to find the one that provides the correct total core flow rate. Then at the desired core pressure drop and flow rate, solutions for a range of assumed reactor power levels are searched and interpolated to find the one that provides the clad surface temperature corresponding to the onset of nucleate boiling. In the new analysis, on the other hand, for each specific reactor flow rate, the flow rate of each fueled channel is determined via a custom-developed hydraulics model. (This model and its governing equation are described in the new Section 4.6 in considerable detail. Key intermediate results for a reactor flow rate of 1580 gpm are given in tables so that one can verify that the analysis was performed correctly.) The new Section 4.6 analysis uses the PLTEMP/ANL V4.0 code[1] to find the power at which onset of nucleate boiling occurs. This code has an internal search method, which determines the power level at which onset of nucleate boiling is first achieved and provides the power level value as a code output. Thus, in the new analysis the double search is not used and Table 4-9, or another similar table, is not needed.

In the new analysis, the onset of nucleate boiling for a reactor flow rates of 1715, 1800, and 1,950 gpm are predicted to occur at a reactor powers of 5.1, 5.3, and 5.7 MW, respectively. The increased allowed power is, in part, attributable to 1) a reduction in the hot channel factors due to more use of statistical, rather than multiplicative, methods of combining uncertainty factors and 2) improvements in the determination of the reactor power distribution.

Reference:

Arne P. Olson and Kalimullah, "A Users Guide to the PLTEMP/ANL V4.0 Code," Global Threat Reduction Initiative (GTRI) – Conversion Program, Nuclear Engineering Division, Argonne National Laboratory, March 24, 2010.

4.23 Note number 2 to Table 4-12 states that the calculations are based on a reactor inlet temperature of 42.3 degrees C. Explain the reason this temperature is used in the analysis given that it is less conservative than the coolant temperatures allowed by the proposed TS.

Revised: The revision is shown in green.

It is true that an inlet temperature of 42.3° C (108° F) it is less conservative than the coolant temperatures allowed by the proposed TS. Section 4.6 in the 2004 SAR has

been superseded by a new Section 4.6 developed for these RAI responses and is attached as "4.6 Steady-State Thermal-Hydraulic Analysis" Reference BB. In the new analysis the inlet temperature is assumed to be 115°F (46.11° C). As explained in the answer to Question 4.17, this is based on the Technical Specifications Safety Limit values for reactor power (2.4 MW), reactor flow (1580 gpm), outlet temperature (125° F), and pool water depth (23.54 feet above the active core).

4.26 Section 4.6.4 gives a surface temperature of 122.93 degrees C for normal flow at 2.6 MW. Table 4-9 indicates that this temperature corresponds to 1,715 gpm at 2.6 MW. Page 4-3 indicates that nominal flow is 1,950 gpm. Clarify this apparent discrepancy.

See the response to RAI 4.24. A new analysis has been completed for which this discrepancy no longer exits.

4.27 Section 4.6.4 states, "for a hot channel analysis, the ONB region would not present a problem for the LEU fuel." Provide justification for this conclusion.

See the response to RAI 4.24.

4.28 Provide the values of coolant temperature and coolant height used in the analysis of Section 4.6.5. Provide justification for the use of these values.

Section 4.6.5 of the 2004 RINSC Reactor SAR provides an analysis of the thermal behavior of the LEU core during steady-state operation in the natural-convection mode. This analysis has been completely redone and is replaced by section 4.7 of Reference AA.

Reference AA refers to the completely redone analysis of the thermal behavior of the LEU core during steady-state operation in the *forced*-convection mode, which is provided in sections 4.6.1 through 4.6.12 of the attached Reference BB and is a replacement for sections 4.6.1, 4.6.2, 4.6.3, and 4.8 of the 2004 RINSC Reactor SAR.

In the Reference AA analysis of natural convection, the inlet (and pool) coolant temperature is 130° F and the coolant height is 23 feet 6.5 inches (23.54 feet) above the active core. These are the Safety Limit values and are more restrictive than the Limiting Trip values of 128° F and 23 feet 9.1 inches, respectively. (See the table at the end of the response to RAI 14.52.)

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.

4.29 Table 4-17. Explain the meaning of the negative value for Margin to Incipient Boiling at 209.1 kW. (See RAI 14.35)

Table 4-17 is part of section 4.6.5 of the 2004 RINSC Reactor SAR, which provides an analysis of the thermal behavior of the LEU core during steady-state operation in the natural-convection mode. This analysis 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.

