Rensselaer

DEPARTMENT OF MECHANICAL,
AEROSPACE, AND NUCLEAR ENGINEERING

RCF 10-03
October 28, 2010

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

Re: Response to Request for Additional Information

Dear Sir:

This letter provides the information requested by your letter dated June 22, 2010, "Rensselaer Polytechnic Institute, Request for Additional Information Regarding the Renewal of Facility Operating License (TAC No. ME1591)."

The requested information is provided in two parts; the first part contains the responses to the detailed technical questions and the second contains the revised Technical Specifications.

The Nuclear Safety Review Board has reviewed and approved the revised Technical Specifications.

Sincerely,

Dr. Timothy H. Trumbull, Director
L. David Walthousen Laboratory
RPI Reactor Critical Facility

I declare under penalty of perjury that the foregoing is true and correct.
Executed on:

10/28/10
Date

Dr. David V. Rosowsky, Dean of Engineering

Cc:
Dr. Yaron Danon, Chairman
RPI NSRB

Dr. Tim Wei, Chairman
MANE

Dr. Peter Caracappa
RPI Radiation Safety Officer
Environmental Health and Safety Office

Mr. Jason Thompson, Operations Supervisor
RPI RCF

A020
NRR

Rensselaer Polytechnic Institute
110 8th Street | Troy, NY 12180-3590 | Jonsson Engineering Center
Phone (518) 276-6351 | Fax (518) 276-6025 | www.rpi.edu/dept/meaem

Response to Detailed Questions

October 27, 2010

11. The definition of core configuration states, "The core configuration includes the number, type, or arrangement of fuel..." Clarify whether the word "or" should be changed to the word "and".

    Response: "Or" changed to "and".

12. The definition of core configuration contains a reference to "reflector elements." The reactor, as described in the SAR, does not appear to use reflector elements. Explain the reason for including reflector elements in the definition, and revise the proposed TS as appropriate.

    Response: Addition of reflector elements is possible but would fall under experiment. "Reflector elements" removed.

13. The definition of excess reactivity states, "...from the point where the reactor is exactly critical (k_{eff} = 1) at reference core conditions..." The definition of reference core condition states, "The condition of the core when it is at ambient temperature (cold) and control rods are on the bottom." These definitions do not appear to be consistent because the definition of excess reactivity states that the reactor is critical and the definition of reference core condition states that the rods are on the bottom. Explain this apparent discrepancy and revise the proposed TS as appropriate.

    Response: The definition of excess reactivity was changed to "Excess reactivity is that amount of reactivity that would exist if all reactivity control devices and movable experiments were moved to the maximum reactive condition from the point where the reactor is exactly critical (k_{eff} = 1)."

14. The first definition of reactor secured states, "...control rods are inserted..." Clarify whether this refers to all control rods, and revise the proposed TS as appropriate.

    Response: This should refer to all control rods. Definition changed to "...all control rods...".

15. The "Applicability" section of proposed TS 2.2 appears to be an incomplete sentence. Revise the section as appropriate.

    Response: Reworded to "These specifications apply to the settings that initiate protective action for instruments monitoring parameters associated with the reactor power limits and rate of power level changes."

16. The bases for proposed TS 2.2 states, "Power increase and energy deposition subsequent to scram initiation are thereby limited to well below the identified safety limit." As written, this statement implies that there are safety limits on reactor power and energy deposition. Ensure that this statement is consistent with every proposed safety

limit (question 8 of the RAI transmitted by letter dated May 5, 2010, requested proposed safety limits for the RCF).

Response: Sentence changed to "Power and maximum fuel pellet temperature increase subsequent ..."

17. The current TS 3.2, "Reactor Parameters," contains limits on the temperature and void coefficients of reactivity and minimum operating temperature. The proposed TS omit these requirements based on a justification that 1) excess reactivity defines the magnitude of the worst-case reactivity accident, and 2) reactivity coefficients do not contribute to the accident analyzed in the SAR because the accident does not lead to significant temperature change or void formation. This justification does not include consideration of accidents that involve positive reactivity coefficients for temperature increase (for example, heating of the moderator using the installed heaters) or void formation (for example, displacement of the moderator by foreign objects). Unless the proposed TS require negative temperature and void coefficients, the justification that temperature increase and voids can't contribute to an accident is invalid. Revise the proposed TS to include the current TS 3.2 (the TS may be renumbered, but the technical content should remain unchanged), or justify not including such requirements. Justification should include analyses and discussions that show that the reactivity accident analysis in the SAR bounds all accidents that could result from positive reactivity coefficients for any core that could be configured within the requirements of the proposed TS.

    Response: Section 3.1 in the proposed TS has been changed to include specifications and bases for the temperature coefficient, void coefficient, and minimum temperature required for operation.

18. Proposed TS 3.2.3 states, "The total control rod drop time for each control rod..." The term "control rod drop time" is not defined in the proposed TS. Clarify whether this term is synonymous with the term "scram time" defined in the proposed TS, and revise the proposed TS and the associated bases as appropriate.

    Response: Terms are synonymous. "Total control rod drop time" was changed to "scram time".

