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
16 Reactor Road
Narragansett, RI 02882-1165

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

September 8, 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 fourth 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. Assistant Director
Rhode Island Nuclear Science Center

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

Executed on: 9/8/10 By:

Docket No. 50-193

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

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4.10 Section 4.5. Describe and justify the methods used to determine the reactor kinetics parameters found in Table 4-3. Provide the names of any codes used and a description of the modeling process, if applicable.

The reactor kinetics parameters were recomputed for the equilibrium core using standard perturbation methods with Argonne's VARI3D computer code[1]. VARI3D uses the DIF3D diffusion theory code to compute real and adjoint flux solutions for use in perturbation theory.

The kinetic parameters were recalculated here because the models that were used to calculate the values in Table 4-3 in the early 1990's are no longer available. The table below shows that the recomputed values of the delayed neutron fraction and the prompt neutron generation time agree very well with those in Table 4-3.

Equilibrium Core Kinetic Parameters

	2004 Report	2010 Calculation	
		With equilibrium Xe	No Xe
Delayed Neutron Fraction, β-eff (%)	0.764	0.755	0.756
Neutron Generation Time, μs	68.3	69.4	68.6
Delayed Parameters by Family Group			
β(1)		2.6580E-04	2.6591E-04
β(2)		1.3707E-03	1.3713E-03
β(3)		1.3188E-03	1.3194E-03
β(4)		2.8985E-03	2.8996E-03
β(5)		1.1990E-03	1.1995E-03
β(6)		5.0074E-04	5.0095E-04
λ(1)		1.3337E-02	1.3337E-02
λ(2)		3.2712E-02	3.2712E-02
λ(3)		1.2075E-01	1.2075E-01
λ(4)		3.0279E-01	3.0279E-01
λ(5)		8.4966E-01	8.4966E-01
λ(6)		2.8538E+00	2.8538E+00

[1]A.P. Olson to J. Matos, "Availability of the VARI3D Code on Linux", Argonne National Laboratory Internal Memorandum, February 11, 2004.

4.11 Section 4.5. Describe and justify the calculation methods for the coefficients of reactivity for temperature, void, and power. Discuss any measurements made to confirm the reactivity coefficients. Include estimates of accuracy for the coefficients.

Coefficients of reactivity for temperature, void, and power were recomputed for the equilibrium core for these RAI responses because the models and results in Section 5 were originally computed in the early 1990's and are no longer available.

Neutron cross sections in seven energy groups as functions of moderator temperature, fuel temperature, and coolant void fraction were prepared using the WIM/ANL cross section generation code1. Keff values were computed using the DIF3D diffusion theory code. Coefficients of reactivity were determined from these data.

RINSC does not make measurements of these parameters. The accepted values for these coefficients are the values provided by the model.

4.12 Table 4-3 lists coolant reactivity coefficients for the coolant temperature range of 20-40 degrees Celsius (degrees C). TS 3.2.1 allows operation of the reactor with coolant temperatures up to 126 degrees Fahrenheit (degrees F) (52 degrees C). Provide coolant reactivity coefficients over the entire coolant temperature range allowed by the TS.

Separate reactivity coefficients were recalculated for this RAI for increasing coolant temperature only and increasing coolant density only using the methods described in RAI 4.11. The reactivity coefficients are shown below for a coolant temperature range between 20 °C and 100 °C.

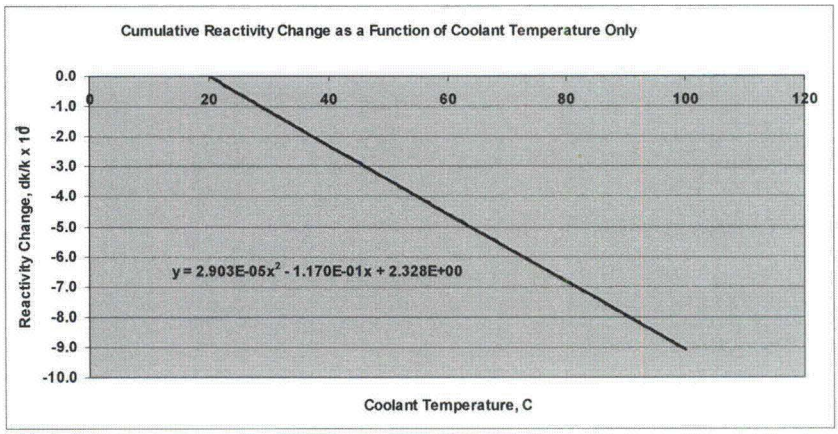
Equilibrium Fission Products

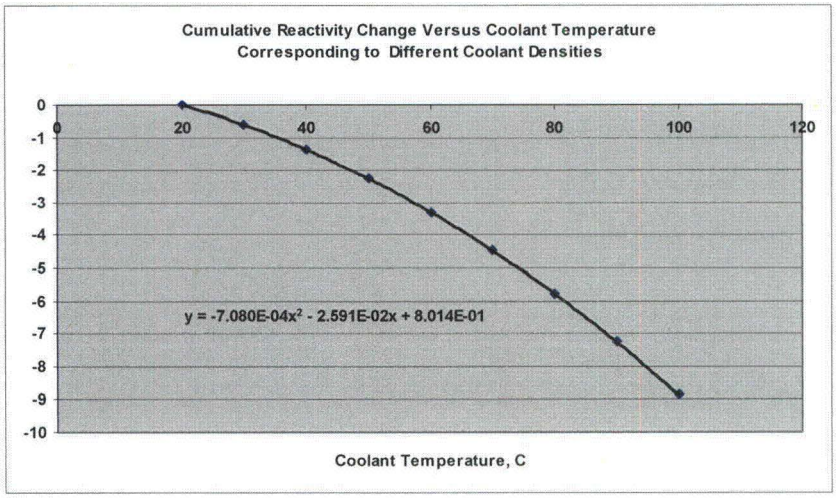
Change Coolant Temperature Only

Water Temp. °C	Reactivity, dk/k	Cumulative Reactivity Change x 10^3 Rel. to 20 °C
20	0.027602	0.00
30	0.026447	-1.1551
40	0.025298	-2.3044
50	0.024155	-3.4478
60	0.023017	-4.5855
70	0.021885	-5.7174
80	0.020759	-6.8434
90	0.019639	-7.9636
100	0.018524	-9.0781

