March 21, 2008

Mr. Glenn Winters, Director
Reactor Critical Facility
Nuclear Engineering and Science Building
Rensselaer Polytechnic Institute
Troy, NY 12181

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE RENSSELAER POLYTECHNIC INSTITUTE (RPI) APPLICATION FOR RENEWAL OF FACILITY LICENSE NO. CX-22 FOR THE RENSSELAER POLYTECHNIC INSTITUTE REACTOR CRITICAL FACILITY (TAC NO. MC9032)

Dear Mr. Winters:

We are continuing our review of your license renewal request for Facility License No. CX-22 for the Rensselaer Polytechnic Institute Reactor Critical Facility submitted November 19, 2002. During our review of your request, questions have arisen for which we require additional information and clarification. As we discussed during our telephone conversation of February 27, 2008, please provide responses to the enclosed requests for additional information no later than June 30, 2008. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation. Following receipt of your response, we will continue our review of your request.

If you have any questions regarding this review, please contact William B. Kennedy at 301-415-2784, or me at 301-415-1631.

Sincerely,

/RA/

Daniel E. Hughes, Project Manager
Research and Test Reactors Branch A
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Docket No. 50-225

Enclosure: As stated
cc w/enclosure: See next page

cc:

Mayor of the City of Schenectady
Schenectady, NY 12305

Barbara Youngberg
Chief, Radiation Section
Division of Hazardous Waste and Radiation Management
NY State Dept. of Environmental Conservation
625 Broadway
Albany, NY 12233-7255

Peter F. Caracappa, Ph.D, CHP
Radiation Safety Officer
NES Building, Room 1-10, MANE Department
Rensselaer Polytechnic Institute
110 8th St.
Troy, NY 12180-3590

Dr. Timothy Trumbull, RCF Supervisor
NES Building, Room 1-10, MANE Department
Rensselaer Polytechnic Institute
110 8th St.
Troy, NY 12180

Peter Collopy, Director EH&S
Rensselaer Polytechnic Institute
21 Union Street
Gurley Building 2nd Floor
Troy, NY 12180

John P. Spath, State Liaison Officer Designee Program Manager
Radioactive Waste Policy and Nuclear Coordination
New York State Energy Research & Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

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Date	3/17/08	3/19/08		

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<u>REQUEST FOR ADDITIONAL INFORMATION</u>

<u>RENSSELAER POLYTECHNIC INSTITUTE REACTOR CRITICAL FACILITY</u>

<u>DOCKET NO. 50-225</u>

Questions related to the Safety Analysis Report

General

The license renewal of a research reactor beyond 40 years requires a re-issuance of the license. This is done in accordance with the regulations by submitting an application for an operating license. The contents of the application are based on the requirements of 10 CFR Parts 50.33, 50.34 and 50.36. Important requirements in 10 CFR 50.34, “Contents of applications; technical information,” include:

> ...analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.
>
> ...information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:
>
> ...description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.
>
> ...description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.
>
> ...such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.

The following requests for additional information (RAIs) are being made to provide the staff with sufficient information to be reasonably assured that the health and safety of the public will be protected during the period of the renewed license.

1. THE FACILITY

1.1 Section 1.3, General Description of the Facility. Figure 1.1 shows an overhead trolley-type crane in the Reactor Room. However, no description is included in the Relicensing Report. Discussions with facility personnel indicate that this crane has not been used for some time, and that power to it is removed at the breaker. In the event this crane is used in the future, describe what inspections will be conducted and what controls will be put in place for movement of loads over vital equipment.

3. DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Sections 3.3, Water Damage, and 2.4, Hydrology. The SAR indicates that the Mohawk River flood stage has exceeded the elevation of the reactor room floor in the last century and has repeatedly exceeded that of the reactor water storage tank pit floor. When considering ground water entering the reactor water storage tank pit or river water entering the reactor room, address the probability of occurrence, safety consequences, access to the building during flooding, and contingency plans that are in place if needed. The issue of concern is that the fuel is maintained in a subcritical configuration and is physically secure.

4. REACTOR DESCRIPTION

4.1 Section 4, Reactor Description. Include a discussion of the auxiliary reactor scram (moderator-reflector water dump) in Chapter 4. What are the criteria for when the moderator dump feature is required and when it can be bypassed?

4.2 Section 4.2.1, Reactor Fuel. This section of the SAR states that the SPERT (F-1) fuel pin design was previously qualified by the DOE and NRC. NUREG-1281, “Evaluation of the Qualification of SPERT Fuel for Use in Non-Power Reactors,” August 1987, is the report on the NRC’s evaluation of the qualification of your fuel. Provide any information that may have a bearing on the conclusions of NUREG-1281 or the suitability of your fuel during the period of the renewed license.

4.3 Section 4.2.2, Control Rods. The SAR states that the control rod drives are designed so rods can be located anywhere in the tank. Clarify whether the intended license basis is restricted to the core arrangement described in the SAR or assumes the use of other control rod configurations. If the latter, provide additional discussion on the design boundaries, safety review process and acceptance criteria for core redesigns. Considering that information, propose Technical Specifications (TS) that ensure configuration control.

4.4 Section 4.2.2, Control Rods (or Section 4.5.2, Reactor Core Physics Parameters). If the control rods can be withdrawn as a gang, verify that the maximum rate of reactivity insertion due to gang control rod withdrawal is bounded by the requirements of TS 3.2.3.

4.5 Section 4.2.4, Neutron Startup Source. Describe the personnel shielding that exists as the neutron source is being withdrawn from the core into the paraffin shield.

4.6 Sections 4.2.5, Core Support Structure, and 4.3, Reactor Tank. Discuss any age-related degradation of the core support structure, the reactor tank, and piping. Discuss any inspections that have been performed on such structures and systems, the results, and any planned actions to correct or manage age-related degradation.