19. Proposed TS 3.2.6 specifies, "The minimum safety channels that shall be operating during the reactor operation." Clarify whether the term "operating" should be replaced with the term "operable," and revise the proposed TS as appropriate. (Also, it appears that the proposed TS contains an extra word "the.")

    Response: TS 3.2.6 changed to "The minimum safety channels that shall be operable during reactor operation are listed in Table 1."

20. Proposed TS 3.2.6 specifies safety channels required during reactor operation. Explain the reason for not requiring any safety channels during reactor evolutions other than reactor operation, e.g., fuel movement, and revise the proposed TS as appropriate.

    Response: TS 3.2.7 added stating, "Startup channels must be operational during facility evolutions (including but not limited to fuel movement, control rod movement,

experimental apparatus insertion) and audible indication must be present in the reactor room." insuring that all personnel in reactor room have indication.

21. Proposed TS 3.2.7 states, "...all control rods are verified to have scrammed..." Explain what the term "scrammed" means, and revise the proposed TS as appropriate.

    Response: "scrammed" changed to "fully inserted".

22. Proposed TS 3.2.7 states, "...and it is deemed wise to retain the moderator shielding in the reactor tank." Given that this statement involves a judgment by the senior reactor operator and gives the reason for closing the moderator dump valve, it seems more appropriate as part of the basis for allowing the senior operator to close the moderator dump valve. Explain the reason for making this statement part of the proposed TS, and revise the proposed TS as appropriate.

    Response: 3.2.8 changed to "After a scram, the moderator dump valve may be re-closed by the SRO on duty if the cause of the scram is known, all control rods are verified to have fully inserted, and the reactor is decreasing in power." Also a paragraph reading "If the shielding provided by the moderator is desired (or for any other reason) the moderator may be retained after a scram. Before the auxiliary scram may be bypassed the rods should be verified to have been fully inserted and the reactor must be decreasing in power." was added to the bases section.

23. Proposed TS 3.2.7 allows the senior operator to close the moderator dump valve following a scram, but does not require the senior operator to verify that the scram has had the desired effect on the reactor, i.e., that the reactor is subcritical. The moderator dump is a backup shutdown mechanism that provides redundancy to the control rods. Explain the reason for not requiring the senior reactor operator to verify that the control rods have inserted sufficient negative reactivity to shut down the reactor prior to closing the moderator dump valve, and revise the proposed TS as appropriate.

    Response: Added condition in TS3.2.8 that the reactor must also be decreasing in power.

24. Proposed TS 3.2.8 specifies interlocks required to be operable during reactor operation. Four of the interlocks prevent control rod withdrawal unless certain conditions are met. It seems that the intent of the "neutron flux" interlock and the "line voltage to recorders" interlock is to ensure that the reactor instrumentation is operable prior to initial withdrawal of the reactor control rods. Given that the reactor is not operating until a control rod is moved from the "bottomed" position (a condition in the definition of reactor operating), explain the reason for not requiring these interlocks prior to operation of the reactor, and revise the proposed TS as appropriate.

    Response: TS 3.2.9 changed to "The interlocks that shall be operable while the rods are not fully inserted are listed in Table 2."

25. Other than the reporting requirement in proposed TS 6.7.1(b), the proposed TS do not appear to contain a surveillance requirement related to the integrated thermal power limit of 2 kilowatts in any consecutive 365 days specified in proposed TS 3.2.9. The requirement in proposed TS 6.7.1(b) is an annual requirement, whereas proposed TS 3.2.9 limits thermal power generation over any consecutive 365-day period. Explain

the reason for not including a specific surveillance requirement to ensure that reactor operation will be in accordance with proposed TS 3.2.9, and revise the proposed TS as appropriate.

Response: 4.2.7 added requiring a quarterly surveillance of integral power: "Integral power shall be tallied quarterly as long as no three consecutive quarters exceed 1.5 kW."

26. Table 1 of proposed TS 3.2 contains a column labeled "Functions." The first entry in this column is "Minimum Flux Level." Explain the function associated with the minimum flux level and revise the proposed TS as appropriate.

    Response: Function changed to "Minimum Flux Level Interlock (see Table 2)".

27. Table 1 of proposed TS 3.2 contains 3 safety channels labeled, "Manual Scram," "Building Power," and "Reactor Door Scram." The proposed TS require annual calibration of these channels, but do not require channel tests to ensure they are operable (other than the channel test included in the calibration) as recommended in ANSI/ANS-15.1, Section 4.2(5)(a). Explain the reason for not requiring channel tests of these safety channels. (See question 42.)

    Response: Checks are preformed as part of our start-up procedure. TS 4.2.3 was changed to "All safety system channels shall be calibrated annually and check daily prior to reactor operation."

28. Footnote (b) of Table 1 of proposed TS 3.2 states, "...provided that no other scram channels are bypassed." The proposed TS do not appear to allow bypassing of any other safety channels. Explain this apparent inconsistency, and revise the proposed TS as appropriate.

    Response: No other safety channels can be bypassed. Foot note changed to "... permission of the Operations Supervisor."