Change Coolant Density Only

Water Temp. °C	Water Density mg/ml	Reactivity, dk/k	Cumulative Reactivity Change x 10^3 Rel. to 20 °C
20	0.99811	0.027249	0.00
30	0.99564	0.026636	-0.6131
40	0.99227	0.025881	-1.3678
50	0.98810	0.024985	-2.2641
60	0.98323	0.023947	-3.3020
70	0.97773	0.022767	-4.4815
80	0.97171	0.021446	-5.8026
90	0.96525	0.019983	-7.2653
100	0.95845	0.018379	-8.8696



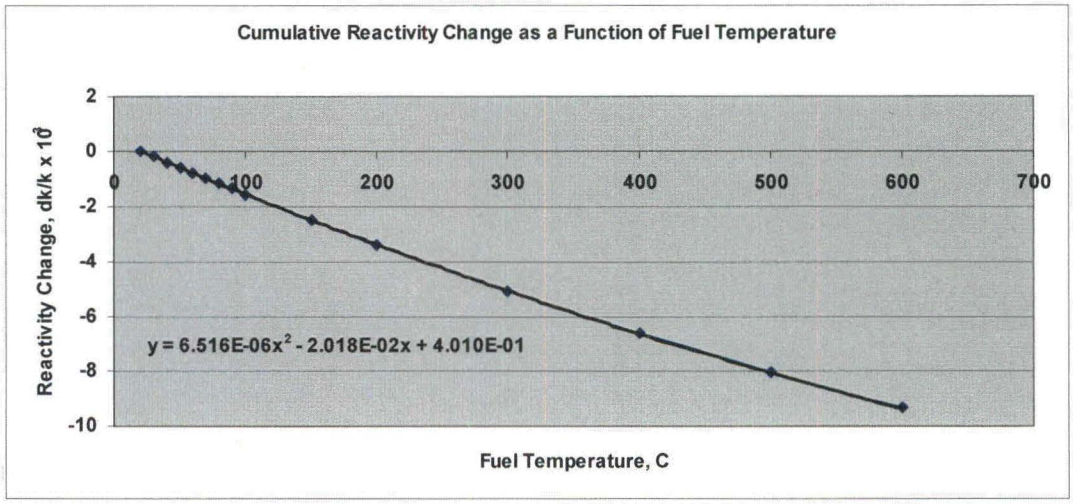


4.13 Table 4-3 includes the Doppler coefficient of reactivity over the temperature range of 20-40 degrees Celsius. This temperature range appears to apply to the coolant temperature and not the fuel temperature. Provide the Doppler coefficient of reactivity over the range of fuel temperatures anticipated during all allowed modes of reactor operation and reactor transients.

Reactivity coefficients were recalculated for this RAI for increasing the fuel temperature only using the methods described in RAI 4.11. The reactivity coefficients are shown below for a fuel temperature range between 20 °C and 600 °C.

Equilibrium Fission Products Change Fuel Temperature Only

	Reactivity, Δk/k	Cumulative Reactivity Change x 10^3 Rel. to 20 °C
20	0.027479	0.0
30	0.027280	-0.19854
40	0.027083	-0.39578
50	0.026887	-0.59172
60	0.026693	-0.78635
70	0.026499	-0.97968
80	0.026307	-1.17170
90	0.026117	-1.36243
100	0.025927	-1.55185
150	0.025000	-2.47940
200	0.024105	-3.37437
300	0.022412	-5.06657
400	0.020851	-6.62845
500	0.019419	-8.06001
600	0.018118	-9.36125



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.

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.

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

The users guide to the PLTEMP/ANL V4.0 code.[2] 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 of 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). 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.

Section 4.6 in the 2004 SAR will be superseded by a new Section 4.6. 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 the original 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.

4.18 Section 4.6.2. Confirm that the units for the values of Tsurface, Tsat, and T_{onb} found in Table 4-9 and Table 4-10 are degrees C.

It is confirmed that the values of Tsurface, Tsat, and T_{onb} found in Table 4-9 and Table 4-10 are degrees C.

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.

Section 4.6 in the 2004 SAR will be superseded by a new Section 4.6. 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 Rohsenow 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 used 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 the PLTEMP/ANL V4.0 code manual.

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

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

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.

Section 4.6 in the 2004 SAR will be superseded by a new Section 4.6. 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 memo dated 3 September 2010 from Earl E. Feldman to James E. Matos entitled "Steady State Thermal Hydraulic Analysis for Forced Convective Flow in the Rhode Island Nuclear Science Center (RINSC) Reactor" 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 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.

4.22 Section 4.6.2 states that the reduced flow trip setting is 1,700 gpm. The requirements of TS 2.2.1 and TS 3.2.1 allow the reduced flow trip to be set at 1,600 gpm. Clarify this apparent discrepancy.

Section 4.6 in the 2004 SAR will be superseded by a new Section 4.6. See the response to RAI 14.36 and 14.32.

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.