The meaning of the negative value of Margin to Incipient Boiling at 209.1 kW in the 2004 RINSC SAR implies that the incipient boiling occurs before 209.1 kW is reached. However, in the analysis of section 4.7 of Reference AA, incipient boiling, which is referred to as "onset of nucleate boiling", is predicted to occur at 369 kW with all uncertainties included. The Limiting Trip value of power is 125 kW and the Safety Limit power is 200 kW. (See the table at the end of the response to RAI 14.52.) Thus, the Reference AA analysis shows a large margin to incipient boiling.

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.

4.32 Section 4.8. Provide a discussion of the correlation and/or calculations used to develop the Departure from Nucleate Boiling (DNB) and Departure from Nucleate Boiling Ratio (DNBR). Include all assumptions made in the analysis and justification for those assumptions. Clarify whether the term "Margin to Departure from Nucleate Boiling" in Table 4-19 is synonymous with the term "DNBR."

Sections 4.6.1, 4.6.2, 4.6.3, and 4.8 of the 2004 RINSC Reactor SAR provide the analysis of the thermal behavior of the LEU core during steady-state operation in the forced-convection mode. This analysis has been completely redone in

sections 4.6.1 through 4.6.12 of the attached Reference BB and is a replacement for sections 4.6.1, 4.6.2, 4.6.3, and 4.8 of the 2004 RINSC Reactor SAR

All assumptions made in the analysis and justification for those assumptions are provided in detail in Reference BB, which includes the hydraulic modeling needed to obtain the individual channel flow rates, the determination of the hot channel factors, a description of the PLTEMP/ANL V4.0 code, which was used in the analysis, and a listing of the code input used in the analysis. In Reference BB section 4.6.10, "Results of Steady-State Thermal Analysis," and section 4.6.11, "Discussion" describe the determination of the power at which departure from nucleate boiling (DNB) is predicted to occur and include a discussion of the correlations and the calculations that were used.

The new analysis does not use either the term "departure from nucleate boiling ratio" or the term "DNBR". Instead, the new analysis indicates the power in kW or MW at which departure from nucleate boiling is predicted to occur.

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.

9.1 Section 9.1.2 provides no detail regarding the design specifications of the normal and emergency ventilation system other than general arrangement. TS 3.7.2 credits the ventilation system with a dilution of waste streams by a factor of 4 x 10^4. Provide sufficient details regarding both the normal and emergency ventilation system flows to confirm the appropriateness of the dilution factor.

Chapter 13 will be re-written based on the basis document entitled "Fuel Damage Radiological Assessment". In this analysis, no credit is taken for dilution air. Consequently, there is no longer a need to justify a dilution factor of 4 X 10^4.

10.1 Section 10.2.1 does not describe the design features of the beam port covers or administrative controls regarding beam port use that support the assumptions made in the loss-of-coolant-accident (LOCA) presented in Section 13.2.3 of the SAR. Provide a description of the design features and administrative controls that is consistent with the assumptions made in the LOCA analysis.

Section 13.2.3, Loss of Coolant Accident (LOCA), has been completely replaced with the analyses provided in the re-written section of RINSC SAR Chapter 13.2.3 "Loss of Coolant Accident".

If one of the beam ports is severed, all six beam ports are flooded because of the common interconnected drain lines. The revised LOCA analysis assumes that water can flow from each of the six beam ports onto the reactor floor. Administrative controls are needed to guarantee that the flow resistance in each

beam port is equal to or more restrictive than a round half-inch diameter, sharp-edged orifice at the exterior (beyond the reactor shielding) end of the beam port. This could be accomplished by having a cover on the exterior of each beam port that seals the beam port, except for an optional hole with a diameter of up to one half inch.

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)

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.

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.

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.

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.

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.

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.

13.20 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.

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.