4.7 Section 4.3, Reactor Tank. Discuss the likelihood and consequences of leaks. In the event of a coolant leak from the reactor tank, the storage tank, or the associated piping, what provisions, if any, are there to contain the leak and prevent an uncontrolled release to unrestricted areas, including groundwater? Is the coolant analyzed periodically for radioactivity so that an estimate of any release can be documented? (This question is related to compliance with 10 CFR 20.1501.)

4.8 Section 4.4, Biological Shielding. This section of the SAR indicates that the shielding is adequate for the power of 1 watt. Please indicate typical radiation levels to show that there is adequate shielding at the licensed power of 100 watts. Describe controls used to ensure ALARA during operation (e.g., roof access control during operation).

4.9 Section 4.5.2, Reactor Core Physics Parameters. Section 4.5.2 does not list any core physics parameters. Temperature and void coefficients are found in Tables 13.2 and 13.3. Shutdown margin is only given as a lower bound (> 0.02) in Table 13.2. Please provide quantitative values for excess reactivity and shutdown margin in Chapter 4 and ensure that these values are consistent with the technical specifications. (See SAR RAI 13.5, TS RAI 1.3.V, and TS RAI 3.2 (D))

5. REACTOR COOLANT SYSTEMS

5.1 Discuss water quality requirements and the process used to maintain water quality to minimize corrosion and to assure adequate visibility to safely handle fuel elements.

5.2 Discuss the allowable range of reactor tank water level for reactor operation and the technical basis. (See TS RAI 3.2.6 (B))

5.3 Describe operating procedures, interlocks, alarms, and administrative controls that exist to control the water level in the reactor tank and to assure that there is sufficient free volume in the reactor water storage tank for a reactor tank dump in the event of a scram.

5.4 Discuss the maximum potential level of contamination that could exist in water that collects in the sump and the likelihood and consequence of release to the environment through cracks in the concrete. (See SAR RAI 4.7)

7. INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Section 7.1, Summary Description. The version of the SAR currently under review was submitted to the NRC in November 2002. As discussed in Section 7.1, substantial instrument and control (I&C) equipment upgrades were in progress at that time. To facilitate the current review please provide the following information:

a. A more detailed description of the objective, scope, design, and current status of instrument system upgrade project.

b. Provide enough information such that the staff can evaluate the acceptability of the instrumentation and control presently installed. If this involves digital equipment consider NRC Regulatory Issue Summary 2002-22, “Use of EPRI/NEI Joint Task Force Report, Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI 01-01: A Revision of EPRI TR-102348 To Reflect Changes to the 10 CFR 50.59 Rule.”

7.2 Section 7.2.1, Design Criteria. The information presented in Section 7.2.1 is limited to a brief, general, description of the functions of the I&C systems. Expand this section to describe the criteria (standards, codes, and guidelines) that form the design bases of the I&C systems (Reference: NUREG 1537 Part 1, Format and Content Guide, Section 7.2).

7.3 Section 7.3, Reactor Control System. Provide a more detailed discussion of instruments provided to monitor various reactor system processes and variables. Examples include, control rod position indication, reactor temperature, reactor tank water level, reactor tank water temperature, equipment status indication (e.g., air compressor) and various alarms, such as reactor tank leak alarms (Reference: NUREG 1537 Part 1, Format and Content Guide, Section 7.3).

7.4 Section 7.3, Reactor Control System. Figure 7.1 shows four ion chamber inputs. Section 7.4 states that there are three. Clarify the apparent discrepancy and indicate the location of the detectors relative to the core.

7.5 Section 7.6, Control Console and Display Instruments. Provide a more detailed discussion of the instruments, controls, and indications provided on the main control console.

7.6 Section 7.7, Radiation Monitoring System. An alarm setpoint for the CAM is specified; please relate the setpoint to the radiological impact.

8. ELECTRICAL POWER SYSTEMS

8.1 In general, this chapter does not provide sufficient information to determine the function and design basis of the Normal and Emergency Electrical Power Systems. Provide a more detailed discussion of the design basis and functional description of the normal and emergency electrical systems. The response should ensure that sufficient information is provided to address each of the applicable items listed in Sections 8.1and 8.2 of NUREG 1537, Part 1. Specific requests include:

a. The ranges of electrical power requirements (voltage, current, frequency);

b. From the verbal response during the site visit, it appears that a loss of normal AC power will result in a loss of all lighting in the facility (with no emergency lighting provided), the fire detection system, and the area radiation monitoring systems. If this is the case, a justification should be provided in the SAR to support this design;

c. How instrumentation and control circuits are protected from electromagnetic interference that may be generated by the electrical power system;

9. AUXILIARY SYSTEMS

9.1 Section 9.2, Handling and Storage of Reactor Fuel. This section references a constraint from the design basis for the fuel vault which places a limit of 15 fuel pins per tube in the vault. Section 1.3 states that the vault has short tubes for the former fuel design and long tubes for the current fuel design. Is it possible to place the current fuel in the short tubes or more than 15 fuel pins in a tube? If either or both are possible, please discuss the consequences of such an accident.

9.2 Section 9.2, Handling and Storage of Reactor Fuel. SAR Section 9.2 and TS Section 5.6 describe the storage of spent fuel and the surveillance requirements and frequency for fuel inventory. Is this surveillance a TS requirement, and if not, justify why it is not?

10. EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 You may not have experimental facilities such as those listed in NUREG 1537 (Part 1), Section 10.2, however, the second paragraph on page 10-2, concerning critical facilities, is applicable. In addition, the paragraph just before Section 10.1 and other parts of this chapter of NUREG 1537 (Part 1), including the Appendices 10.1 and 10.2, concerning experimental utilization are also applicable.

