RAI 4.34

The guidance in NUREG-1537, Section 4.5.1, "Normal Operating Conditions," discusses both the limiting core configuration (LCC) and operational core configuration (OCC), and requests the licensee to characterize the neutronic conditions that are applicable. The conditions include power distributions, control rod worth, and core reactivity. The reviewers are considering the material submitted in SAR Chapter 4 to determine whether the LCC models used to support the safety analysis are acceptable. Acceptability of the model is determined by comparing the calculated and measured behavior of the OCC. In SAR Table 4-2, the excess reactivity values for 10 different core configurations are presented. In SAR Table 4-3, several reactivity coefficient calculations are presented and it is stated in the following paragraph that measurements are taken for confirmation.

a. Provide measured excess reactivity values for comparison to the calculated estimates in Table 4-2.
b. Provide a tabulation of control rod worth calculations and measurements for the core conditions described in Table 4-2.
c. Provide confirming measurements for the coefficient calculations in Table 4-3.

d. Describe the core condition that is considered to represent LCC conditions (i.e., the core with the highest power density at full power or at the limited safety system Setting (LSSS) set point (please specify)). For that core, provide the peak fuel assembly power as well as the core and fuel plate peaking factors that are required to support safety analysis.

In 2010, the analyses and material that was in the SAR that was submitted in 2004 was largely abandoned. The reviewers should not be looking at this document for information. None of the people that were involved in preparing that information are still associated with the RINSC facility or Argonne National Laboratory. Consequently, new analyses were done starting in 2010.

RAI 4.34 a Response

Our core configuration is now in the equilibrium configuration with the most efficient neutron reflection. When a refueling operation occurs, four of the most highly burned fuel elements are removed from the center of the core, the remaining elements are shuffled toward the center of the core leaving the four corner grid spaces open, and four fresh elements are positioned in the corner spaces. See "Core Change Summary for Conversion from RINSC LEU Core #5 to LEU Core #6" for the beginning of core data for the current core loading.

RAI 4.34 b Response

See "Core Change Summary for Conversion from RINSC LEU Core #5 to LEU Core #6" for the beginning of core data for the current core loading.

RAI 4.34 c Response

See RAI 4.12 Response for coolant temperature reactivity coefficient and coolant density reactivity coefficient.

See RAI 4.13 Response for temperature and void reactivity coefficient.

RAI 4.34 d Response

At this point, RINSC has two core configurations, both of which have the reflector elements positioned in the most efficient reflector configuration possible. The standard core is a 14 element core. However a 17 element core was also approved by the NRC. The most limiting core is the 14 element core because at full power, 2MW is distributed across only 14 elements, rather than 17 elements. RAI 4.1 Response Figure 4-6 shows the power peaking factors for both cores. RAI 14.67 Response shows the first four configurations of the LEU core which transitioned from the least reflective configuration, to the most reflective configuration.

RAI 4.35

The guidance in NUREG-1537, Section 4.6, "Thermal-Hydraulic Design," states that the licensee should take into account uncertainties in thermal-hydraulic and nuclear parameters. SAR Section 4.6.1 discusses a report titled "Report on the Determination of Hot Spot Factors for the Rhode Island Nuclear Science Research Reactor Using LEU [low-enriched uranium] Fuel (Reference 4-Y)." If this information is used in support of the application, describe where and how it is used, and provide a copy for staff review.

See RAI 4.16 Response.

RAI 4.36

The guidance in NUREG-1537, Section 4.5.3, "Operating Limits," requests an analysis of a rod withdrawal accident. The response to RAI 4.33 (correspondence dated March 15, 2013) provides the requested analysis. The methodology appears to be consistent with current guidance and the appropriate safety limit (SL) (i.e., fuel temperature) is cited as a basis for acceptability; the temperatures attained (~80°C) is suitably below the safety limit (530°C).

a. Confirm that the core configuration used for this analysis is the LCC and the cited reactivity insertion rate (0.02%$\Delta k/k$ per second) establishes the maximum control rod reactivity insertion rate.
b. Propose a technical specification (TS) with the reactivity rate converted into a linear withdrawal rate or provide an explanation why a TS is not required to protect this safety analysis assumption.

RAI 4.36 a Response

This question has to do with the fact that the control rod worth increases as the core becomes more compact, and / or as the core reflection is made more efficient. NRC has approved a 17 element core, as well as a 14 element core. The 14 element core is the loading that is typically in use, and is the core loading that was used for all of the thermal-hydraulic and transient analyses. In October of 2008, core reflection was changed to be the most efficient neutron reflection possible. As required by the facility Technical Specifications, control rod calibrations and reactivity insertion rates are determined annually. The results of those determinations have never shown the reactivity insertion rate to be greater than or equal to 0.02 %Δk/k per second.