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 will be superseded by a new Section 4.6 developed for these RAI responses. In the new analysis described in the memo dated 3 September 2010 from Earl E. Feldman to James E. Matos entitled "Steady State Thermal Hydraulic Analysis for Forced Convective Flow in the Rhode Island Nuclear Science Center (RINSC) Reactor", 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).

4.24 The rising power transient analysis of Section 4.6.4 of the SAR shows that the reactor power would reach 2.78 MW. Explain how the analysis supports the safety limit of 2.4 MW given in TS 2.1.1.1. (See RAI 14.32)

The rising power transient analysis of Section 4.6.4 of the SAR is misplaced and will be moved to the section on reactivity insertion accident analyses in Chapter 13.

The analysis in the SAR was redone using the PARET/ANL code for a slow insertion of reactivity from high power. Since the power level is 2.0 ± 0.2 MW, including 10% uncertainty in the power level measurement, an initial power of 1.8 MW was selected to achieve the maximum rise in power after the trip on power is initiated at 2.3 MW. The description of the transient is given below.

The reactor is initially operating at 1.8 MW, 123 F coolant inlet temperature, and 1740 gpm. There is a water head of 23' 9.1" above the top of the fuel meat, which provides a pressure of 1.715E+5 Pa. Then a long, slow ramp reactivity insertion begins at a ramp rate of 0.02 % Δk/k / s, continuing for 100 s. Power rises slowly. The power trip at 2.3 MW is actuated at 6.774 s. Since no actual negative reactivity from the control system occurs for 100 ms after the trip, the reactor power continues to rise from the trip level of 2.3 MW to a maximum of 2.313 MW at 6.874 s. The reactor power drops rapidly to shutdown conditions.

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

4.25 Section 4.6.4. The assumptions used in the rising power transient calculation are not consistent with the requirements of the TS. The analysis assumes a minimum reactor period of slightly greater than 7 seconds, while the TS allow a minimum reactor period of 4 seconds. Also, the surface temperature value of 122.93 degrees C appears to be based on a coolant flow rate of 1,715 gpm, which is greater than the TS requirement of 1,600 gpm. Provide a revised calculation that supports the requirements in the TS. Include justification of all assumptions, including the assumed coolant temperature and coolant height.

See the response to RAI 4.24.

6.1 Describe the operating parameters and design features of the confinement, including: free volume, number and type of penetrations, etc.

The volume of the confinement building is approximately 203,695 cubic feet. The volume of the pool structure and water is approximately 21,740 cubic feet, leaving 181,955 cubic feet of open space. The control room takes up about 3,612 cubic feet of this space, with the air changed regularly with the confinement air for temperature and humidity control.

During normal operation air enters the confinement building through the Confinement Air Intake opening. This air will disperse throughout the confinement building until it is collected at the Normal Ventilation Intake, where air is drawn in by way of the Confinement Exhaust Blower. A sample of this air is collected through a copper tube and delivered to a

Continuous Air Monitor in the basement, known as the Stack Monitor. This monitor samples the air for gaseous and particulate contamination then returns the sample to the exhaust line. The air is then sent up the stack, where it is diluted with air from outside the EPZ pulled in by the Dilution Blower, then released to the atmosphere at 115 feet above ground. Air from the various irradiation and experiment facilities are collected and combined with the confinement exhaust before the Confinement Exhaust Blower.

At five locations throughout the facility there are red evacuation buttons. Three are within the confinement building: one to the right of the portal entrance, one to the right of the roll-up door, and one in the control room. In addition to an audible alarm heard throughout the building, pressing any of these buttons will cause the confinement air handling systems to go into emergency mode. When this is initiated power is cut to the Confinement Exhaust Blower, the Off Gas Blower, and the Rabbit Blower. As the flow rate drops pneumatic switches trigger the pneumatic Exhaust and Intake Dampers to close. As this occurs, the Emergency Blower is energized, pulling air through the Emergency Ventilation Intake. This air is passed through a series of filters, including a charcoal filter, removing the majority of Iodine that may be present. This air is then directed to the Emergency Ventilation Exhaust where the air is diluted with air from outside the EPZ that is pulled in by the Dilution Blower. This air then travels up the stack and is released 115 feet above ground level. Because the normal sampling location for stack exhaust activity no longer represents the air being released to the environment, sampling is instead taken from approximately half way up the stack. This is achieved by way of a motorized three-way valve located at the Stack Monitor that is activated during emergency mode.

All other major penetrations through the confinement remain closed during operation, with the exception of the Lab Wing Portal, which utilizes a secondary door to maintain confinement integrity. During operation, the confinement building maintains at least a one-half inch of negative pressure in relation to the outside environment. Any other penetrations not mentioned, such as for coolant piping or wiring, are sealed or provide negligible effects on potential radioactive release or unauthorized access. A summary of penetrations are provided in the following table:

Summary of Confinement Penetrations

[redacted]

7.1 Provide a listing of the interlocks of the reactor protection system.

Table 3.1will be revised to be:

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

7.4 Section 7.2.15. Provide more detail regarding the fixed radiation monitoring instrumentation. Include a list of the positions and types of fixed monitors and indicate which have local readouts and/or alarms.