29. Table 2 of proposed TS 3.2 contains interlocks labeled "Reactor Period <15 sec," Neutron Flux <2 cps," and "Line Voltage to Recorders < 100 V." It appears that the less-than symbols are incorrect. Clarify whether the symbols should be greater-than symbols (>). Clarify whether the word "on" should be changed to "off" in the interlock labeled "Moderator-Reflector Water Fill On," and revise the proposed TS as appropriate.

    Response: 4 of the 5 interlock descriptions were wrong. All three < were changed to > and the last interlock was changed to "Moderator-Reflector Water Fill "Off"".

30. The "Applicability" section of proposed TS 3.7 states, "...requirements for reactor operation." The proposed TS contains requirements for both reactor operation and fuel handling. Revise the "Applicability" section and/or the "Specification" section to be consistent.

    Response: Application changed to "These specifications apply to the minimum radiation monitoring requirements for reactor operations and fuel handling."

31. The "Objective" section of proposed TS 3.7 states, "... preclude undetected radiation hazards or uncontrolled release of radioactive material." Explain how the radiation monitoring requirements preclude the uncontrolled release of radioactive material.

Response: TS 3.7 changed to read, "The purpose of these specifications is to ensure that adequate monitoring is available to preclude undetected radiation hazards to facility personnel and the public."

32. Proposed TS 3.7.1.a requires a criticality detector system that monitors the fuel storage area during reactor operation. Given that fuel can be moved in and out of the fuel storage area when the reactor is not operating, explain the reason that the criticality detector is only required during reactor operation, and revise the proposed TS as appropriate.

Reason was oversight. TS 3.7 was changed so that item 3, now the only item concerned with the criticality detector now reads "A criticality detector system that monitors the main fuel storage area is required at all times except while the fuel vault is locked and maintenance on this system is being performed. This system shall have a visible and an audible alarm in the control room. This system may the same as the area gamma monitor required by 3.7.1a(2)."

33. Proposed TS 3.7.2 states, "During normal operation..." Clarify whether the term "normal operation" is synonymous with reactor operation.

Response: "Normal" removed.

34. Proposed TS 3.7.2 and proposed TS 3.7.3 both state that certain radiation monitors "*will* be available" during certain situations. Revise the proposed TS to use terminology consistent with the definitions in the proposed TS, i.e., "*shall* be available."

Response: This was changed in the proposed TS before RPI received these RAI's and was already removed.

35. The bases for proposed TS 3.7 contain a reference to particulate monitoring of the reactor room air, but the proposed TS do not contain any requirements for particulate air monitoring. Explain this apparent inconsistency, and revise the proposed TS as appropriate.

Response: TS 3.7.4 was added stating, "A continuous air monitor that draws air from near the surface of the reactor tank is required to be operating while the reactor is operating."

36. ANSI/ANS-15.1, Section 3.8.3 recommends that experiments shall be designed such that they will not contribute to the failure of other experiments or the fuel cladding and that reactor transients will not cause experiments to fail in ways that could contribute to an accident. Explain the reason for not including these design requirements in the proposed TS, and revise the proposed TS as appropriate.

Response: New section 3.8.3 was added to the TS reading "No credible experiment failure shall interfere with another experiment or affect fuel cladding."

37. Proposed TS 3.8.5 and 3.8.8 specify that the failure of a singly-encapsulated experiment shall not result in doses in excess of the regulatory limits for occupational personnel or members of the public. The SAR does not contain an analysis of this type of an accident. In accordance with 10 CFR 50.36(b), provide an analysis in the SAR, including all assumptions, of this type of an accident. (See NUREG-1537, Part 1, Chapter 13 for more information.)

Response: This condition (now numbered 3.8.10) has been revised to specify a maximum activity that if instantaneously and uniformly mixed with the reactor room air would result in a concentration of radioactive material that would not exceed limits specified in Appendix B of 10 CFR Part 20.

The volume of the reactor room is taken to be 1019 m^3 (30 feet x 30 feet x 40 feet).

The activity in the room as a function of time is given by:

$$A = A_0 e^{-(Q/V)t}$$

where Q is the flow rate and V is the volume of the room. The concentration of radionuclide in the room would be given by A/V. By assuming uniform mixing, the concentration of the effluent will necessarily be equal to the concentration in the reactor room. The most limiting condition would occur when the reactor room air is discharged as slowly as possible, but the lower bound is a discharge rate of 1.2 m^3/hour, as this is equivalent to the breathing rate of a standard person. If the discharge rate were any slower, the intake by a member of the public would necessarily be mixed with fresh air.

According to Appendix B of 10 CFR Part 20, the concentration values specified in Table 2, Column 1 of the Appendix represent the radionuclide concentration which, if inhaled continuously over the course of a year, would result in a total effective dose equivalent of 0.05 rem (50 mrem, or half the annual dose limit for members of the public). This takes no credit for dilution of the material between the release point and the receptor point.

Integrating the above equation over a year with a constant discharge rate of 1.2 m^3/hour, the average effluent concentration is found to be equal to 0.097 times the initial effluent (and reactor room) concentration. Rounding this ratio to 0.1, an initial concentration 10 times the Appendix B, Table 2, Column 1 value will result in an average annual effluent concentration equal to the Appendix B, Table 2, Column 1 value. An additional factor of one-fifth is added to bring the maximum dose consequence to a member of the public from 50 mrem to 10 mrem.