From the list of experiments described in Section 1.6 of the SAR it would appear that experiments performed at the RCF are limited to the measurement of reactor characteristics (rod position measurements, subcritical multiplication measurements, etc.). However, during the site visit reactor use was characterized as "...used for demonstrations about 95% of the time. Periodically gold foils are activated, but only to a level that does not require the foils to be placed in lead pigs for transport. (Core flux is not high enough to highly activate the foils). All experiments are reviewed by NSRB."

Discuss your experimental program, including information which more fully describes the types of experiments performed and the facilities, apparatus or equipment used to perform them. In addition, describe the process for experiment approval and oversight. Please use the references mentioned above for guidance.

10.2 Section 10 states that new experiments that raise a USQ will be reviewed by the NSRB. Note that under 10 CFR Part 50.59 this will require a license amendment. Please resubmit wording for the referenced paragraph incorporating the current wording of 10 CFR Part 50.59.

11. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Section 11.1.5, Radiation Exposure and Dosimetry. Discuss typical dose rates throughout the RCF during reactor operation, fuel handling operations and shutdown so as to give a perspective of the radiation environment.

11.2 Section 11.1.5, Radiation Exposure and Dosimetry. Please provide information indicating that the radiation levels at the site boundary are within the regulatory limits during and after reactor operation.

11.3 Section 11.1.7, Environmental Monitoring. It is stated that 5 mrem/yr has been measured at site boundary and 15 mrem/yr at the exclusion area boundary above that measured at the GE facility more than 1.6 km away. The staff is reading this as: 5 mrem/yr above background at the site boundary and 15 mrem/yr above background at the exclusion area boundary with the background taken at the General Electric Company Guard Station. During the site visit it was stated that this was old information and that recent results reported in the annual effluent report for the RCF indicate no detectible radiation at either the site boundary or the exclusion area boundary. First, if there is more recent and accurate environmental monitoring data available, please provide an update for section 11.1.7 of your SAR; otherwise discuss how you satisfy the requirements of 10 CFR 20.1101(d) and verify that you meet the requirements of 10 CFR 20.1301(a)(2). Second, clarify the discrepancy between the above statements and TS 5.1 and TS 5.2 which indicate that the exclusion area boundary and the site boundary are both defined by the outer fence surrounding the reactor building.

12. CONDUCT OF OPERATIONS (INCLUDES TS SECTION 6, ADMINISTRATIVE CONTROLS)

10 CFR 50.36 contains the regulations for technical specifications. 10 CFR 50.36(b) states that the TS will be derived from the analysis and evaluation included in the safety analysis report. However, SAR Section 12 is quite brief and in many sections just refers to the TS, which is reverse from the intent of the regulations. Please resubmit Section 12 of your SAR, addressing each of the issues identified and questions raised in the following RAIs.

12.1 SAR Section 12.1, Organization, and TS Section 6.1, Organization. NUREG-1537 and ANSI/ANS-15.1-1990, "The Development of Technical Specifications for Research Reactors," provide guidance on the organizational structure. The guidance notes that there should be a multi-level organization chart in the SAR and a description of the relationships with the line organization. The SAR and TS contain such descriptions and charts, however, the charts and titles are not completely consistent, e.g., it appears that the operations supervisor, reactor supervisor, and supervisor of critical facility and radiation safety officer may be one and the same. Please clarify and make terms agree between the SAR and TS descriptions and the Figures.

12.2 SAR Section 12.1, Organization, and TS Section 6.1, Organization. ANSI/ANS-15.1-1990, defines the responsibilities of the Level 1 Management position as responsible for the reactor facility's licenses or charter (i.e., Unit or Organizational Head). Verify that the RCP Director has authority and responsibility and speaks for RPI in all matters concerning License CX-22. As an example, decommissioning funding is required by 10 CFR 50.75 (e)(1) and typically the Level 1 Manager has authority to provide the financial assurance required by the regulations.

12.3 SAR Section 12.1, Organization, and TS Section 6.1, Organization. The organization illustrated in Figure 12.1 of the SAR is different than that in Figure A1 of the TS. Please resolve those differences and justify the structure.

12.4 SAR Section 12.1, Organization, and TS Section 6.1, Organization. TS 6.2 states that the Nuclear Safety Review Board (NSRB) advises the Facility Director, TS 6.2.2 (a) states that the Chairman of the NSRB is approved by the Facility Director, and the SAR Section 12.2 has NSRB audit reports going to the Facility Director, whereas SAR Figure 12.1 and TS Figure 6.1 show the NSRB reporting to the Operations Supervisor. Clarify the relationship between the Facility Director and the NSRB such that independence of the review and audit function of the NSRB is assured. The ANSI/ANS-15.1- 1990 and NUREG-1537 provide guidance that may be helpful.

12.5 Section 12.1.3, Staffing, and TS Section 6.1.3, Staffing. ANSI/ANS -15.1-1990 provides definitions of reactor secured and reactor shutdown. The TS provide similar definitions for "reactor shutdown," (TS 1.3.O), and "secured shutdown" (TS 1.3.U). The TS only specify the minimum staffing when the reactor is not shut down. Thus, the TS do not specify the required staffing when the reactor is shut down, but not secured shut down. Propose a TS that specifies the minimum staff required when the reactor is shut down, but not secured shutdown.

12.6 Section 12.1.3, Staffing, and TS Section 6.1.3, Staffing. ANS-15.1-1990 recommends for the SRO to be capable of getting to the reactor facility within a reasonable time (e.g., 30 minutes). The proposed TS 1.3.P defines "Readily Available on Call," used in TS 6.1.3(a) (3) as within 30 miles or 60 minutes. The existing TS 1.3.P defines, "Readily Available on Call," as 15 miles or 30 minutes. Please justify that 60 minutes is an acceptable response time for the SRO readily available on call.