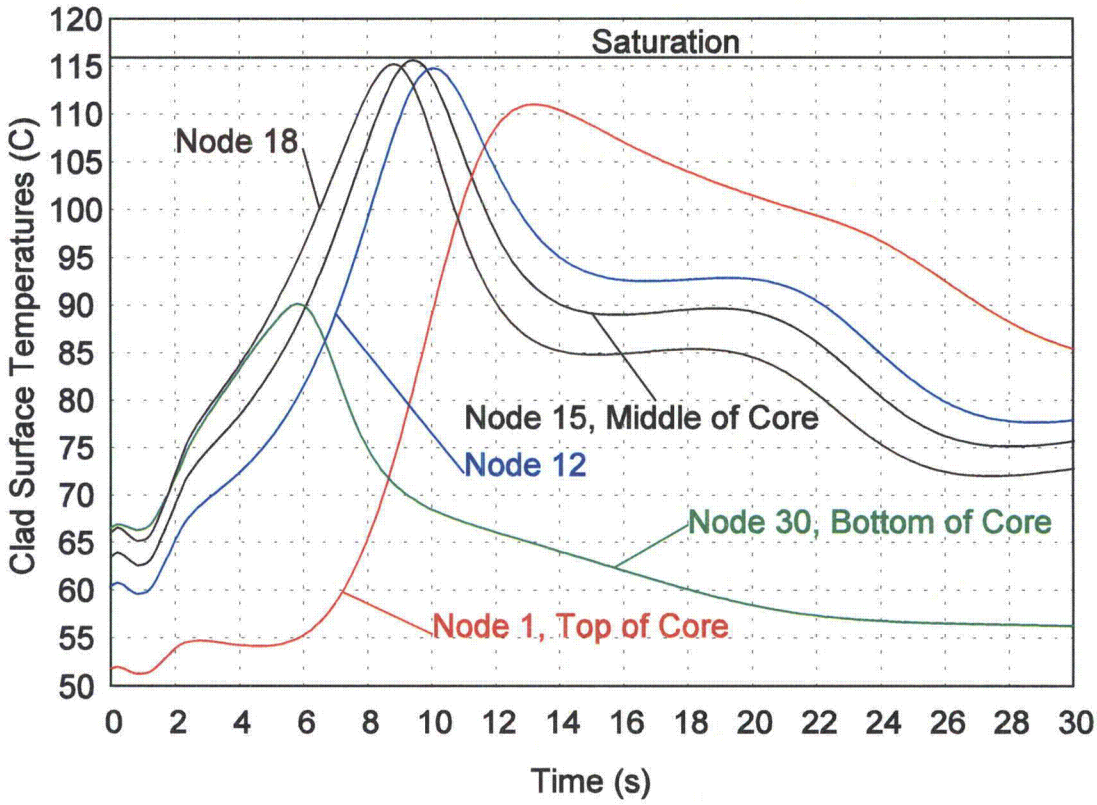
In the second step, the spatial mesh was refined in the X-Y plane through D6 and E6 in order to capture additional spatial detail regarding power distribution with the fuel assembly. The spatial mesh spanning D6 and E6 was increased in order to provide a mesh interval for every plate, and one for each 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.

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 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.

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)

See the response to RAI 4.29.

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)

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.

14.62 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)

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.

14.90 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.

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."

14.92 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.

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."

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

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

3.4 Confinement

3.4.1 Operations That Require Confinement

Applicability:

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

Objective:

To assure that operations that have the potential to release airborne radioactive material are performed under conditions in which the release to the environment would

be detected, and be limited to levels below 10 CFR 20 limits.

Specification:

1. The confinement system shall be operable whenever:

    1. The reactor is operating.
    2. Irradiated fuel handling is in progress.
    3. Experiment handling is in progress for an experiment that has a significant fission product inventory, and for which the experiment is not inside a container.
    4. Any work on the core or control rods that could cause a reactivity change of more than 0.65% dK/K is in progress.
    5. Any experiment movement that could cause a reactivity change of more than 0.65% dK/K is in progress.

Bases:

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

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

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

These specifications ensure that the confinement system will be operable during conditions for which there is any potential for fuel cladding damage or failure to occur, as

well as for experiment failures in which fission products could potentially be released.