Available Radiation Instrumentation:

Instrument	Position	Detector Type	Readout	Alarm	Setpoint
Stack Gaseous Monitor	Stack	Gamma Sensitive	Local and Control Room	Local and Control Room	2.5 X Normal
Stack Particulate Monitor	Stack	Gamma Sensitive	Local and Control Room	Local and Control Room	2 X Normal
Main Floor Particulate Monitor	Reactor Main Floor	Gamma Sensitive	Local Only	Local Only	2 X Normal
Reactor Bridge Area Monitor	Reactor Bridge Area	Gamma Sensitive	Local and Control Room	Local and Control Room	2 X Normal
Fuel Safe Area Monitor	Fuel Safe Area	Gamma Sensitive	Local and Control Room	Local and Control Room	Higher of 2 X Normal or 5 mR/hr
Thermal Column Area Monitor	Thermal Column Area	Neutron Sensitive	Local and Control Room	Local and Control Room	Higher of 2 X Normal or 2 mR/hr
Heat Exchanger Area Monitor	Heat Exchanger Area	Gamma Sensitive	Local and Control Room	Local and Control Room	2 X Normal
Primary Clean-Up Demineralizer Area Monitor	Primary Clean-Up Demineralizer Area	Gamma Sensitive	Local and Control Room	Local and Control Room	2 X Normal

Required Radiation Instrumentation:

1. Area Radiation Monitors:

   A. A minimum of one radiation monitor that is capable of warning personnel of high radiation levels shall be at the experimental level. The Thermal Column Area Monitor, or equivalent may serve in this capacity.

   B. A minimum of one radiation monitor that is capable of warning personnel of high radiation levels shall be over the pool. The Reactor Bridge Area Monitor, or equivalent may serve in this capacity.

   C. Alarm set points may be adjusted higher with the approval of the Director or Assistant Director.

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

2. Air Monitors:

A. A minimum of one radiation monitor that is capable of warning personnel of high radiation levels in the confinement gaseous effluent shall be operating. The Stack Gaseous Monitor, or equivalent may serve in this capacity.

B. A minimum of one radiation monitor that is capable of warning personnel of high radiation levels in the confinement particulate effluent shall be operating. The Stack Particulate Monitor may serve in this capacity.

C. Alarm set points may be adjusted higher with the approval of the Director or Assistant Director.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The reactor is initially operating at 1.8 MW, 123 °F coolant inlet temperature, and 1740 gpm. There is a water head of 23' 9.1" above the top of the fuel meat, which provides a pressure of 1.715 x 10^5 Pa. The coolant inlet temperature for which an outlet temperature of 123 °F is reached was iteratively determined to be 113.6 °F (45.34 °C). Starting from this initial condition, a long, slow ramp reactivity insertion begins at a ramp rate of 0.02 % Δk/k / s [TS 3.2.4], continuing for 100 s. Power rises slowly. The power trip at 2.3 MW is actuated at 6.774 s. Since no actual negative reactivity from the control system occurs for 100 ms after the trip, the reactor power continues to rise from the trip level of 2.3 MW to a maximum of 2.313 MW at 6.874 s. The reactor power drops rapidly to shutdown conditions.

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

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

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

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

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

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

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

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

Reference

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

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

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

See the analysis of reactivity insertions provided in RAI 13.7

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

See the response to Question 13.19.

13.19 Section 13.2.4.1 states that the peak clad temperature during a loss-of-flow-accident induced by a loss of electrical power is 103 degrees C.

Provide an analysis that supports this statement. Justify all assumptions made in the analysis.

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

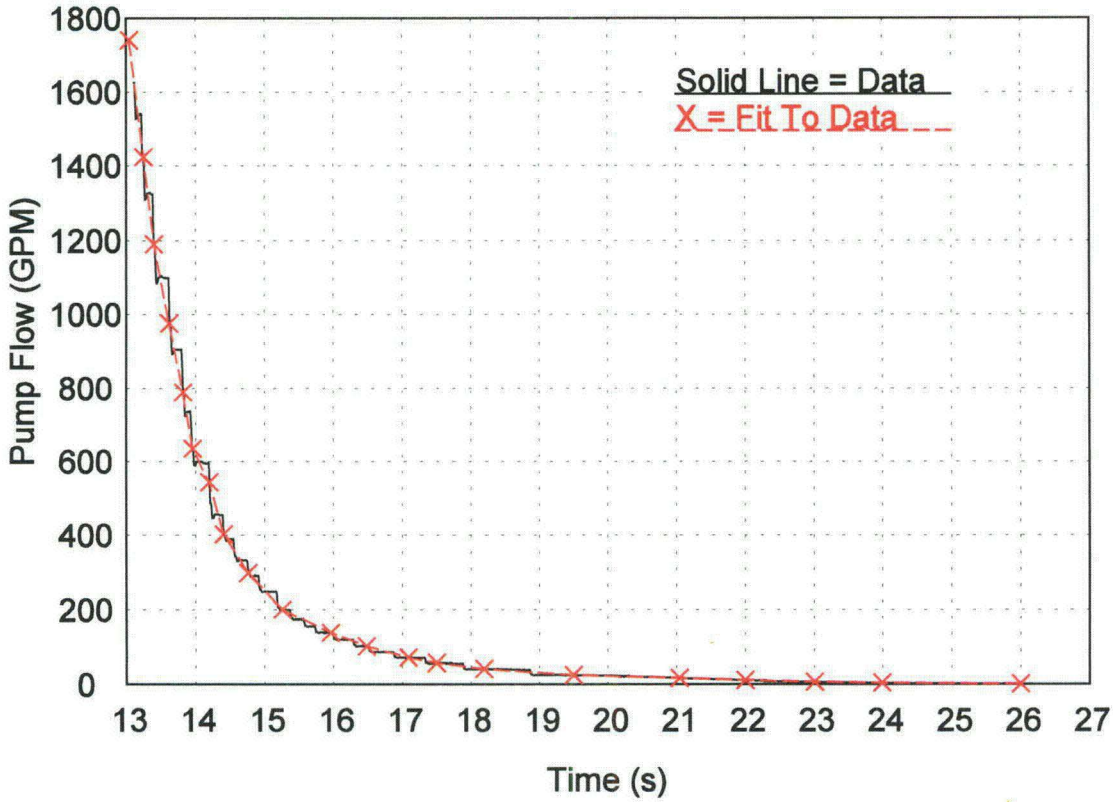
The RELAP5 model included the peak fuel channel, representing the highest power stripe in the highest power plate and its coolant. This model also included the average fuel channel, representing the rest of the fuel plates and their coolant. Bypass flow around the graphite reflectors, through the beryllium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

