STATE OF RHODE ISLAND AND PROVIDENCE PLANTATIONS

RHODE ISLAND ATOMIC ENERGY COMMISSION
Rhode Island Nuclear Science Center
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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

August 6, 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 second set of answers to the questions specified in your letter. As requested, copies of the references used to answer these questions have not been included.

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: 8/6/10 By: Michael J. Davis

Docket No. 50-193

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

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Enclosure 1

Second Response to Request for Additional Information Letter Dated April 13, 2010

2.2 Section 2.4.4. Provide a discussion of potential impacts of the RINSC on groundwater, or the lack thereof, including the potential for neutron activation of groundwater, leakage from the reactor pool and primary coolant system, and leakage from contaminated water systems at the facility.

The major potential impact that the facility could have on groundwater arises from the fact that there is a small amount of tritium production in the reactor pool water. If the pool were to have a significant leak, this would be released to the ground or sewer system. The tritium concentration in the pool has been measured to be 3 X 10^{-4} µCi / cc. 10 CFR 20 Appendix B Table 3 indicates that the concentration limit for the release of tritium to the sewer system is 1 X 10^{-2} µCi / cc. Consequently, the concentration in the reactor pool is two orders of magnitude less than the release limit. This indicates that there is no significant potential facility impact on the groundwater.

4.6 Section 4.2.5. Describe the design characteristics of the reactor that ensure the control blades will fully insert despite motion of the core support structure (e.g., shaking of the core due to an earthquake). The response should include tolerances between the control blades and the control blade shrouds that prevent binding of the blades within the shrouds.

The core is suspended from a bridge that is mounted over the top of the reactor pool. General Electric Drawing 198E299 shows how the suspension frame holds the control rod housings and core grid box together. The reactor pool sits on a military gun pad. The pool is constructed of a large mass of reinforced concrete. Consequently, in the event of an earthquake, the pool, bridge, and core are expected to move as a unit. The shim safety control blades fit inside a shroud, which is part of the core grid box. When the shim safety control blades are not fully inserted into the core, each blade is suspended by an electromagnet which holds it in its withdrawn position. When fully withdrawn, the ends of the blades remain inside the shroud, which prevents misalignment on release. A significant earthquake would likely shake the shim safety blades free from the magnets. However, the reactor is fitted with a seismic scram device, which scrams the reactor upon detection of an earth tremor. General Electric Drawing 197E647 shows that the total spacing between the control blades and the shrouds is 0.125 inches.

4.7 Section 4.2.4. Provide a discussion of design features of the neutron startup sources that allow for reliable operation and replacement of the sources. The discussion should include calibrations, source checks, interlocks, and risk of damage to the sources. Include a discussion of any design features and/or administrative controls that reduce the potential for damage to the sources. The discussion should also describe whether improper operation or damage to the sources could potentially lead to instrument error or mislead reactor operators. If the potential exists for damage to the neutron startup sources from operation of the reactor, propose TS requirements to ensure there will be no damage to the sources, or provide justification for not having such TS requirements.

There are three neutron sources that are available for use as a start-up source. The first is a pair of PuBe sources that are stored together in a common container, the second is an SbBe source, and the third consists of the Be reflectors in the core. The reactor Start-Up channel has a neutron count interlock of 3 cps, which is the minimum neutron count rate that must be present in the core in order to start the reactor up. Any one of the available neutron sources may be used as a start-up source, however given the typical reactor operating schedule, the Be reflector elements are generally used as the neutron start-up source. Gamma decay from fission fragments interact with the Be to produce a sufficient level of photo neutrons that the external sources of neutrons are generally not needed in order to have a neutron count rate of at least 3 cps in the core. It is not anticipated that there is risk of damage to the sources. The PuBe sources are leak tested every six months. The Be reflector elements are inspected as part of the fuel element inspection program.