12.7 Section 12.1.4, Selection and Training of Personnel, and TS Section 6.1.4, Selection and Training of Personnel. The TS cites ANSI/ANS 15.4-1977 rather than the more recent version, 1988. Please update this reference if possible, otherwise discuss the reason for not updating it.

12.8 Section 12.1.4, Selection and Training of Personnel, and TS Section 6.1.4, Selection and Training of Personnel. Discuss how your training program meets the requirements of 10 CFR Part 19.

12.9 Section 12.1.5, Radiation Safety. 10 CFR 20.1101 requires that each licensee shall develop, document, and implement a radiation protection program. The NRC staff must have adequate information about your radiation protection program to be reasonably assured that it meets the requirements of 10 CFR 20. NUREG-1537 and Section 6.3 of ANS-15.1-1990 recommend a TS on Radiation Safety and ANSI/ANS-15.11-1993, "Radiation Protection at Research Reactor Facilities," provides guidance. Currently the brief descriptions of the radiation safety organization in the SAR and TS are not coordinated and do not use the same terms. The Radiation Safety function is not included on Figure A.1 of the TS. In Figure 12.1 of the SAR, the Director, Office of Radiation and Nuclear Safety is connected to 5 levels of the organization without any description of chain of command, reporting, coordination, etc. The health physicist of the TS is not mentioned in the SAR, so it is not clear where the person resides within the organization. There is no commitment to ANSI/ANS-15.11 or mention of an ALARA program in the TS. In addition, there is no discussion of how and when the radiation safety staff communicates with the facility manager and Level 1 management to resolve safety issues.

10 CFR 20.1101(b) requires an ALARA program. Who is responsible for the ALARA and radiation safety programs? When the RCF is in use, is there a person responsible for radiation safety present at the facility or on call? If on call, does the SRO have sufficient training in radiation safety to perform those duties until assistance arrives? SAR Section 12.1.5 suggests that RCF staff have responsibility for radiation safety and the campus support is only available for occasional assistance.

Please add discussion in the SAR to address the above questions and issues. As appropriate, propose TSs and supporting bases that reference discussion in the SAR.

12.10 Section 12.2, Review and Audit Activities, and TS Section 6.2.1, Composition and Qualification. ANS-15.1-1990 states that members and alternates of the review/audit committee should be appointed by and report to Level 1 management, the level above the individual responsible for facility operation. NUREG-1537 states that members should be appointed by the highest level of upper management. However, this does not appear to be the case in the SAR, the TS, and the Organizational Charts, Figures 12.1 of the SAR and A.1 of the TS. Please discuss and provide assurance that the NSRB is independent of the direct management of the facility. Propose TSs and SAR bases as necessary to address this issue (see also 12.2 above).

12.11 Section 12.2, Review and Audit Activities, and TS 6.2.3, Review and Approval Function. NUREG-1537 suggests that the TS should explicitly state that the NSRB addresses the review function of 10 CFR 50.59. Neither the SAR nor TS explicitly mentions this function. During the site visit, the licensee noted that the NSRB does perform this review function. Discuss how the 10 CFR 50.59 process is implemented at RCF.

12.12 Section 12.2, Review and Audit Activities, and TS 6.2.4, Audit Function. NUREG-1537 and ANSI/ANS-15.1-1990 note areas that should be addressed by the Audit Function. Items listed there, but not explicitly included in the SAR or TS audit function, are: TS conformance, the physical security plan, requalification training program, emergency plan, and radiation protection program. Provide assurance that these areas are part of the audit function of the NSRB.

12.13 Section 12.3, Procedures and TS Section 6.3, Procedures. ANSI/ANS-15.1-1990 lists the activities that should be addressed by written procedures. Provide justification for not including personnel radiation protection procedures, including ALARA during normal operations per ANSI/ANS 15.11-1993, and administrative controls for operations and experiments in TS 6.3.

12.14 Section 12.3, Procedures, and TS Section 6.3, Procedures. 10 CFR 50.36(c)(5) requires administrative TS. NUREG-1537 and ANSI/ANS-15.1-1990, Section 6.4, provides guidance for meeting the requirements for review and approval of procedures. The SAR and TS do not discuss how facility operations and management prepare, review, and approve the procedures. Discuss your review and approval process to provide assurance that there is adequate independence.

12.15 TS Section 6.4, Experiment Review and Approval. Reword this TS utilizing the terminology of the present version of 10 CFR 50.59.

12.16 TS Section 6.4, Experiment Review and Approval. Regulatory Guide 2.2, “Development of Technical Specifications for experiments in Research Reactors,” 1973, provides guidance in meeting the requirements of 10 CFR 50.34(b)(4) and 10 CFR 50.36(b) with respect to the experimental program. Provide adequate discussion and propose TSs as necessary to allow the NRC staff to assess the risk to the health and safety of the public from the operation of your facility.

12.17 Section 12.4, Required Actions, and TS 6.5, Required Actions. The regulation 10 CFR 50.36(c)(5) requires administrative TS. ANSI/ANS-15.1-1990 provides guidance and the RPI TS define actions to be taken in case of a reportable occurrence; the actions include "reactor conditions shall be returned to normal or the reactor shall be shut down." NUREG-1537 Chapter 14, App. 14.1, Section 6.6.2 states that the TS should establish in advance specific criteria for the two alternative actions; return to normal and shutdown (an example is given in the reference). Discuss the criteria used at RCF and propose TS changes as necessary.

12.18 Section 12.5, Reports, and TS 6.6.1, Operating Reports. The applicable regulations include 10 CFR 50.36(c)(5)&(7). ANSI/ANS-15.1-1990 Section 6.7.1 suggests a list of those items for inclusion in the annual operating report. The TS includes these with the exception of major preventive maintenance and a summary of exposures over 25% of allowable for visitors. ANSI/ANS-15.1-1990 also calls for a summary of environmental surveys performed outside the facility, but the TS only lists TLD dose rate readings. Are there other environmental results that should be included? Also TS 6.6.1(a)(5) and (e) correctly cited 10 CFR 50.59, but “(a)” and “(b)” respectively, should be dropped from the 10 CFR 50.59 citations.