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

Initial Steady-State Conditions for the LOF Cases

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

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



Measured Pump Coastdown Curve

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

Key Parameters for the Loss-of-Flow Transients

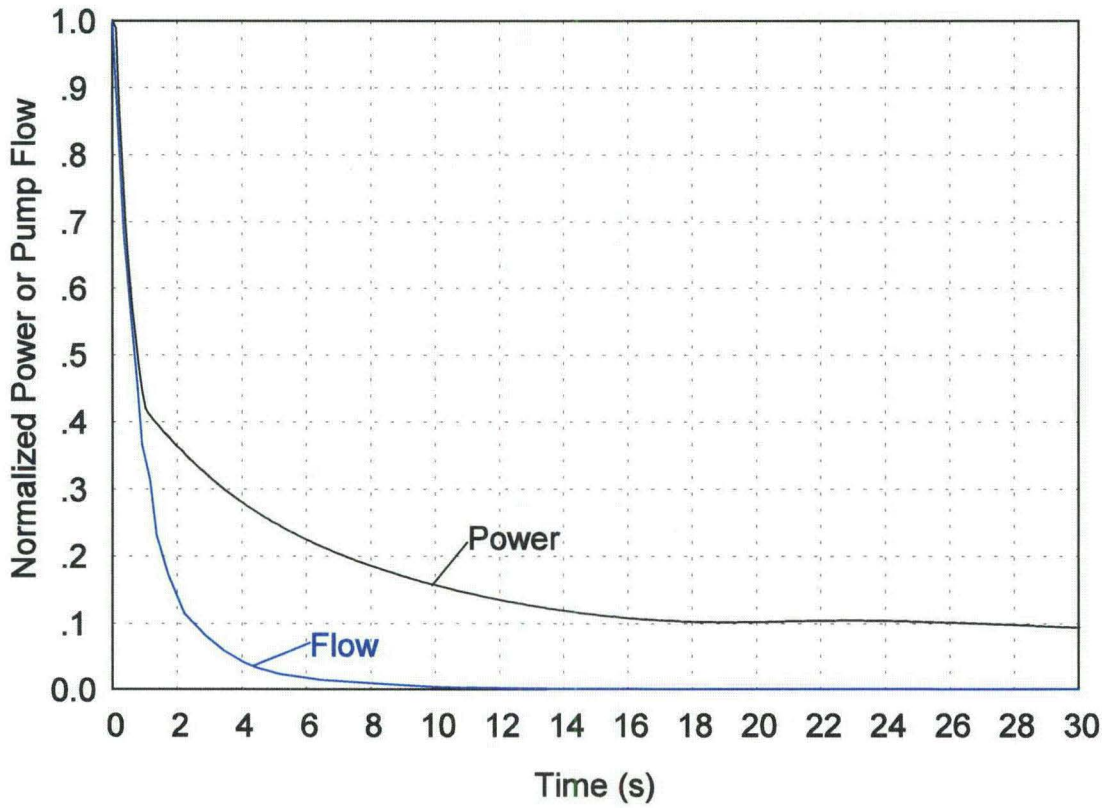
Parameter	Value	Justification
Pump coast-down	As shown in figure above	Measured
Time when natural circulation gate valves open	9 seconds after start of pump coast-down	Measured
Scram reactivity	1 % Δk/k	Limiting Conditions for Operation
Scram reactivity insertion time	1 second	Limiting Conditions for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

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

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

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

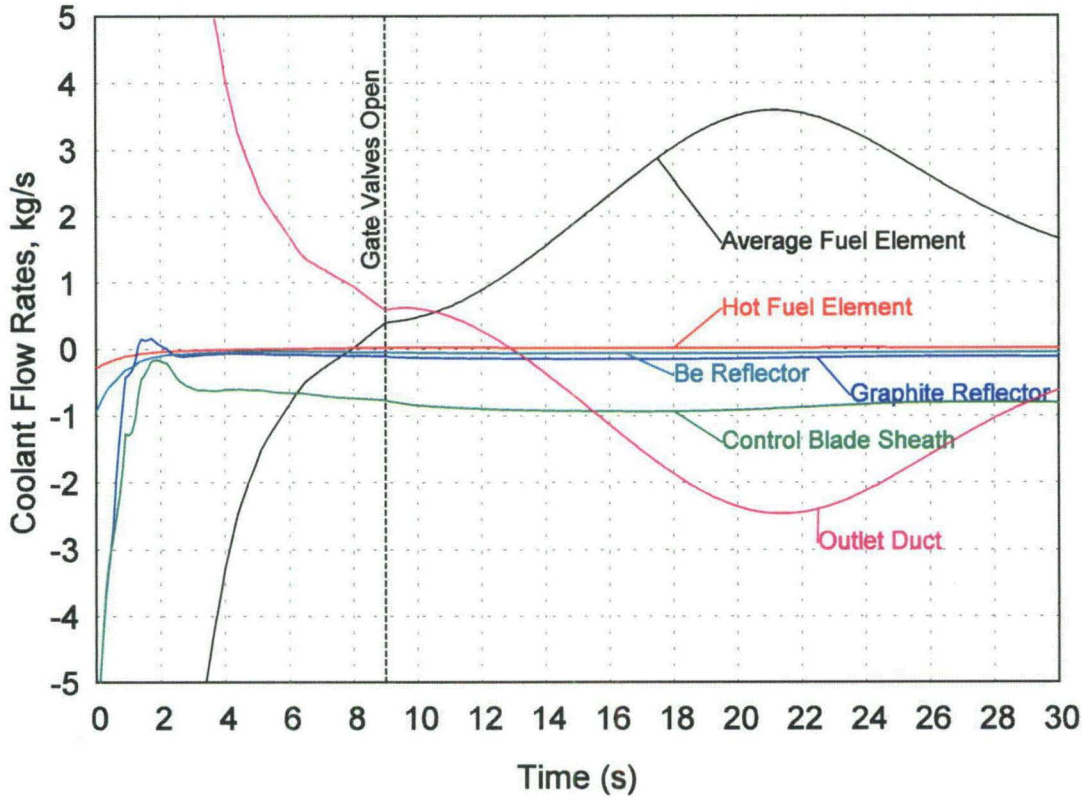
The normalized power and pump flow for the base case are shown in the figure below. For the first second, while the control blades are inserting, the power and flow fall at about the same rate. After that the power falls slower, and the pump flow drops to zero.