4.20 Section 4.6.2. Provide the uncertainties for the limiting safety system setting (LSSS) values for coolant height and overpower trip. Provide justification for all uncertainty values associated with the LSSS for coolant height, overpower trip, coolant temperature, and coolant flow. (See RAI 14.44)

Pool Level:

A low pool level is determined by the change in state of a float switch. The height of the switch is set so that there is no uncertainty that the switch will change state before the height of the water level above the core is less than 23.7 ft. However, in order to be conservative, an error of 0.5 inches is assumed.

Temperature Measurement:

Inlet, outlet, and bulk pool temperatures at RINSC are measured with an RTD sensor. The technical manual for the meters associated with these sensors indicates that they have an accuracy of + or - 0.5 C. See the reference entitled "Flow and Temperature Meter Specifications".

The normal operating temperature range of the primary coolant is between 90 F and 110 F. Consequently, an error of 0.5 C is:

5/9(90 F – 32) = 32 C

1.5 C / 32 C = 0.015 = 1.5%

Flow Measurement:

Coolant flow rate is measured by looking at the pressure differential across an orifice plate. This differential is transmitted to a meter as a voltage signal. The errors associated with this measurement are:

+ or – 2% of the upper range for the orifice plate (See the reference entitled "Flow Measurement Uncertainty")

+ or – 0.25% accuracy for the transmitter (See the reference entitled "Flow Measurement Uncertainty")

+ or – 0.05% accuracy for the meter voltage input (See the reference entitled "Flow and Temperature Meter Specifications")

Power Level:

From the "Report on the Determination of the Hot Spot Factors for the Rhode Island Nuclear Science Center Research Reactor Using LEU Fuel", Eugene Spring, August 24, 1989 which is used for hot spot analysis, the error associated with power level is 10 %.

4.30 Section 4.7. Clarify whether the correct reference to the figure showing the expanded core configuration is Figure 4-1 or Figure 4-2.

Section 4.7 will be modified to say:

A modification to the standard 14 element core shown in Figure 4-1 was analyzed and approved. The objective was to increase the

neutron flux in the thermal column experimental facility. Several options were considered, including shifting the entire fuel matrix toward the thermal column. However, the RERTR group at Argonne National Laboratory suggested that an increase in the thermal column neutron flux could be obtained by expanding the core into the 17 element core shown in Figure 4-2. This core is achieved by replacing three graphite reflectors on the thermal column side of the core with fuel.

4.31 Section 4.8 – Clarify which figure is meant by "Figure 4" in the text.

The first Paragraph (lines 25-29) will be modified to say:

> Two fresh core models were chosen for the thermal hydraulic calculations that would produce the highest possible fuel cladding temperatures under normal reactor operation at 2 MW reactor power. Figure 4-1 show the standard 14 element core, while Figure 4-2 shows the expanded 17 element core. Both cores were analyzed.

The reference to Figure 4-5 in the third paragraph, line 29 will be changed to "Figure 4-3".

9.3 Figure 9-2 displays the fuel element cut-off saw. This saw is not described in the SAR. Provide a discussion of its use, when it is used, and the design features and controls in place to prevent cutting into the fissile material and control of cutting debris.

References to this saw will be removed from the SAR. This saw is not related to the safety margin associated with the operation of the reactor.

10.3 Section 10.2.4 discusses the thermal column experiment facility. The discussion states that cooling air is required to remove heat generated in the thermal column graphite in order to prevent the graphite from overheating. However, there is no analysis of the flow rate necessary to adequately cool the graphite, and TS 3.2.1 does not provide a set point for the safety channel associated with the thermal column. Provide an analysis of air cooling of the graphite that includes the minimum flow rate necessary to cool the graphite. Explain the basis for the graphite temperature limit of 107 degrees C. (See RAI 14.74)

The temperature limit of 107 °C is cited in the original reactor operating manual [Operation and Maintenance Manual, One-Megawatt Open Pool Reactor for Rhode Island Atomic Energy Commission, Providence, R.I., General Electric Document GEI-77793, October 1962] but no basis is

given. Since the ignition temperature of graphite is well above this temperature (ranging from approximately 400 °C upwards, depending on the specific type and form of the graphite) it is reasonable to assume the limit was placed on the graphite temperature to preclude any unexpected releases of the stored lattice energy (i.e., Wigner energy) induced by neutron irradiation.