At any flow rate greater than 1.2 m^3/hour, the average annual effluent concentration will be less, and therefore the dose consequences smaller.

Exhaust from the reactor room is achieved only through the stack effect. The flow rate from the stack due to the stack effect is given by:

$$Q = C\,A\,\sqrt{2\,g\,h\,\frac{T_i - T_o}{T_i}}$$

where:

Q	= stack effect draft/draught flow rate, m^3/s
A	= flow area, m^2
C	= discharge coefficient (usually taken to be from 0.65 to 0.70)
g	= gravitational acceleration, 9.81 m/s^2
h	= height or distance, m
T_i	= average inside temperature, K
T_o	= outside air temperature, K

Assuming a flow area of 0.051 m^2 (10 inch diameter stack), a stack height of 15.24 m (50 feet), and a flow rate equal to 1.2 m^3/hour, the temperature difference (Ti – To) necessary to achieve that flow rate is 0.007 to 0.008 degrees K for an indoor temperature between 0 and 40 degrees C (32 and 104 degrees F). As the outside temperature will vary much more rapidly than the inside air temperature, it can be assumed that any period of time where the flow rate does not exceed 1.2 m^3/hour will be fleeting at most.

Although the above equation is specified only for conditions where the inside air temperature exceeds the outside air temperature, when the reverse is true, the same equation holds true (substituting To – Ti), except that the air is forced from the outside into the reactor room, forcing the air out of the reactor room at ground level. As no credit is taken for the height of the release, a negative temperature differential in excess of 0.008 degrees K will still result in sufficient discharge rate of reactor room air with no different dose consequences to members of the public.

The SAR will be updated to reflect this calculation methodology.

38. The sixth paragraph of the bases for proposed TS 3.8 states, "...no experiment will be performed with materials that could... produce airborne activity..." This statement appears to be inconsistent with the requirements in proposed TS 3.8.5 and 3.8.8. Explain this apparent inconsistency, and revise the proposed TS as appropriate.

    Response: Phrase "produce airborne activity in excess of the limits of TS §3.8.8" removed

39. Proposed TS 3.9 states, "...and shall monitor the operation with appropriate radiation instrumentation." Explain what "the operation" means. Explain whether this requirement is specifically related to fuel transfers described in the preceding requirement in proposed TS 3.9, and revise the proposed TS as appropriate.

    Response: "the operation" changed to "all operations and evolutions".

40. Proposed TS 4.1.c requires that the "reactor power measurement" be determined during testing of an unknown core. Clarify whether determining the reactor power measurement is the same as calibrating the "Linear Power" and "Log-N; Period" safety

channels. If not, explain how this surveillance requirement ensures that known cores will satisfy the limiting conditions for operation for the nuclear instrumentation.

Response: TS 4.1 changed to read "... reactor power instrument calibration; and ..."

41. ANSI/ANS-15.1, Section 4.2 recommends measuring control rod drop time following work done on the rod or rod drive system. Explain the reason for not including such a requirement in proposed TS 4.2.1, and revise the proposed TS as appropriate.

    Response: Requirement 4.2.6 added stating that "In addition to the scheduled surveillances, any system shall be tested to prove operability after all modification, maintenance, or repairs."

42. ANSI/ANS-15.1, Section 4.2 recommends performing operability tests of reactor control and safety systems following modifications or repairs. Explain the reason for not including any such requirements in proposed TS 4.2, and revise the proposed TS as appropriate.

    Response: 4.2.7 added stating "In addition to the scheduled surveillances, any system shall be tested after all modification, maintenance, or repairs."

43. Proposed TS 4.2.3 requires all safety system channels to be calibrated annually. Clarify whether the intent is to include the "Manual Scram," "Building Power," and "Reactor Door Scram" safety system channels in the calibration requirement. If not, revise the proposed TS as appropriate.

    Response: 4.2.3 now reads, "The start-up and power safety channels shall be calibrated annually. In addition all safety system channels shall be check daily prior to reactor operation."

44. Proposed TS 4.2.4 requires daily channel tests of the intermediate and power range instruments. Explain the reason for not requiring channel tests of all safety channels required by Table 1 of proposed TS 3.2, and revise the proposed TS as appropriate.

    Response: "(intermediate and power ranges)" removed leaving "...the safety system channels...". All are now included.

45. Proposed TS 4.2.4 requires "checks" and "rechecks" of the interlock system. Explain what "checks" and "rechecks" mean in terms of the surveillance activities defined in the "Definitions" section of the proposed TS (e.g., channel check, channel test, or channel calibration), and revise the proposed TS as appropriate.

    Response: Check changed to test, rechecked changed to retested.

46. Proposed TS 4.2.6 provides circumstances under which "tests" may be waived. The use of the word "tests" makes it unclear which surveillances may be waived. Clarify whether the meaning of the word "tests" includes all of the surveillance activities required by proposed TS 4.2, or if it only includes surveillance activities specifically referred to as tests (or channel tests). Revise the proposed TS as appropriate.