3.4.2 Components Required to Achieve Confinement

3.4.2.1 Normal Operating Mode Confinement

Applicability:

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

Objective:

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

Specification:

1. The following confinement system components shall be operable:

    1. Normal Personnel Access Door
    2. Roll Up Door
    3. Roof Hatch

Bases:

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

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

3.4.2.2 Emergency Operating Mode Confinement

Applicability:

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

Objective:

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

Specification:

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

    1. Emergency Confinement System Buttons
    2. Confinement Air Intake Damper
    3. Confinement Air Exhaust Damper
    4. Emergency Personnel Access Door

Bases:

Under emergency conditions, operability of any of the emergency confinement system buttons allows the path of the airflow from confinement, through the ventilation system to be changed so that it goes through the emergency filter. Operability of the confinement air intake and exhaust dampers allows the confinement building to be isolated from the outside so that no exhaust confinement air escapes through a pathway other than the emergency pathway. Emergency mode operation of the ventilation system ensures that under emergency conditions, confinement air will be drawn through the emergency filter before being exhausted through the stack. Operability of the filter minimizes the environmental consequence of a potential airborne radioactivity release. Emergency mode operation of the ventilation system also ensures that dilution air will be added to the confinement air from the emergency filter. Operability of the emergency

personnel access door allows the reactor operator to have a confinement egress route that does not require the individual to go through the main confinement room. When the door is shut, confinement is maintained.

3.4.3 Conditions Required to Achieve Confinement

3.4.3.1 Normal Operating Mode Confinement

Applicability:

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

Objective:

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

Specification:

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

1. The Normal Personnel Access Door is closed, except for entry and exit.
2. The Roll Up Door is closed.
3. The Roof Hatch is closed.
4. The Emergency Personnel Access Door is closed, except for entry and exit.
5. The Confinement Dampers are Open.
6. The negative differential pressure inside confinement with respect to the outside is at least 0.5 inches of water.

Bases:

Normal confinement is maintained by keeping all of the doors and the roof hatch closed, except for entry

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

3.4.3.2 Emergency Operating Mode Confinement

Applicability:

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

Objective:

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

Specification:

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

1. The Normal Personnel Access Door is closed, except for entry and exit.
2. The Roll Up Door is closed.
3. The Roof Hatch is closed.
4. The Emergency Personnel Access Door is closed, except for entry and exit.
5. The Confinement Dampers are Closed.
6. The negative differential pressure inside confinement with respect to

the outside is at least 0.5 inches of water.

Bases:

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

3.5 Ventilation System

3.5.1 Ventilation System Components Required for Normal Operating Mode

Applicability:

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

Objective:

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

Specification:

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

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

        1. Roughing Filter

2. Absolute Filter

3. Confinement Exhaust Stack

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

Bases:

The Confinement Exhaust Blower produces a differential pressure across confinement to ensure that all confinement air pathways are through controlled pathways. The Confinement Exhaust Filter System ensures that the majority of the radioactive particulates that would be likely to be released in the event of a fuel failure would be filtered out prior to being released to the environment, until the emergency operating mode ventilation system is activated. The Confinement Exhaust Stack ensures that the plume of confinement air that is released to the environment, is released at an elevation of 115 feet above ground level, which provides for an opportunity for the air to disperse prior to the plume reaching ground level.

3.5.2 Ventilation System Components Required for Emergency Operating Mode

Applicability:

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

Objective:

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

Specification:

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

1. Emergency Blower

2. Emergency Filter System, which shall include:

    1. Emergency Filter Intake System Roughing Filter
    2. Emergency Filter System Intake Absolute Filter
    3. Emergency Filter System Charcoal Filter
    4. Emergency Filter System Exhaust Absolute Filter

3. Dilution Blower
4. Confinement Exhaust Stack

2. The exhaust rate through the emergency filter shall be less than or equal to 1500 cfm.
3. The emergency filter is at least 99% efficient at removing iodine.
4. The Emergency Filter System Exhaust Absolute Filter shall be certified by the manufacturer to have a minimum efficiency of 99.97% for removing 0.3 micron diameter particulates.

Bases:

Under emergency conditions, the Confinement Exhaust Blower turns off, and differential pressure across confinement is maintained by the Emergency Blower. The Emergency Blower directs confinement air through the Emergency Filter to remove any radioactive iodine that would be expected to be released during a fuel failure. An airflow limit of 1500 cfm though the filter ensures that the flow rate is low enough to allow the filter to adsorb at least 99 % of the iodine that would be expected to be released in the event of a fuel cladding failure. The Emergency Filter System Exhaust Absolute Filter prevents charcoal particulates from the charcoal filter from being released to the building exhaust air stream. The Dilution Blower provides a non-contaminated source of air to mix with the confinement air, so that any airborne radioactivity that is released is diluted prior to release. The Confinement Exhaust Stack ensures that the plume of confinement air that is released to the environment, is released at an elevation of 115 feet above ground level, which provides

for an opportunity for the air to disperse prior to the plume reaching ground level.