Numerous references address the release of stored energy in graphite including:

1. Radiation Defects in Graphite, R. H. Telling, University of Sussex, December 18, 2003;
2. Evaluation of Graphite Safety Issues for the British Production Piles at Windscale: Graphite Sampling in Preparation for the Dismantling of Pile 1 and the Further Safe Storage of Pile 2, B. J. Marsden et al., AEA Technology plc;
3. Nuclear Engineering Handbook, Harold Etherington, ed., McGraw-Hill Book Company, 1958.

As neutron-irradiated graphite is annealed (heated above irradiation temperatures) little or no stored energy is released below approximately 100-125 °C. A significant release peak occurs at approximately 200 °C, so limiting the RINSC thermal column graphite to temperatures below 107 °C provides a margin of nearly a factor of two to this energy release temperature. It is not credible that the entire thermal column contains lattice defects. The threshold energy for the lowest form of induced defect is approximately 1 eV (see Telling paper). After passing through the first several inches of graphite, the neutron flux in the thermal column is, as the name indicates, thermalized to an energy spectrum with a peak near 0.025 eV, which is below the threshold for inducing lattice defects. The figure below shows the energy release rate as a function of temperature (from Ref. 2 above).

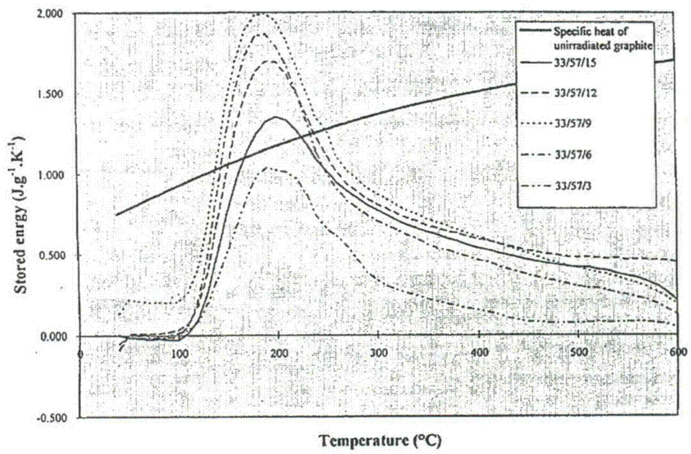

Figure 8. Typical curves for rate of release of stored energy for a number of blocks in Pile 2 channel 33/57 TR. [Evaluation of Graphite Safety Issues for the British Production Piles at Windscale: Graphite Sampling in Preparation for the Dismantling of Pile 1 and the Further Safe Storage of Pile 2, B. J. Marsden et al., AEA Technology plc.]

The only graphite that is exposed to neutrons of energies of 1 eV or greater is the graphite in the first several inches toward the core. This section of the graphite protrudes into the reactor pool via the thermal column extension. Consequently, the cooling for this is provided by the reactor pool rather than by airflow through the thermal column. The purpose of the airflow is to prevent Ar-41 from being released into the reactor room. As a result, no airflow rate specification is needed.

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

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

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

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

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

2.1.1 Safety Limits in the Forced Convection Mode

Applicability:

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

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

Objective:

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

Specifications:

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

Bases:

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

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

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

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

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

2.2.1 Limiting Safety System Settings in the Forced Convection Mode

Applicability:

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

Objective:

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

Specifications:

The limiting safety system settings for reactor thermal power (P), primary coolant flow through the core (m), height of water above

the top of the core (H), and reactor coolant outlet temperature (T_o) shall be as follows:

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

Bases:

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

Measured Parameter	LSSS Value	Measurement Uncertainty	Limiting Trip Value	Safety Limit	Safety Margin
P	2.1 MW	± 10% (± 0.2 MW*)	2.3 MW	2.4 MW	0.1 MW
m	1800 gpm	± 3% (± 60 gpm*)	1740 gpm	1580 gpm	160 gpm
H	23 ft 9.6 in.	± 0.5 in.	23 ft 9.1 in.	23 ft 6.5 in.	2.6 in.
T_o	120 ^oF	± 2% (3 ^oF*)	123 ^oF	125 ^oF	2^oF

*Uncertainties in measured values (± 0.2 MW, ± 60 gpm, 2 ^oF) are based on the nominal operating values of 2 MW, 1950 gpm, and 90 ^oF to 115 ^oF for the power, flow and outlet temperature, respectively.