12.19 Section 12.5, Reports, and TS 6.6.2, Non-Routine Reports. ANSI/ANS-15.1-1990, Section 6.7.2, specifies a 30-day report for permanent changes in the Level 1 or 2 facility organization, but the TS include this as an annual report. 10 CFR 50.36(c)(7) and the guidance in NUREG-1537 Chapter 14, App. 14.1, Section 6.7.2, Special Reports, states that the telephone reports should be made to the NRC Operations Center and the regional staff. Written reports fall under 10 CFR 50.36(c)(5) and should be submitted as specified in 10 CFR 50.4. Propose TS to require a 30-day report notifying the NRC of permanent changes in the Level 1 or 2 facility organization. Propose changes to the TS so that all written reports are submitted as specified in the first paragraph of TS 6.6.

12.20 Section 12.6, Records, and TS 6.7, Operating Records. The applicable regulation includes 10 CFR 50.36(c)(5). The following records specified in ANSI/ANS-15.1-1990 should be added to the TS listing: fuel receipts (5 years), approved changes to operating procedures (5 years), NSRB audit reports (5 years), training records of certified operations personnel (one certification cycle), radiation exposure for visitors (life of facility).

13. ACCIDENT ANALYSIS

13.1 Section 13.1.5, Mishandling or Malfunction of Fuel. Section 4.5 states that "... removing multiple fuel pins from the interior sections of the core can result in significant reactivity addition, beyond the excess reactivity limit of 60 cents set in the Technical Specifications." Please provide justification in Section 13.1.5 to support the statement that mechanical rearrangement of the fuel to obtain a supercritical configuration inadvertently or with intent, is not a credible occurrence.

13.2 Section 13.2, Accident Analysis and Determination of Consequences. In Table 13.1, the ratios bi / beff appear to have come from G. R. Keepin, "Physics of Nuclear Kinetics," 1965, but the value of beff = 0.00765 is different from the b given in Keepin's book. Please explain how beff was determined.

13.3 Section 13.2, Accident Analysis and Determination of Consequences. Section 13.2 is the SAR information which supports TS 3.1 and 3.2. While the TS appear reasonable, they are not fully supported by, or consistent with the SAR. The analysis is done at 20°C while the TS minimum allowable temperature is 50°F (10°C) where there is a greater positive reactivity coefficient. Is the 20°C calculation a worst case for other reasons? Since there is no interlock on temperature, please discuss why the initial temperature used in the analysis should not be less than the 50°F TS limit.

13.4 Section 13.2, Accident Analysis and Determination of Consequences. Section 13.2 begins by stating the reactor was operating at 200 watts at the start of the scenario. Later it says that Table 13.1 lists nuclear characteristics used in the analysis but is inconsistent in that it states the power to be 100 watts. Please clarify.

13.5 Section 13.2, Accident Analysis and Determination of Consequences. SAR Table 13.2 lists a column of TS values which in some cases differ from those used in the TS. For example, a Shutdown Margin of >0.02 (2.6 $) is stated in Table 13.2; the limit stated in TS 3.2.2 is 0.7 $. The limiting reactivity worth of a standard fuel assembly is listed to be <0.039 (5 $); TS 3.2.1 specifies a maximum of 0.20 $. In the first example, the TS value is less conservative than the value listed in the table and therefore, appears not to meet the basis in the SAR however, the problem appears to be with the terminology. The value listed in the table as Shutdown Margin appears to be Shutdown Reactivity as defined in TS 1.3 V. The limit in TS 3.2.2 is consistent with the table value of "Reactivity with One Stuck Rod," however, the table does not specify that the stuck rod is the most reactive rod. In addition, the accepted definition for "Shutdown Margin" (see the definition in Section 1.3 of ANSI/ANS-15.1-1990) is not specified in the TS. In the second example the TS is more conservative but the disparity is so large the basis is questionable. Please provide more discussion about Table 13.2 including the source of the values and their relationship to the TS. Clarify the confusion with “Shutdown Margin” and “Shutdown Reactivity” by providing a definition of the former in the TSs and correcting Table 13.2 to be consistent with the definitions. (See TS RAI 1.3.V)

13.6 Section 13.0, Accident Analysis, Figures 13.2 and 13.3. Notes on Figures 13.2 and 13.3 infer that the analysis was done for a 421-424 pin core with an 0.585 inch pitch whereas SAR Chapter 4 describes a 329 to 333 pin core with a 0.64 inch pitch core lattice plate. However, Chapter 4 states that other approved lattice plates exist. In the SAR and possibly the TS clarify what constitutes an “approved” lattice, the approval process and why the safety analysis presented envelopes other lattices.

13.7 Section 13.0, Accident Analysis. Please address the following editorial observations:

a. Near the end of Table 13.3 it states that the temperature coefficient is negative when T< 16 or T< 32 for core A and B respectively. This does not appear to be correct or consistent with Table 13.2 and Figures 13.2 and 13.3.

b. On Figures 13.2 and 13.3 the temperature coefficient shows a positive exponent ($\times 10^5$) which is not consistent with the equation. Please correct.

c. On Figure 13.3 the final exponent (-4) of the equation is missing, possibly a photocopying artifact. Please correct.