Power and Pump Flow for the RINSC LOF

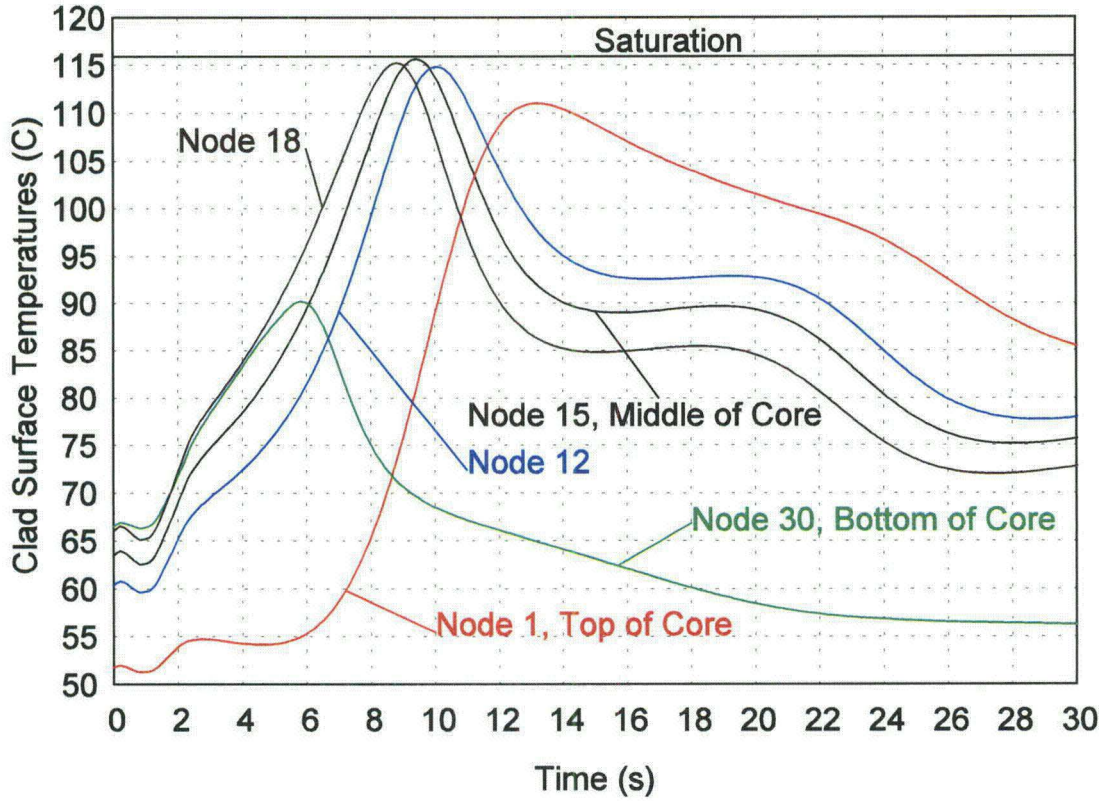
Flow rates through the fuel channels, the reflectors, the control rod sheaths and the outlet duct are shown in the figure below. In this figure positive flow rates are upward, and negative flow rates are downward. As

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



Flow Rates for the RINSC LOF

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



Clad Surface Temperatures in the Peak Fuel Channel

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

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

The smallness and the sign of the difference in peak clad surface temperature due to the failure of the gate valves to open can be explained by two factors. First, in both cases the peak temperature occurred only 0.41 second after the gate valves should open. The flow rate figure above indicates that the change in the outlet duct flow rate in the first 0.41 second after the gate valves open is small, so the change in peak clad temperature should be small. Second, close examination of the flow rate figure near 9 seconds indicates that if the gate valves open the outlet duct upward flow rate immediately after the gate valves open is slightly higher

than what would be obtained by extrapolating the flow rate from before the gate valves open. This leads to slightly more coolant being sucked out of the outlet plenum and slightly less up-flow through the fuel channels, causing slightly higher temperatures in the fuel channels if the gate valves do open.

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

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

14.33 The bases for TS 2.1.1 reference a version of the SAR that is different than the version of the SAR submitted with the license renewal application. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

See the revised TS 2.1.1 in RAI 14.32.

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

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

Revise Section 2.2.2 as follows.