RAI 4.36 b Response

Proposed Technical Specifications 160226 includes the following proposed specifications for limiting reactivity insertion rates:

3.1.1.3.1 The total absolute 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 by doing criticality studies:

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

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

Historically, there has never been a linear withdrawal rate. Control rod reactivity varies from year to year, so a linear withdrawal rate would lead to a variable reactivity insertion rate limit. Since it is the reactivity insertion rate that is the important parameter, we intend to keep the specification related to reactivity insertion rate, rather than withdrawal rate.

RAI 4.37

The guidance in NUREG-1537, Section 4.6, "Thermal-Hydraulic Design," requests a thermal-hydraulic (T&H) analysis for the LCC (see RAI 4.34, above). The T&H analysis results are submitted under forced and natural convection conditions. However, the acceptability of the results are quoted in terms of: (1) the power at which the onset of nucleate boiling is predicted to occur; (2) the onset of flow instability; and (3) the onset of critical heat flux. There are no conclusions based upon the SL for RINSC (530°C) or the guidance in NUREG-1537 (departure from nucleate boiling ratio [DNBR] > 2.0). For the LCC operating at the LSSS, document the results of T&H calculations showing compliance with the SL and the cited DNBR guidance. Ensure that the peaking factors determined from the LCC are used. If such analysis is provided using a computer code, provide the supporting documentation demonstrating that the code(s) in question are validated for such purposes and fully describe the model used.

NUREG 1537 is guidance only. There is no requirement to show compliance with the cited DNBR guidance.

Please see the document "Steady State Thermal-Hydraulic Analysis for Forced Convective Flow in the Rhode Island Nuclear Science Center (RINSC) Reactor". Table 4.6-9 shows that onset of nucleate boiling is predicted to occur when the fuel cladding temperature reached 122.6 C, under the condition that the flow rate is 1580 gpm. This is well below the safety limit of 530 C. When the onset of nucleate boiling is reached, the heat transfer coefficient increases due to energy expended in heat of vaporization. Consequently, showing that the operating conditions would never allow onset of nucleate boiling to be reached clearly shows that the safety limit would never be reached, and that there is a significant margin of safety. Figure 4.6-8 shows this nicely.

The computer codes that were used for this analysis were submitted, reviewed, and accepted years ago. Re-analyzing them is beyond the scope of this project at this point. Please see the enclosed letter dated March 15, 2013 indicating that these code manuals and inputs were sent to NRC on that date.

RAI 13.26

The guidance in NUREG-1537, Section 13.2, "Accident Analysis and Determination of Consequences," states that analysis should consider functions and actions assumed to occur that change the course of the accident or mitigate the consequences, such as reactor scram. The rising power transient analysis, in response to RAI 13.7 (correspondence dated September 8, 2010), uses a previously-approved method, but concludes acceptability based upon a trip value of 2.3 megawatt (MW) (the LSSS is 2.1), and cites a 2.4 MW SL, which is not the value in the TS. It is not clear which flow rates and feedback coefficients are used or whether the analysis is suitably conservative. Provide a revised analysis for the LCC using limiting peaking factors, conservative conditions, approved methods, and show that the TS SL (530°C) is not exceeded.

Please see the document entitled "Rhode Island Nuclear Science Center Transient Analyses Revised Jan. 20, 2016 by Arne P. Olson, ANL". These analyses were revised in order to make it possible to set the over power trip to be 115% for both natural convection and forced convection cooling modes of operation. These analyses used the original computer codes that were previously used, with updated inputs to build the desired safety envelope.

RAI 5.2

The guidance in NUREG-1537, Section 5.2, "Primary Coolant System," requests that the licensee describe fully the reactor coolant system. The Safeguards Report for the Rhode Island Open Pool Reactor (1962) identifies a single primary coolant loop. SAR Section 5.2.1.4 identifies two primary coolant loops. The presence of two connected heat exchangers impacts the review for potential loss-of-coolant accidents.

a. Identify the means used for making this change to the facility.
b. Describe all piping and electrical connections to this system.
c. Explain whether the second system constructed has undergone any functional tests and if so, describe how those tests were accomplished.
d. Explain whether the second system has previously been operated, is currently operable, and if so, describe the operational characteristics, performance attributes, the conditions under which this system is used or usable, and modify your submittal for Chapter 5 to incorporate the required documentation.
e. Explain whether both primary loops are assumed to be operating in any of the safety analyses submitted for this review.

The cooling system was upgraded 23 years ago as part of relicensing the facility for the fuel conversion from HEU fuel to LEU fuel. The safety analysis for this change was reviewed and approved as part of that license change. The LOCA analysis was done for the two loop cooling system that is currently in place.