Questions related to the Technical Specifications

General

In accordance with 10 CFR 50.36, provide proposed Technical Specifications (TSs). The proposed TSs should be in conformance with ANSI/ANS-15.1-1990, “American National Standard for The Development of Technical Specifications for Research Reactors,” as appropriate. The standard provides valuable guidance in the development of the TSs such that they meet the requirements of 10 CFR 50.36. Each individual change in the proposed TSs from the current TSs incorporated in Facility License No. CX-22 (current TSs) should be cited. Substantive changes should be justified with analysis or discussion, as appropriate. In addition, each TS editorial change should be described in your response. Change citations and the accompanying justifications and descriptions should not appear in the proposed TSs. The proposed TSs shall be reviewed and approved by the Nuclear Safety Review Board in accordance with the Administrative Controls required by the current TS 6.1.5.3, “Review and Approval Function.”

Pursuant to 10 CFR 50.36(b), the technical specifications will be derived from the analyses and evaluations included in the safety analysis report (SAR). Many of the following RAIs request you to provide reference to analysis in the SAR as basis justification of the TSs. This may be accomplished by referencing analysis already contained in the SAR, providing replacement SAR pages that contain the analysis, or by providing a separate analysis, discussion, and/or reference. In the latter case, the staff may incorporate that response in its Safety Evaluation Report by reference, and you may provide replacement pages for your SAR at a later time.

Pursuant to 10 CFR 50.36(a), summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the proposed specifications, but shall not become part of the technical specifications.

Pursuant to 10 CFR 50.36(b), the Commission may include such additional TS as the Commission finds appropriate, and the approved TSs and any additional TSs will be incorporated into the renewed license.

The following is a list of specific sections of the proposed TSs submitted as Appendix A of the "RPI Reactor Critical Facility Relicensing Report," with your application dated November 19, 2002, that require clarification or additional information.

General

The proposed TSs should be included as a separate attachment to your response to this letter.

The proposed TSs should not have the heading, "RPI Reactor Critical Facility Relicensing Report, 12/2002," that appears on each page of the proposed TSs submitted November 19, 2002.

The proposed TSs should have a title page and table of contents similar to those contained in the current TSs.

1.2 Format

1.2 Section 1.2 of the proposed TS references ANSI/ANS 15.1. Update this reference to include the appropriate revision date and ensure that all references to ANSI/ANS 15.1 that appear in the TSs are to the same revision of the standard. (See TS RAI 1.3.X)

1.3 Definitions

1.3 The terms "known core" and "unknown or untested core" appear in the TSs, but are not defined. Provide definitions of these terms. (See TS RAI 4.1 (A))

1.3.D The definition in the current TSs contains references to EuO_3 in a stainless steel cermet, stainless steel, and an alloy of silver-cadmium-indium as possible materials for the control rod absorber sections. The proposed definition does not reference these materials. Confirm that these materials will not be utilized for the control rod absorber sections.

1.3.O (A) The definition of reactor shutdown is circular in that it contains the phrase, "reactor is shutdown by at least 1.00$." Revise the definition to eliminate the circularity.

1.3.O (B) The definition does not account for all possible states of the reactor. For example, if the core contains 50% of the fuel pins required for criticality and a control rod is manually withdrawn (e.g., for maintenance or testing), the reactor is neither secured, nor shutdown, nor operating. Explain any formal controls in place to preclude the reactor being in an undefined state, or revise this definition to eliminate the possibility that the reactor could be in an undefined state.

1.3.O (C) Consider adding a separate definition for "Reactor Operating," instead of including it in the definition of "Reactor Shutdown."

1.3.P The proposed definition specifies the maximum permissible distance and travel time for the Licensed Senior Operator (LSO) on call as 30 miles or 60 minutes. The current definition specifies the maximum permissible distance and travel time for the LSO on call as 15 miles or

30 minutes. Provide justification for the increases in the permissible distance and travel time. (See SAR RAI 12.6)

1.3.T Provide justification that the restraining forces that hold the fuel pins in the reactor core will be adequate to restrain any secured experiment. Alternately, revise the requirements for the magnitudes of restraining forces needed to ensure that secured experiments will not become unsecured during normal operation and credible accidents.

1.3.V The SAR and TSs refer to both shutdown reactivity and shutdown margin as though the two terms are interchangeable. Provide a definition of one of the terms and use that term consistently throughout the SAR and TSs. (See SAR RAI 13.5 and TS RAI 3.2 (D))

1.3.X The definition references standard ANSI/ANS 15.1 (1982). Ensure that revision of the standard is the revision referenced throughout the TSs. (See TS RAI 1.2)

2.0 Safety Limits and Limiting Safety System Settings

2.1 (A) The SAR contains no discussion of the technical basis for the safety limit. Provide discussion and analysis of the technical basis for the safety limit.

2.1 (B) 10 CFR 50.36c(1)(i)(A) requires safety 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.” TS 2.1 does not adequately address protection of the fuel cladding integrity. Provide analysis that shows that no material degradation of the fuel cladding will occur if the fuel pellet temperature is limited to 2000°C. Otherwise, revise the safety limit and provide analysis or discussion that shows the new safety limit will reasonably protect the integrity of the fuel and the cladding.

2.1 (C) The reference to W.A. Duckworth, ed., “Physical Properties of Uranium Dioxide,” <u>Uranium Dioxide: Properties and Nuclear Applications</u> (Washington, D.C.: Naval Reactors, Division of Reactor Development), 1961, pp.173-228, that appears in the current TS does not appear in the proposed TS. Provide a reference to this document or reference to analysis in the SAR that supports the basis for TS 2.1.

2.2 (A) 10 CFR 50.36c(1)(ii)(A) requires that the limiting safety system setting must be so chosen that automatic protective action will correct any abnormal situation before a safety limit is exceeded. Provide reference to analysis in the SAR that demonstrates the limiting safety system settings for reactor power and reactor period will not result in the safety limit being exceeded.

2.2 (B) The bases of TS 2.2 refer to “energy deposition,” “enthalpy rise,” and “power increase,” whereas the safety limit is specified on the fuel pellet temperature. Provide reference to analysis in the SAR that relates the three above-mentioned terms to fuel pellet temperature, or revise the bases of TS 2.2 to use temperature-related terminology with reference to supporting analysis in the SAR.