Response: TS 4.2.8 reworded to "Requirements 1 thru 6 may be waived when the instrument, component, or system is not required to be operable, but the instrument, component or system shall be tested prior to being declared operable. If a system is not required to be operable it also does not need to be in calibration but must be calibrated before it is declared operable if the calibration is out of date."

47. The bases for proposed TS 4.2 do not appear to contain a basis for proposed TS 4.2.6. In accordance with 10 CFR 50.36(a)(1), provide a basis for proposed TS 4.2.6.

    Response: New paragraph added reading, "If components are not needed, such as during prolong secured periods, components are not used and are not required to be operable. Before reactor operations can resume all requirements must be meet including all safety system channels and verified as operable."

48. The "Objective" section of proposed TS 4.3 states, "...ensure the continued validity of radiation protection standards in the facility." Explain the meaning of this statement as it applies to the reactor coolant system.

    Response: TS 4.3 now reads "No coolant system exists." To be consistent with SAR.

49. Proposed TS 5.3 gives the weight percent of uranium enrichment in the fuel as 4.81 weight percent or less. The current TS specifies the weight percent of uranium enrichment in the fuel as 4.8 weight percent. Explain the reason for the change in the specified uranium enrichment.

    Response: A mass spec. analysis was performed and the enrichment was reported to be 4.8074%. The value of the enrichment has been rounded to 4.8% and all references in the TS changed to be consistent.

50. The last sentence of the first paragraph of proposed TS 5.3 states, "The core shall consist of all SPERT (F-1) fuel described in 5.4.3." It appears the reference to "5.4.3" is a holdover from a previous version of the proposed TS. Provide the correct reference, and revise the proposed TS as appropriate.

    Response: TS 5.3 *Specifications*, paragraph one modified to read, "The reactor core shall consist of uranium fuel in the form of 4.8 weight percent or less enriched UO_2 pellets in metal cladding, arranged in roughly a cylindrical fashion with four control rods placed symmetrically about the core periphery. The total core configuration and the arrangement of individual fuel pins, including any experiment, shall comply with the requirements of these Technical Specifications found in Sections 3.1 and 3.2 of this license.. Core fuel pins to be utilized are 4.8 weight percent enriched SPERT (F-1) fuel rods. Each fuel rod is made up of sintered UO_2 pellets, encased in a stainless steel tube, capped on both ends with a stainless steel cap and held in place with a chromium nickel spring. Gas gaps to accommodate fuel expansion are also provided at both the upper end and around the fuel pellets. NUREG-1281 describes these fuel pins in additional detail."

51. Proposed TS 5.3 references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed TS will become part of the TS and license. Clarify whether the intent is to make the referenced section of the SAR a requirement in the proposed TS. If not, revise the proposed TS as appropriate.

Response: TS 5.3 revised to remove references to the SAR.

52. Proposed TS 5.4 specifies limits for storage tubes located in the fuel storage area in terms of the total weight of fuel and total number of fuel pins that may be stored in any tube. It appears that the weight limit is inconsistent with the limit on the number of pins per tube. Explain this apparent inconsistency, and revise the proposed TS as appropriate. Explain how the fuel storage tube limit is consistent with the requirement that the infinite multiplication factor is less than 0.9.

    Response: Weight limit replaced with "15 SPERT (F-1) fuel pins".

53. Proposed TS 5.4 references the SAR. Any portion of the SAR referenced in the "Specification" section of the proposed TS will become part of the TS and license. Clarify whether the intent is to make the referenced section of the SAR a requirement in the proposed TS. If not, revise the proposed TS as appropriate.

    Response: No references to the SAR are in the proposed TS 5.4.

54. Figure 6.1 of proposed TS 6.1 does not indicate any communication lines as recommended in Figure 1 of ANSI/ANS-15.1. Proposed TS 6.1.2 and 6.2 describe communication lines between the Facility Director and the Nuclear Safety Review Board (NSRB) and the RPI Radiation Safety Officer. Revise Figure 6.1 to include communication lines, or justify not including such requirements.

    Response: A new Figure 6.1 was drawn showing reporting and communication lines.

55. Figure 6.1 of proposed TS 6.1 does not include Level 3 and Level 4 of the organizational structure. Revise Figure 6.1 to include Level 3 and Level 4 of the organizational structure, including appropriate reporting and communication lines, or justify not including such requirements.

    Response: A new Figure 6.1 was drawn showing reporting and communication lines.

56. Proposed TS 6.1.1 specifies that the Facility Director (Level 2) reports to the Chair, Mechanical, Aerospace, Nuclear Engineering for administrative purposes. The Chair, Mechanical, Aerospace, Nuclear Engineering is not assigned a level in the management structure, and the responsibilities of this position are unclear in terms of interactions with the Facility Director, the NSRB, and the Dean, School of Engineering (Level 1). Additionally, proposed TS 6.1.2 does not specify the responsibility of the Chair, Mechanical, Aerospace, Nuclear Engineering in terms of facility safety. Is the Chair, Mechanical, Aerospace, Nuclear Engineering considered part of a level (e.g., Level 1 or Level 2) in the organizational structure? Explain the function and responsibilities of the Chair, Mechanical, Aerospace, Nuclear Engineering in terms of the management organizational structure and facility safety. Revise the proposed TS as appropriate.