2.2.2 Limiting Safety System Settings in the Natural Convection Mode

Applicability:

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

Objective:

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

Specifications:

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

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

Bases:

These specifications were set to prevent coolant temperatures from approaching the value at which damage to fuel cladding could occur (see NUREG-1313, "Safety Evaluation Report related to the Evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Nonpower Reactors"). Power and temperature limits were chosen to ensure that the integrity of the cladding is maintained even under transient conditions. The

uncertainty in the pool temperature measurements is ± 2%. The uncertainty in the measured power level under natural convection conditions is ±10% (see RAI 4.20 response). The uncertainty in the measurement of the pool height is estimated to be 0.5 in. At the limits of the uncertainty bands, there are still margins of 75 kW, 2 °F and 2.6 in. to the SL values for power, pool temperature and pool height, respectively. The following table summarizes the bases for the LSSS settings.

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

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

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

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

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

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

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

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

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

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

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

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

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

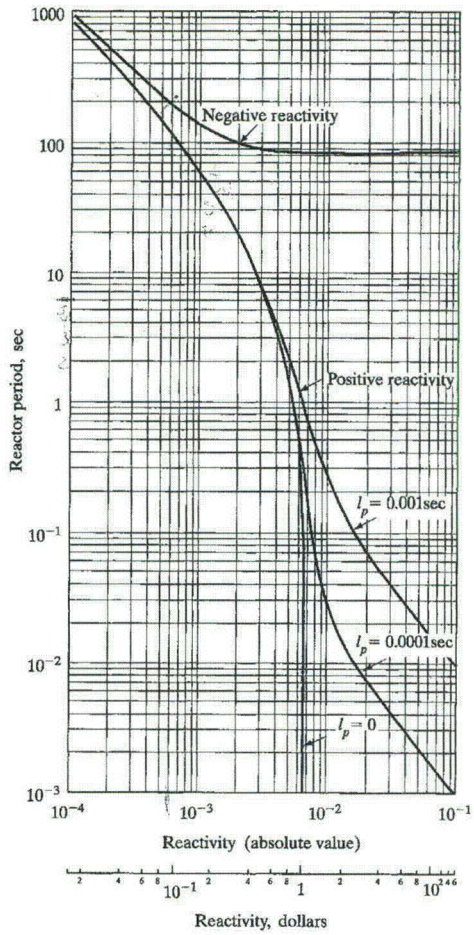


Figure 7.2 Reactor period as a function of positive and negative reactivity for a ^{235}U-fueled reactor.

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

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

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

1. At t = 0.0 seconds the reactivity insertion begins, and reactor power begins to increase.
2. At t = 0.1 seconds the reactivity insertion ends, and reactor power continues to increase.
3. At t = 30 seconds a stable period of about 75 seconds is reached.
4. At t = 1166.6 seconds power reaches 2.3 MW, and the over power trip is actuated. At this point, feedback reactivity from Doppler,

water expansion, and voids cause the period to decrease to approximately 1375 seconds, effectively making power constant at 2.3 MW.

5. The model assumed that there would be a 100 msec delay between the time that the trip was actuated, and the time that negative reactivity from the control system would begin to be inserted. Consequently, at t = 1166.7 seconds, the reactor power is still approximately 2.3 MW, but negative reactivity begins to be inserted.
6. Reactor power drops rapidly to shutdown conditions.

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

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

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

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

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

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

LEU Core #1 Startup Core Configuration

[redacted]

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

LEU Core #2 Configuration

[redacted]

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

LEU Core #3 Configuration - 2 May 03

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

LEU Core #4 Configuration - 24 October 2008

[redacted]

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

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

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

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

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

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

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

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

The bridge misalignment switch is attached to the end of the track at the high power section. When the bridge is moved away from the high power section the switch is released and the bridge misalignment scram is

triggered. This channel is not necessary or functional during natural convection cooled operation.

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

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

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

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

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

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

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

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

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

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

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

TS 3.2.4 will be moved to TS 3.1.11.

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

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

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

See the answer to RAI question 14.78.

14.137 TS 4.1.3 requires measurement of an experiment's reactivity worth prior to the "initial use" of the experiment. The bases for TS 4.1.3 state, "The specified surveillance relating to the reactivity worth of experiments will assure that the reactor is not operated for extended

periods before determining the reactivity worth of experiments." The specification and bases imply that the reactor can be operated without determining the reactivity worth of experiments. Explain how TS 4.1.3 ensures that the experiment reactivity requirements of TS 3.1 are met, and revise the proposed TS as appropriate.

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

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

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

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

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

3.1.3 The total reactivity worth of experiments shall not exceed the following limits, except when the operation of the reactor is for the purpose of measuring experiment reactivity worth:

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

3.1.4 The maximum reactivity worth of any individual experiment shall not exceed the following limits, except when the operation of the reactor is for the purpose of measuring experiment reactivity worth:

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

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

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

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

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

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

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

Revise TS 4.7.a.3 to say:

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

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

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

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

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

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

Change this section to say:

4.9.a Beryllium Reflectors

Applicability:

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

Objective:

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

Specification:

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

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

2. The beryllium reflectors shall be visually inspected and functionally fit into the core grid box on a rotating basis not to exceed five years such that:

    a. The surveillance each year shall include at least one fifth of the beryllium reflectors,
    b. If a beryllium reflector is removed from use and the time since its last surveillance exceeds five years, it shall be visually inspected and functionally fit into the core grid box prior to being placed in use, and
    c. If damage is discovered, then the surveillance shall be expanded to include all of the beryllium reflectors prior to use, and annually thereafter.

Bases:

The neutron fluence limit is based on an analysis that was done by the University of Missouri Research Reactor (MURR). In their analysis, they note that the HFIR Reactor has noticed the presence of small cracks at fast fluences of 1.8×10^{22} nvt, and they suggest that "a value of 1×10^{22} nvt (>1MeV) could be used as a conservative lower limit for determining when replacement of a beryllium reflector should be considered." The RINSC limit of 1×10^{22} nvt is even more conservative than what this analysis considers because the RINSC limit is not limited to fast neutron flux.