3.6 Emergency Power

3.6.1 Required Emergency Power Sources

Applicability:

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

Objective:

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

Specification:

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

Bases:

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

3.6.2 Components Required to be Supplied with Emergency Power

Applicability:

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

Objective:

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

Specification:

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

    1. Emergency Blower
    2. Dilution Blower

Bases:

In the event of a power outage, the reactor will scram due to the loss of magnet current to the shim safety blades. The confinement air intake and exhaust dampers are pneumatically operated and will fail closed to isolate the confinement room. The confinement exhaust blower will shut off due to loss of power. As long as the emergency and dilution blowers continue to be operable, the emergency confinement system will continue to perform its intended function. In the event of a power outage, the emergency power source will supply the emergency and dilution blowers with electricity so that they will continue to operate, and the emergency confinement system will continue to be functional.

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

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

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

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

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

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

Specification 3.4.3.1.6:

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

Specification 3.4.3.2.6:

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

Specification 3.5.2.2:

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

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

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

Specification 3.4.2.2:

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

Specification 3.4.3.2:

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

Specification 3.5.2:

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

Specification 3.5.2.3:

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

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

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

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

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

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

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

In the event of a power outage, the reactor will scram due to the loss of magnet current to the shim safety blades. The confinement air intake and exhaust dampers are pneumatically operated and will fail closed to isolate the confinement room. The confinement exhaust blower will shut off due to loss of power. Confinement is only maintained if the emergency confinement system is turned on.

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

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

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

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

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

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

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

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

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

without a minimum of one radiation monitor on the experimental level of the reactor building and one monitor over the reactor pool operating and capable of warning personnel of high radiation areas. If one looks carefully at the asterisk, one would note that the radiation monitors are on the reactor bridge, [redacted] while the thermal column is on level 3, there is only one radiation monitor on the experimental level normally, i.e., the one by the thermal column. This configuration has existed for many years and the safety analysis did not identify a need for any additional monitoring. Based on the safety analysis in Chapter 13, the critical area radiation monitor is the one on the reactor bridge since it is the first warning of a fuel failure (MHA). This technical specification will be changed and updated when the technical issues relating to the MHA have been resolved.

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

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

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

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

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

computer program.

Surveillance Requirement: Airborne effluents shall be monitored by a continuous air monitor installed, calibrated and maintained in accordance with ANSI 13.1. The annual total effective dose equivalent to the individual member of the public likely to receive the highest dose from air effluents will be calculated using a generally-accepted computer program.

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

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

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

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

Thus, during routine operations, no individual would be in the pathway of the plume more than about 10% of the time. Calculations of annual dose equivalent due to the primary airborne effluent, Argon-41, using the COMPLY Code show less than the allowable ALARA limitation given in 10 CFR 20.1101 for the hypothetical maximum exposed individual member of the general public.

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

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

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

Please replace the technical specification with the following:

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

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

Please see the response to RAI 14.113.

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

Please see the response to RAI 14.113.

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

requirement pertain to the sum of all experiments or individual experiments? Explain how this requirement ensures compliance with the ALARA dose constraint of 10 CFR 20.1101(d). (See RAI 14.123)

Please change TS 3.8.10 to read:

> "The radiation dose received by any individual member of the public shall not exceed 100mrem (1mSv) in any calendar year as a result of all experiments conducted at the facility.
>
> The radiation dose in unrestricted areas shall not exceed 2mrem (0.02mSv) in any one hour from any single experiment or set of experiments.
>
> Annual air emissions of radioactive materials from routine operations and all experiments conducted shall not result in doses greater than 10mrem (0.1mSv) total effective dose equivalent (TEDE)"

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

Please see the response to RAI 14.120.

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

Please see the response to RAI 14.120.

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

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

4.4 Confinement System Surveillance

4.4.1 Normal Operating Mode Confinement System

Applicability:

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

Objective:

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

Specification:

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

Bases:

If the conditions specified in section 3.4.3.1 are met, then the normal operating mode confinement system is functioning. By ensuring that the normal operating mode confinement system is functional prior to each day of reactor start-up, conditions are verified to be in place to make certain that any airborne radioactivity release would be directed to the stack, mixed with dilution air, and detected by the stack radiation monitor system.