    Response: Chain of command organization has been clarified. Chair of MANE department removed, description of levels now only includes the position, description of responsibilities and lines of communication are explained elsewhere.

57. ANSI/ANS-15.1, Section 6.1 and Section 6.1.2 recommend the TS contain information on functions, assignments, and responsibilities of key organization staff. Revise

proposed TS 6.1.2 to include this information for the Chair, Mechanical, Aerospace, Nuclear Engineering and Level 4 personnel, or justify not including such requirements.

Response: All references to Chair, MANE were removed. Responsibility for Level 4 personnel added to TS 6.1, *Responsibility*.

58. ANSI/ANS-15.1, Section 6.1.2 recommends that the TS contain information that individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and technical specifications. Revise the proposed TS to include similar requirements, or justify not including such requirements.

    Response: A paragraph has been added to TS 6.1, *Responsibility*, "Personnel at the various management levels, in addition to the duties and responsibilities outlines above, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and technical specifications."

59. ANSI/ANS-15.1, Section 6.1.2 recommends that in all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications. Revise the proposed TS to include similar requirements, or justify not including such requirements.

    Response: Added requirements that delegate must have the appropriate qualifications.

60. Proposed TS 6.1.3(a)(2) states, "...certified by the Reactor Supervisor as qualified..." The term "Reactor Supervisor" is not defined in the proposed TS. Revise proposed TS 6.1.3(a)(2) to use defined terminology.

    Response: "Reactor Supervisor" changed to "SRO on duty."

61. Proposed TS 6.1.3(b) states, "...but not in safe shutdown is a..." The term "safe shutdown" is not defined in the "Definitions" section of the proposed TS. Revise proposed TS 6.1.3(b) to use defined terminology, or define the term "safe shutdown" in the proposed TS.

    Response: "safe shutdown" changed to "secured."

62. Proposed TS 6.1.4 states that years spent in baccalaureate or graduate study may be substituted for operating experience when meeting the minimum requirements for the Operations Supervisor position. Explain what fields of study are acceptable to substitute for operating experience, and revise the proposed TS as appropriate.

    Response: Nuclear engineering or US Navy Nuclear Power School specified.

63. Proposed TS 6.1.4 does not provide explicit qualification requirements for Level 2 of the organizational structure. Clarify whether the reference to ANSI/ANS-15.4 covers the qualification requirements of Level 2 facility management (i.e., the minimum requirements in ANSI/ANS-15.4 for Level 2 management apply to the Facility Director). If not, revise the proposed TS to include the minimum qualification requirements for Level 2 facility management.

Response: The following paragraph has been modified in proposed TS 6.1, Selection and Training of Personnel: "Level 1 and 2 personnel are not required to have operating licenses and will be appointed by the appropriate bodies at RPI. The minimum qualification for the Facility Director is an advanced degree in nuclear science or nuclear engineering. Five years of experience during reactor operation may be substituted for an advanced degree."

64. ANSI/ANS-15.1, Section 6.2.1 recommends that the review and audit group shall be composed of a minimum of 3 members. Proposed TS 6.2.2 does not specify the minimum number of NSRB members. Revise the proposed TS to include the minimum membership allowed by the NSRB Charter, or justify not including such requirements.

    Response: A minimum of 3 persons was added under Composition and Qualifications.

65. ANSI/ANS-15.1, Section 6.2.2(2) recommends that operating personnel do not constitute the majority of a quorum. Proposed TS 6.2.2 does not specify that operating staff shall not constitute the majority of a quorum. Revise the proposed TS to specify that operating staff shall not constitute the majority of a quorum, or justify not including such requirements.

    Response: Section 6.2.2(2) amended to include "In addition the majority of the quorum shall not be composed operating staff (administrative levels 3 and 4)."

66. ANSI/ANS-15.1, Section 6.2.2(4) recommends that the NSRB Charter include provisions for dissemination, review, and approval of NSRB minutes. Proposed TS 6.2.2(c) requires distribution of the NSRB meeting minutes, but does not require any review or approval of the minutes. Revise the proposed TS to include requirements for review and approval of meeting minutes, or justify not including such requirements.

    Response: 6.2.3 changed to read "Minutes of each NSRB meeting shall be distributed, reviewed, and approved by the Chairman and NSRB members, and such others as the Chairman may designate."

67. ANSI/ANS-15.1, Section 6.2.3 recommends that the NSRB review determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not require a license amendment, as described in 10 CFR 50.59. The proposed TS do not require the NSRB to review such changes to equipment or systems. Explain the reason for not requiring the NSRB to review determinations that proposed changes in equipment and systems do not require a license amendment, as described in 10 CFR 50.59, or justify not including such requirements.

    Response: Section added reading, "Proposed changes in equipment, systems, tests, experiments, or procedures do not require a license amendment, as described in 10 CFR 50.59".