RAI 5.2 a Response

This cooling system was approved by NRC in 1992 as part of the HEU to LEU fuel conversion license change.

RAI 5.2 b Response

Proposed Technical Specification Section 5.2 provides a description of the cooling system.

RAI 5.2 c Response

This system was constructed and put into service long before any of the current employees were associated with the facility. It is unknown whether or not any functional tests were conducted at that time. The system has been in service for 23 years.

RAI 5.2 d Response

The system has been in service for 23 years.

Again, in 2010, the analyses and material that was in the SAR that was submitted in 2004 was largely abandoned. The reviewers should not be looking at this document for information. The entire document will be re-written after the technical details have been defined.

RAI 5.2 e Response

None of the analyses depend on both cooling loops to be operating.

RAI 6.2

The guidance in NUREG-1537, Chapter 6, "Engineered Safety Features," requests that the licensee describe fully any engineered safety feature in use. Section 6.2.3, "Emergency Core Cooling System [ECCS]," identifies specific information that must be included for a system relied upon "... to remove decay heat from the fuel to prevent failure or degradation of the cladding if cooling is lost." The Safeguards Report for the Rhode Island Open Pool Reactor (1962) does not identify an ECCS. SAR Section 1.7.4.1 states that such a system was added as part of the power upgrade program. However, SAR Chapter 6 does not include a description of this system. SAR Section 1.7.4.1 does state that the system was installed. In addition it states, in part, that "the primary pump diaphragm isolation valves were replaced with low d/p [differential/pressure] butterfly valves. As part of the primary flow upgrade, new components with a larger range were added to the reactor flow sensing circuits to monitor the increased flow capability."

a. Identify if the installed system is intended to be an ECCS.
b. Identify the means for making this change to the facility.
c. Describe all piping and electrical connections of this system and how such connections interact with the primary coolant system.
d. Explain whether this system has undergone any functional tests and if so, describe how those tests were accomplished.
e. Describe how use of the system will not degrade any systems relied upon for safety (e.g., what prevents important to safety electrical connections from being wetted?).
f. Explain whether this system is currently operable or has been operated, and if so, describe the operational characteristics, performance attributes, the conditions under which this system is used or usable, and revise Chapter 6 of the SAR to incorporate the required documentation.
g. Explain whether this system has been modeled and/or included for the safety analyses submitted for this review.

In 1992 as part of the fuel conversion from HEU to LEU fuel, the facility upgraded its cooling system. As part of that upgrade, a water line that runs from the facility fire sprinkler system to the top of the high power section of the reactor pool was added. It has two manual ball valves that are locked out to prevent them from being opened inadvertently. It could be used as a last resort for adding town water to the pool. However, none of the safety analyses take credit for the existence of this system. Consequently, when a new SAR is written, this system will not be described.

RAI 13.27

The guidance in NUREG-1537 requests that a spectrum of potential accidents be evaluated to demonstrate acceptable levels of safety for the facility. The accidents evaluated generally fall into two categories: (1) accidents for which dose consequences need to be evaluated; and (2) accidents for which SL compliance needs to be evaluated. In the SAR, the accidents analyzed for SL compliance, the SL of 2.4 MW is consistently used. This is not the SL in the latest version of the RINSC TSs. Provide a comprehensive restatement of your safety analysis for all applicable accidents ensuring that the analysis uses conservative peaking factors, feedback coefficients, and flow conditions. In the case of accidents or transients for which power is an important component, ensure that the LSSS power is used in the evaluation. Demonstrate that the TS SL - 530°C - is conservatively bounded by these analyses.

The SAR will be re-written once the technical details have been worked out. It will reference the safety limit that is based on fuel temperature rather than power level.

RAI 13.28

NUREG-1537 requests that the licensee identify and document the analysis of the maximum hypothetical accident. In the responses to RAIs 13.2 through 13.5 the licensee has provided this analysis. However, the following issues are noted and need to be addressed:

a. The 200-MeV per fission the fission rate is estimated to be ~3.1×10^{10} fissions per watt-sec. The analysis uses 3.1×10^{11} fissions per watt-sec. Revise the analysis with the corrected value and provide the result or explain why your values are correct.
b. Per 10 CFR Part 20, Appendix B, a breathing rate volume under light work condition is 20 liters per minute. The analysis uses 2 liters per minute. Revise the analysis with the corrected value and provide the result or justify the use of your value.
c. It appears that the derived air concentration value of I-130 is incorrectly used for I-131. Revise the analysis with the corrected value and provide the result or justify the use of your value.
d. The analysis does not appear to consider the fuel plate power peaking factors in determining the radionuclide inventory in the damaged fuel plate. Revise the analysis to use the hot plate power density and provide the results or justify your methodology.
e. The atmospheric dispersion calculation refers to an effective x/Q, which includes a factor of 1/3 for the building dispersal effects. This assumption differs from the guidance in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," which recommends

a horizontal plume meander in a building wake. Revise the analysis with the corrected values or provide additional justification for the factors that were used.