Reflector elements are visually inspected and functionally fit into the core grid box in order to verify that there are no observable fuel defects or swelling. The rotating inspection schedule ensures that all of the reflectors in the core will be inspected at least once every five years. Since core element handling represents one of the highest risk opportunities for mechanically damaging the fuel cladding, this schedule is deemed appropriate, given the limited amount of information that is gained from these inspections. The discovery of a damaged reflector triggers an increase in the inspection schedule to an annual period.

4.9.b LEU Fuel Elements

Applicability:

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

Objective:

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

Specification:

The fuel elements shall be visually inspected and functionally fit into the core grid box on a rotating basis not to exceed five years such that:

1. The surveillance each year shall include at least one fifth of the fuel elements,
2. The surveillance each year shall include fuel elements that represent a cross section with respect to burn-up,
3. If a fuel element is removed from use and the time since its last surveillance exceeds five years, it shall be visually inspected and functionally fit into the core grid box prior to being placed in use, and
4. If damage is detected by Technical Specification 4.3.3 or otherwise discovered, then the surveillance shall be expanded to include all of the fuel elements prior to use, and annually thereafter.

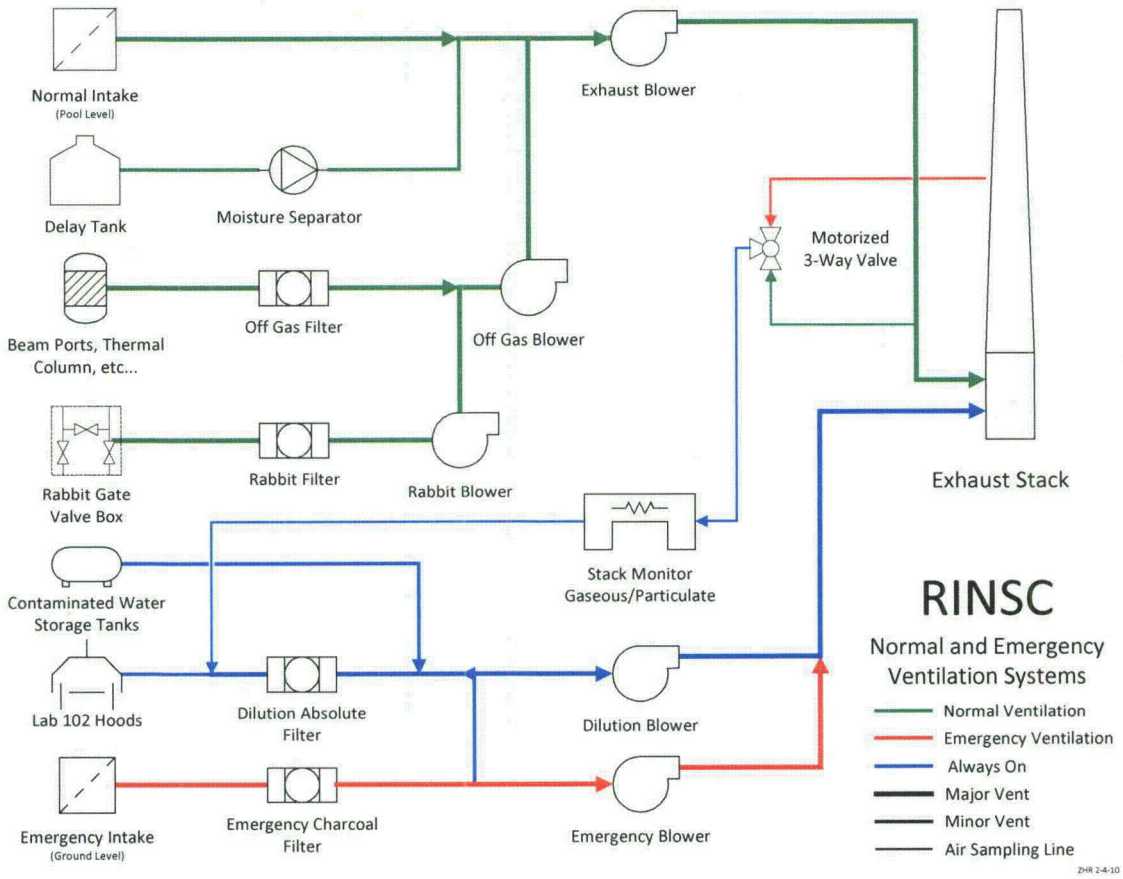
Bases:

RINSC Technical Specification 4.3.3 requires periodic pool water analysis to test for the presence of radioactivity that could potentially indicate a fuel cladding failure. Fuel elements are visually inspected and functionally fit into the core grid box in order to verify that there are no observable fuel defects or swelling. The rotating inspection schedule ensures that all of the fuel elements in the core will be inspected at least once every five years. Since fuel handling represents one of the highest risk opportunities for mechanically damaging the fuel cladding, this schedule is deemed appropriate, given the limited amount of information that is gained from these inspections. The pool water analysis is the most sensitive mechanism for detecting fuel cladding failure. A detected fuel failure triggers an increase in the inspection schedule to an annual period. Fuel inspections include a cross section of elements with respect to burn-up history in order to ensure

that each inspection includes high burn-up elements that would be most likely to start to fail over time.

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

This statement applies to both, normal and emergency operation. Under normal operation, air exits confinement through the normal exhaust intake, goes through the exhaust blower, and enters the base of the stack. The experimental gas systems (off gas and rabbit) tie into the normal exhaust system so that air from these systems go through the exhaust blower and enter the base of the stack. This part of the system is shown in green on the following diagram:



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

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