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

See rewritten TS 2.2.1 in response to 14.36.

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

See rewritten TS 2.2.1 in response to 14.36.

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

See rewritten TS 2.2.1 in response to 14.36.

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

See rewritten TS 2.2.1 in response to 14.36.

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

2.3 MW. Explain this apparent inconsistency between the bases and the specification.

See rewritten TS 2.2.1 in response to 14.36.

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

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

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

See rewritten TS 2.2.1 in response to 14.36.

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

See rewritten TS 2.2.1 in response to 14.36.

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

See rewritten TS 2.2.1 in response to 14.36.

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

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

14.49 The bases for TS 2.2.1 reference a version of the SAR that is different than the version of the SAR submitted with the license renewal application. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

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

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

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

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

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

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

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

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

14.57 The bases for TS 3.1.1 reference a version of the SAR that is different than the version of the SAR submitted with the license renewal application. Revise the proposed TS to refer to the SAR submitted with the license renewal application, as amended.

This reference has to do with predictions that were made about what the shutdown margin would be, prior to when the LEU core was configured, and the shutdown margin was measured. Since this core has been in operation for more than fifteen years, and the shutdown

margin for it has been measured at least annually, this reference is no longer relevant. Consequently it will be removed.

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

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

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

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

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

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

Rewrite these Technical Specifications as follows:

3.1.3 The total reactivity worth of experiments shall not exceed:

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

3.1.4 The maximum reactivity worth of any individual experiment shall not exceed:

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

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

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

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

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

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

Add the following surveillance item:

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

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

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

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

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

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

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

This LCO has to do with the fact that the temperature coefficient must be negative. The statement "surveillance will be conducted at initial startup and change in fuel type" was meant to indicate that the temperature coefficient would be verified to be negative at initial start-up, and if there was a change in fuel type. This was verified during the initial startup with the LEU fuel. Any change in fuel type would require a change in the license. Consequently, this surveillance is no longer necessary. As a result, it will be removed.

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

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

states that the "thermal shield is cooled by water which is currently forced around the shield using the pressure difference between the inlet and outlet primary coolant lines."

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

See also the response to RAI 10.3.

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

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

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

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

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

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

No revision to the Technical Specifications is necessary.

14.80 TS 3.2.1, Table 3.2 contains footnote (b) which states, "The reactor shall not be continuously operated without a minimum of one radiation monitor on the experimental level of the reactor building and one monitor over the reactor pool operating and capable of warning personnel of high radiation levels." Explain what "continuously operated" means. Explain why the radiation monitors subject to footnote (b) do not need to be operating for reactor operations that are not considered "continuous." Explain how each radiation monitor located on the experimental level can individually provide adequate monitoring of the entire experimental level.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

> To assure that the release of airborne radioactive material from the RINSC will not cause the public to receive doses that are greater than the limits established in 10 CFR 20.

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

This limit is based on an analysis that was done by the University of Missouri Research Reactor (MURR). In their analysis, they note that the HFIR Reactor has noticed the presence of small cracks at fast fluences of 1.8×10^{22} nvt, and 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 it is not limited to fast neutron flux. See the reference entitled "Be N Fluence".

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

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

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

Technical Specification 4.2.6 will be changed to say:

The shutdown margin shall be determined in accordance with operating procedures:

Annually,

When a new core is configured,

Following control blade changes.

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

Technical Specification 4.2.7 will be changed to say:

The excess reactivity shall be determined in accordance with operating procedures:

Annually,

When a new core is configured,

Following control blade changes.

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

Technical Specification 4.2.8 will be changed to say:

The excess reactivity shall be determined in accordance with operating procedures:

Annually,

When a new core is configured,

Following control blade changes.

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

The truck door IS the overhead door.