2.2 (C) Given that TS 3.2.7 requires a minimum of 2 counts per second on the start-up channel and TS 3.2.9 requires an interlock blocking rod withdrawal when neutron flux is less than 2 counts per second, remove the limiting safety system setting for minimum flux level and the associated basis from TS 2.2.

3.0 Limiting Conditions for Operation

3.1 Section 13.2 of the SAR lists the initial temperature of the reactor coolant as 20°C. 10 CFR 50.36c(2)(ii)(B) requires a technical specification limiting condition for operation (LCO) on “a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis...” Accordingly, propose a technical specification for the maximum reactor coolant temperature. Include a basis that references analysis contained in the SAR.

3.1.2 Given that TS 3.1.3 allows reactor operation at temperatures 50°F and above, set limits on the void coefficient of reactivity in the temperature range from 50°F to 100°F, or provide justification for not doing so.

3.2 (A) Update the reference to the “Hazards Summary Report” to reflect the current safety analysis document.

3.2 (B) Footnote (a) to Table 1 indicates that the “Log Count Rate” safety channel may be bypassed when linear power channels are reading greater than 3×10^{-10} amps. Provide the count rate or power level that corresponds to 3×10^{-10} amps.

3.2 (C) Update Table 2 of proposed TS 3.2 to reflect Amendment No. 11 to Facility License CX-22 dated September 7, 2004, which approved removal of the interlock, “Failure of 400 Cycle Synchro Power Supply.”

3.2 (D) Table 13.2 of the SAR lists the value of the shutdown margin used in the accident analysis as >0.02. 10 CFR 50.36c(2)(ii)(B) requires a LCO on “a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis...” Accordingly, propose a technical specification for the shutdown margin. Include a basis that references analysis contained in the SAR. (See SAR RAI 13.5 and TS RAI 1.3.V)

3.2.1 (A) Provide discussion and/or analysis in the SAR of the technical bases for the core excess reactivity and the maximum reactivity worth of a clean fuel pin. Provide reference to that discussion and/or analysis in the bases for TS 3.2.1

3.2.1 (B) Table 13.2 of the SAR gives the reactivity worth of a standard fuel assembly as <0.039, which does not appear to be consistent with the maximum reactivity worth of 0.20$ specified by proposed TS 3.2.1. Explain the apparent discrepancy or update Table 13.2 of the SAR to be consistent with the proposed TS 3.2.1. (See SAR RAI 13.5)

3.2.3 Provide discussion and/or analysis in the SAR of the technical bases for the maximum control rod reactivity rate. Provide reference to that discussion and/or analysis in the bases for TS 3.2.3

3.2.4 Clarify whether the magnet release time of 50 milliseconds includes the safety system response time, i.e., the time required for interruption of power to a magnet once a measured value reaches the safety system setting. If not, revise TS 3.1.4 to include the safety system response time and provide reference to appropriate analysis in the SAR.

3.2.5 The basis for TS 3.2.5 states that “the requirement that negative reactivity be introduced in less than one minute following activation of the scram is established to minimize the consequences of any potential power transients.” The SAR does not mention any power transients the consequences of which would be minimized by the auxiliary reactor scram (moderator-reflector water dump), nor does the SAR explain the technical basis for the requirement that negative reactivity be added within one minute of activation of the auxiliary scram. As written, the SAR provides inadequate justification for considering the auxiliary reactor scram a safety feature, and therefore, according to 10 CFR 50.36c(2)(i), a LCO should not be placed on the auxiliary scram. Provide discussion and analysis in the SAR of the technical basis for the safety function of the auxiliary reactor scram, including quantitative analysis of the requirement that negative reactivity be added within one minute of its activation, or remove TS 3.2.5 and its associated bases from TS 3.2. In addition, modify TS 3.2.8 as appropriate.

3.2.6 (A) The current TS 3.2 contains a basis for TS 3.2.6 that states, “the normal moderator-reflector water level is established not greater than 10 inches above the top grid of the core...” The proposed TS 3.2 does not contain a basis for proposed TS 3.2.6. Provide reference to analysis or discussion in the SAR of the technical basis for establishing the moderator-reflector water level not greater than 10 inches above the top grid of the core.

3.2.6 (B) Justify not specifying a limit on the minimum moderator-reflector water level, or include a LCO on minimum moderator-reflector water level and an associated basis with appropriate reference to discussion or analysis in the SAR.

3.2.8 See TS RAI 3.2.5.

3.2.9 (A) Table 2 provides insufficient information about the interlocks that prevent rod withdrawal. Include the appropriate symbols (i.e., <, >, and/or =) for “Reactor Period 15 sec” and “Neutron Flux 2 cps,” such that the interlocks are consistent with the analysis in the SAR Include the failure condition or conditions for “Failure of Line Voltage to Recorders” (e.g., line voltage less than “x” volts).

3.2.9 (B) Table 2 of the current TS 3.2 specifies the interlock “Water Level in Reactor Tank 10 ± 1” Above Core Top Grid.” Table 2 of the proposed TS 3.2 does not specify that interlock. Provide justification for not including that interlock in the proposed TS 3.2.

3.2.10 (A) Sections 1.2, 3.1, 4.1, 4.4, and Table 4.1 of the SAR make references to an administratively-imposed maximum thermal power level of 15 watts and other operating thermal power levels below 100 watts. Confirm that the safety conclusions presented in the SAR do not take credit for a power level less than 100 watts as specified by TS 3.2.10.

3.2.10 (B) Provide a basis for the specification that integrated thermal power for any consecutive 365 days shall not exceed 200 kilowatt-hours. Provide reference to analysis in the SAR that supports the basis.