68. ANSI/ANS-15.1, Section 6.2.3 recommends that the NSRB review proposed changes in reactor facility equipment or system having safety significance. The proposed TS do not require the NSRB to review such changes. Revise the proposed TS to require the NSRB to review proposed changes in reactor facility equipment or system having safety significance, or justify not including such requirements.

Response: Section added reading, "Proposed changes in reactor facility equipment or system having safety significance."

69. ANSI/ANS-15.1, Section 6.2.3 recommends that the NSRB review audit reports. Proposed TS 6.2.3 does not contain any such requirement. Revise the proposed TS to require the NSRB to review audit reports, or justify not including such requirements.

Response: Added "Audit reports" to TS 6.2 *Review Function.*

70. ANSI/ANS-15.1, Section 6.2.4 recommends audits of the facility emergency plan and implementing procedures. Proposed TS 6.2.4 does not explicitly require such audits. Revise the proposed TS to require audits of the facility emergency plan and implementing procedures, or justify not including such a requirement.

Response: Added reading "Facility emergency plan and implementing procedures" to TS 6.2, *Audit Function.*

71. ANSI/ANS-15.1, Section 6.2.4 recommends that deficiencies identified by the audit group be reported in writing to Level 1 management. Proposed TS 6.2.4 does not contain such a requirement. Revise the proposed TS to require deficiencies identified by the audit group to be reported in writing to Level 1 management, or justify not including such requirements.

Response: "The case of that any deficiency is identified during the audit, the auditing group shall report, in writing, directly to the Dean, School of Engineering." added to the end of section 6.2.

72. Proposed TS 6.4.4 states that there will be procedures for the periodic surveillance of continuous air monitors. The proposed TS do not contain any requirement to have continuous air monitors. Explain this apparent inconsistency, and revise the proposed TS as appropriate.

Response: 3.7.4 added reading "A continuous air monitor that draws air from near the surface of the reactor tank is required to be operating while the reactor is operating."

73. Proposed TS 6.4.4 states that there will be procedures for the implementation of the facility emergency plan in accordance with 10 CFR 50, Appendix E. The reference to 10 CFR 50, Appendix E seems inappropriate given that Appendix E states that for licensees other than power reactors, the required degree of compliance with the requirements will be determined (by the NRC) on a case-by-case basis. The approved facility emergency plan should satisfy the required degree of compliance with Appendix E. Revise the proposed TS to require procedures for the implementation of the approved facility emergency plan, or justify the need for the reference to 10 CFR 50, Appendix E.

Response: TS 6.4.5 changed to read "Procedures for implementing the approved facility emergency plan." Reference to 10 CFR 50, Appendix E has been removed.

74. ANSI/ANS-15.1, Section 6.4(8) recommends that procedures be established for the use, receipt, and transfer of byproduct material. If these activities are carried out under the reactor license, revise the proposed TS to include requirements for procedures for use,

receipt, and transfer of byproduct material or provide justification for not requiring such procedures.

Response: 6.4.8 added stating "Use, receipt, and transfer of byproduct material."

75. ANSI/ANS-15.1, Section 6.4 recommends that temporary deviations from procedures be reported to Level 2 management within 24 hours. Revise the proposed TS to include such a requirement, or justify not including such a requirement.

Response: Sentence changed to read "All such temporary changes to the procedures shall be documented, reported to the Facility Director within 24 hours, and subsequently reviewed by the NSRB".

76. Proposed TS 6.5 uses the term "unreviewed safety question." This term is no longer used in the regulations. ANSI/ANS-15.1, Section 6.5 recommends that all new experiments or class of experiments shall be reviewed by the NSRB and approved in writing by the Facility Director or designated alternates. Revise the proposed TS to eliminate the term "unreviewed safety question" and require that all new experiments or class of experiments shall be reviewed by the NSRB and approved in writing by the Facility Director or designated alternates, or justify not including such requirements.

Response: "that might involve an unreviewed safety question" removed from sentence.

77. Proposed TS 6.5 contains a duplication of a large portion of 10 CFR 50.59, which is unnecessary given that it is a regulatory requirement and may be incorporated by reference (Note: The duplication begins with "A licensee shall obtain a license amendment..." and ends with proposed TS 6.5.1(h)). ANSI/ANS-15.1, Section 6.2.3 recommends that the NSRB review determinations that proposed changes in tests and experiments do not require a license amendment, as described in 10 CFR 50.59. Revised the proposed TS to include such a requirement and eliminate the duplication with the regulations, or justify not making these changes.

Response: TS 6.5.1 modified to include, "NSRB approval shall ensure that compliance with the requirements of the license technical specifications and 10 CFR50.59 and shall be documented. This includes NSRB review of determinations that proposed changes in tests and experiments do not require a license amendment, as described in 10 CFR 50.59." The duplications cited have been deleted.

78. ANSI/ANS-15.1, Section 6.5(2) recommends that substantive changes to previously approved experiments shall be made only after review by the NSRB and approved in writing by the Facility Director. Proposed TS 6.5.2 does not require approval in writing by the Facility Director. Revise the proposed TS to require approval in writing by the Facility Director, or justify not including such a requirement.