4.4.2 Emergency Operating Mode Confinement System

Applicability:

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

Objective:

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

Specification:

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

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

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

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

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

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

Bases:

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

4.5 Ventilation System Surveillance

4.5.1 Normal Operating Mode Ventilation System

Applicability:

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

Objective:

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

Specification:

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

Bases:

By ensuring that the normal operating mode ventilation system is functional prior to each day of reactor start-up, conditions are verified to be in place to make certain that any airborne radioactivity release would be directed to the stack and be detected by the stack radiation monitor system.

4.5.2 Emergency Operating Mode Ventilation System

Applicability:

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

Objective:

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

Specification:

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

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

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

    1. The following blowers are de-energized:

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

2. The following blowers are energized:

    1. Emergency Exhaust Blower
    2. Dilution Blower

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

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

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

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

Bases:

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

4.6 Emergency Power System Surveillance

Applicability:

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

Objective:

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

Specification:

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

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

    1. Biennially
    2. Following emergency system load changes

Bases:

Periodic tests of the emergency power system ensures that in the event of a facility power outage, the emergency power system would automatically start, and be capable of handling the load required to power the emergency confinement system. The testing periods that are specified conform to ANSI 15.1 recommendations.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1. Section 3.4.2.2.1.1 requires that the confinement and clean up systems become activated when an evacuation button is depressed.
2. Section 3.4.3.2 covers the requirement that gas leaks between confinement and the outside shall be inward.
3. Section 3.4.3.2.6 requires that a negative differential pressure be maintained between confinement and the outside.
4. Section 3.5.1.1.2 requires that the confinement room exhaust air go through a roughing filter and an absolute filter prior to being released to the environment.
5. Section 3.5.2.1.2 describes the components of the emergency filter system that are required.
6. Sections 3.5.2.1.2 and 3.5.2.1.4 require the confinement exhaust to exit through an emergency filter system and a stack.
7. Section 3.5.2.3 requires that the charcoal filter in the emergency filter system be capable of removing 99% of the radioiodines likely to be present in the event of a fuel failure.
8. Section 3.5.2.4 requires that the exhaust absolute filter for the emergency filter system be certified by the manufacturer to be capable of removing 0.3 micron particulates.
9. Section 3.6 requires that emergency power be available to operate the clean-up system in the event of a facility power failure.
10. Section 4.4.2.2.2 requires that the dampers close when an evacuation button is depressed.
11. Section 4.4.2.2.4 requires that the confinement room vent fans and HVAC system shut off when an evacuation button is depressed.

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

5.0 Design Features

5.1 Site and Facility Description

The Rhode Island Nuclear Science Center (RINSC) is located on a 3 acre section of a 27 acre auxiliary campus of the University of Rhode Island. The 27 acre site was formerly a military reservation prior to becoming the Bay Campus of the university. The parcel of land is located in the town of Narragansett, Rhode Island, on the west shore of the Narragansett Bay, approximately 22 miles south of Providence, and approximately 6 miles north of the entrance of the bay from the Atlantic Ocean.

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

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

5.2 Reactor Coolant System

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

The primary section of the forced convection cooling system takes water from the pool outlet line, and directs it to a delay tank where its progress through the cooling system is held up for approximately 70 seconds in order to reduce the N-16 concentration in the water. From the delay tank, the forced cooling system is divided into two loops. Each primary loop consists of a pump and a heat exchanger.

For each loop, the water from the delay tank goes through a primary pump, through a primary heat exchanger, and back to the pool via the pool inlet line, where the two loops recombine. The piping for the primary cooling system is aluminum. Nominal temperatures and pressures are less than 130 F and less than 100 psig respectively.

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

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

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

5.3 Reactor Core and Fuel

The RINSC fuel is MTR plate type fuel that has a nominal enrichment of 19.75% U-235. The chemical composition of the fuel is U_3Si_2. Each fuel assembly consists of 22 fuel plates, bound by side plates that hold the plates evenly spaced apart. At each end of the assembly, the side plates are attached to square end boxes,

that are capable of being inserted into a core grid box. The cladding, side plates, and end boxes are aluminum. Each fresh fuel assembly is loaded with 275g U-235 nominal.

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

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

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

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

5.4 Fissionable Material Storage

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

Fixed racks that are mounted on the pool wall

Moveable racks that rest on the pool floor

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

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

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

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

Applicability:

This specification applies to the fissionable material storage facilities used for storing materials while they are not in use, or in an approved shipping container.

Objective:

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

Specification:

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

Bases:

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

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

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

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

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