3.3.1 TS 3.3.1.c uses the phrase, “whenever the reactor is to be operated.” This phrase is not defined in the TS and appears redundant to the general applicability of TS 3.3. Reword TS 3.3.1.c to clarify whether particulate monitoring is required whenever the reactor is not secured, or whenever the reactor is not secured and not shut down. (See TS RAI 1.3.O (C))

3.3.2 Include the minimum inventory and types of portable survey instruments required by TS 3.3.2, or provide justification for not including this information in TS 3.3.2.

3.4 The bases for TS 3.4.8 and TS 3.4.9 contain outdated references to 10 CFR 20.101, 10 CFR 20.103, 10 CFR 20.105, and 10 CFR 20.106. Update these references.

3.4.3 TS 3.4 does not contain a basis for the reactivity worth or allowed frequency of moveable experiment which may be oscillated in the core. Provide a basis for TS 3.4.3 that references analysis in the SAR.

3.4.5 TS 3.4.5 appears contradictory to the requirements of TS 3.4.8 and TS 3.4.9 regarding materials that may produce airborne radioactivity. Clarify the intent of TS 3.4.5 as it applies to experiments that are not encapsulated, singly-encapsulated experiments, and doubly-encapsulated experiments.

3.4.8 (A) The exposure time for persons in unrestricted areas (2 hours) must be consistent with the ability and any plans RPI has in place to control occupancy of unrestricted areas, i.e., public evacuation plans and procedures. If RPI does not have approved plans and procedures for controlling occupancy in unrestricted areas, the exposure time for persons in unrestricted areas should be based on the maximum possible exposure time for a release from the particular experiment and the reactor building (e.g., plume passage time). Provide justification of the use of a 2-hour exposure time, or revise the TS to account for the maximum possible exposure time.

3.4.8 (B) Provide a discussion of the method used to ensure compliance with the requirements of TS 3.4.8. Include the methods and assumptions used to calculate doses to persons in the restricted area and unrestricted area.

3.4.9 (A) See TS RAI 3.4.8 (A)

3.4.9 (B) See TS RAI 3.4.8 (B)

4. Surveillance Requirements

4.0 Specify surveillance methods, requirements, and acceptance criteria to ensure monitoring of the fuel integrity and preclude the use of damaged (e.g., corroded, bowed, leaky, etc.) fuel pins. Include a basis that references or summarizes discussion in the SAR. Otherwise, provide justification for not requiring surveillance of the fuel pins.

4.1 (A) TS 4.1 refers to an “unknown or previously untested core.” The proposed TSs do not provide a definition or the characteristics of an unknown or untested core. Provide a definition of an unknown or untested core. Revise the basis for TS 4.1 to summarize or reference discussion or analysis in the SAR that addresses the specific qualitative and/or quantitative characteristics that differentiate an unknown or untested core from a known core. (See TS RAI 1.3)

4.1 (B) The basis for TS 4.1 refers to the initial test period of the reactor. Provide clarification as to whether the initial test period of the reactor is the initial test period for any unknown or untested core and revise the basis for TS 4.1 as appropriate.

4.1 (C) Provide reference to analysis or discussion in the SAR that describes the methods used to determine the reactor parameters specified in TS 4.1 during the initial testing of an unknown or untested core. Include a discussion of safety precautions and controls.

4.1.a The proposed TS contains the word “back,” while the current TS contains the word “bank.” Clarify.

4.1.d See SAR RAI 13.5, TS RAI 1.3.V, and TS RAI 3.2 (D)

4.2 (A) 10 CFR 50.36c(3) requires surveillance requirements “to assure that the necessary quality of systems and components is maintained, that facility operations will be within safety limits, and that the limiting conditions for operation will be met.” TS 4.2 does not specify surveillance requirements to support each technical specification in TS 3.2, specifically TS 3.2.1, TS 3.2.2, and TS 3.2.3. Propose appropriate surveillance requirements to verify each LCO in TS 3.2, or justify omitting surveillance requirements.

4.2 (B) The first paragraph of the bases states, “past performance of control rods and control rod drives and the moderator-reflector water fill and dump valve system have demonstrated that testing semiannually is adequate to ensure compliance with Specification 3.2, Items 3, 4, and 5.” Please clarify which surveillance requirement specified by TS 4.2 ensures compliance with TS 3.2, Item 3.

4.2 (C) The second paragraph of the bases states, “redundancy of all safety channels is provided...” Table 1 of TS 3.2 requires a minimum of 1 “log count rate” safety channel and 1 “log-N; period” safety channel. Clarify the apparent discrepancy.

4.2.5 Verify that the reference to TS 3.2.5 should not be TS 3.2.6

4.3 Propose surveillance requirements for the portable detection and survey instruments specified in TS 3.3.2, or justify omitting surveillance requirements.

5. Design Features

5.4.1 The first paragraph of this section states that the reactor tank is a stainless steel lined tank. Section 4.3 of the SAR states that the reactor tank is stainless steel. Clarify the apparent discrepancy.

5.4.2 This section describes the core as consisting of all SPERT (F-1) fuel, or approximately half of SPERT (F-1) fuel with the remainder being low enriched uranium light water reactor type fuel of typical power reactor design and arrangement. If the intent is to maintain this capability, provide additional information on this latter fuel, such as design parameters, qualification, and operating limits. Describe any special handling or storage considerations.

5.4.4 The first sentence of the second paragraph of this section contains a typographical error in that the word “on” appears in the proposed TS where the word “one” appears in the current TS. Clarify.

5.6 (A) The core loading specifications described in this section are LCOs and should be properly formatted and placed in the appropriate sections of TS 3. Otherwise, justify not making such a change.

5.6 (B) Item 4 of the proposed TS contains a typographical error in the word “one” appears in the proposed TS where the word “on” appears in the current TS. Clarify.

6. Administrative Controls

RAIs pertaining to TS 6 are incorporated in the RAIs covering Section 12 of the SAR.