f. The noble gas and halogen release fractions are based on the RINSC interpretation of University of Virginia fuel failure results without justification. In general, the inventory of fission gas release depends on the extent of damage, fuel temperature, cladding integrity, and location (under water or in air). Release assumptions should be consistent with the expected conditions of the fuel plate at the time of the accident. Provide a technical justification for the use of this data.

g. The build up of Kr-85 is mainly through the isomeric transition of Kr-85m. Given the yield for the 85 atomic mass (1.317 percent), the Kr-85 yield is about 0.277 percent (1.317% x 0.211); the analysis uses a yield of 0.026 percent. Revise the analysis with the corrected value and provide the result or explain why your value is correct.

h. The reactor building concentration is based on a volume of 6.15×10^9 cm^3, whereas TS 5.1 provides a volume of 5.15×10^9 cm^3. Revise TS 5.1 to define the free volume in the reactor building; if a volume other than the TS 5.1 value is used, provide the supporting justifications and the related calculations. A larger free volume is considered to be less conservative.

i. The analysis uses a charcoal filter efficiency of 0.99 for iodine removal. The basis for this efficiency, given the conditions that can adversely affect adsorption, is not provided. Although TS 3.5.2 does provide a limiting condition for operation requiring the filter for operation, there is no acknowledgment of the potential for filter efficiency degradation over time and the resulting required replacement frequency that should be used to maintain the cited efficiency. Describe the methodology in place at RINSC that will ensure the cited efficiency is maintained.

j. There appears to be transposition errors in the calculations. The dose conversion factor for I-132 from FGR 11 should be 0.381 mrem/μCi (it is cited as 0.011). In addition, the release of Kr-85 to the reactor building is increased by a factor of 3.7 without any explanation. Justify the conversion factor utilized and revise the analysis with the corrected values and provide the results for any updated correction factors.

RAI 13.28 a Response

At some point, the fuel failure analysis was revisited and a document entitled "Fuel Failure Addendum" was produced. An updated version of this that addresses some of the questions associated with this RAI, is attached. In that document, the error associated with the number of fissions per Watt-second was corrected on page 1.

RAI 13.28 b Response

The breathing volume rate is corrected to be 2 X 10^4 cm^3 per minute in the definition of the DAC on page 16 of the new analysis. This is the breathing rate that was used for this analysis.

RAI 13.28 c Response

The new analysis takes the I-131 DAC from 10 CFR 20 Appendix B Table 1 Column 3, which lists it as 2E-8 μCi / cc. This is shown in the first table on page 23 in the new analysis.

RAI 13.28 d Response

The analysis makes the incredible assumption that the fission fragment inventory of an entire fuel plate is released to the pool water. We think that this is a very conservative bounding condition.

RAI 13.28 e Response

The new analysis takes credit for the confinement stack, which makes building wake and horizontal meander effects irrelevant.

RAI 13.28 f Response

The new analysis makes the following assumptions based on NRC Regulatory Guide 1.183:

A. Noble gases are unaffected by the pool water.
B. The pool water retains 99.5% of the radioiodines that are released.
C. The radioiodines are composed of 45% elemental, and 55% organic species.
D. Activity released from the pool to confinement air occurs over a two hour period.
E. All other fission products are retained either in the fuel, or in the pool water.

RAI 13.28 g Response

Fission product yield data was taken from Table 11.4 in:

Lamarsh, John R., *Introduction to Nuclear Engineering*. Massachusetts: Addison-Wesley Publishing Company. 1977.

The double dagger footnote associated with the table indicates that the yields given are equal to the yield of the nuclide plus the cumulative yield of the precursor.

RAI 13.28 h Response

The "Confinement Building" section of the Fuel Failure Addendum on pages 5 and 6 indicate that the free volume of the confinement room was determined to be 5.15×10^9 cc. This is consistent with Proposed Technical Specification 5.1.1.

RAI 13.28 i Response

See RINSC Procedure MP – 02 Emergency Air Filter Efficiency Test Rev. 3 which describes how the iodine efficiency of the charcoal filter is tested. Proposed Technical Specification 4.5.2.4 provides the surveillance interval for this test.

RAI 13.28 j Response

The individual that wrote the Radiological Assessment Attachment retired a number of years ago. Please review the Fuel Failure Addendum.