Response: First sentence of TS 6.5.2 now reads "Substantive changes to previously approved experiments shall be made only after review and approval in writing by NSRB and the Facility Director."

79. 10 CFR 50.36 requires that records of the results of each review of exceeding the safety limit, the automatic safety system not functioning as required by the limiting safety system settings, or any limiting condition for operation not being met be retained by the

licensee until the Commission terminates the license for the facility. Proposed TS 6.8.1(b) requires records of reportable occurrences be retained for five years. The regulations in 10 CFR 50.36 require some records categorized in the proposed TS as records of reportable occurrences to be retained for the life of the facility. Revise the proposed TS to include a requirement that records of the results of each review of exceeding the safety limit, the automatic safety system not functioning as required by the limiting safety system settings, or any limiting condition for operation not being met be retained until the U.S. Nuclear Regulatory Commission terminates the license for the facility.

Response: Added "4. records of the results of each review of exceeding the safety limit, the automatic safety system not functioning as required by the limiting safety system settings, or any limiting condition for operation not being met" to TS 6.9, operating records to be kept for the life of the facility.

Explanation and Justification of other Changes to the Technical Specification

1. TS 3.9, 3rd paragraph under *Specifications* the following sentence was removed: "Should any interruption of the loading occur (more than four days), all fuel elements except the initial loading step shall be removed from the core in reverse sequence and the operation repeated."

   Justification: RPI determined that the 4-day limit on fuel loading interruptions had no technical basis and was a carryover from older specifications.

2. TS 3.2 Specifications, Item 1: the word clean was removed.

   Justification: No definition of "clean" was given in the TS. Since we do not produce appreciable concentrations of fission products in the fuel all fuel at the RCF is considered satisfies the typical definition of "clean."

3. Definition of fail was added to TS chapter 1 stating "A component or experiment has failed if it is no longer able to perform its intended function or causes the unintentional addition or removal of reactivity.

   Justification: "Fail" appears several times in the TS and no formal definition was given.

4. Third sentence in TS 5.4 Specifications was changed to : "The fuel shall be stored in cadmium clad steel tubes with a minimum center-to-center separation of 8.5 inches and with no more than 15 SPERT (F-1) fuel pins per tube mounted on a steel wall rack".

   Justification: The RCF fuel vault is unique to the RCF and some basic description should be given to allow for easy reference.

TECHNICAL SPECIFICATIONS

CRITICAL EXPERIMENTS FACILITY

RENSSELAER POLYTECHNIC INSTITUTE

October 2010

Approved: Yaron Danon 10/10/2010

Dr. Yaron Danon, Chair
Nuclear Safety Review Board

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TABLE OF CONTENTS

Technical Specifications

1. INTRODUCTION

1.1 Scope

The following constitute the Technical Specifications (TS) for the Rensselaer Polytechnic Institute (RPI) Critical Experiments Facility (RCF), as required by 10 CFR 50.36.

1.2 Application

Content and section numbering are in accordance with section 1.2.2 of ANS-15.1-2007.

1.3 Definitions

bottomed: A control rod is bottomed if it is resting on the carrier plate in the hydraulic buffer at the bottom of the core.

channel: A channel is the combination of sensor, line, amplifier, and output devices that are connected for the purpose of measuring the value of a parameter.

channel calibration: A channel calibration is an adjustment of the channel such that its output corresponds with acceptable accuracy to known values of the parameter that the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip, and shall be deemed to include a channel test.

channel check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same parameter.

channel test: A channel test is the introduction of a signal into the channel for verification that it is operable.

control rod: A control mechanism consisting of a stainless steel basket that houses two absorber sections, one above the other. These absorber sections contain boron in iron clad in stainless steel. All are of the same dimensions, nominally 2.6 inches square, with their poisons uniformly distributed. When the control rods are bottomed the absorbers shall extend above the top and to within one inch of the bottom of the fueled portion of the core.

core configuration: The core configuration includes the number, type, and arrangement of fuel elements, and control rods occupying the core grid.

excess reactivity: Excess reactivity is that amount of reactivity that would exist if all reactivity control devices and movable experiments were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$).

experiment: Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate reactor characteristics or that is intended for irradiation within the reactor.

fail – A component or experiment has failed if it is no longer able to perform its intended function or causes the unintentional addition or removal of reactivity.

fully inserted: A control rod is fully inserted if it is within one inch of being bottomed.

known core: A core configuration for which the power indicating instrumentation has been calibrated in accordance with surveillance procedures and the following parameters have been measured:

1. excess reactivity,
2. shutdown reactivity, all rods bottomed and one rod stuck in the full out position,
3. reactivity worth of most reactive fuel pin.

license: The written authorization, by the NRC, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

measured value: The measured value is the value of a parameter as it appears on the output of a channel.

movable experiment: A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating.

operable: Operable means a component or system is capable of performing its intended function.

operating: Operating means a component or system is performing its intended function.

protective action: The initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit.

reactor operating: The reactor is operating whenever the reactor tank contains moderator and any fuel, and any control rod is not bottomed.

reactor operator (RO): An individual who is deemed capable and qualified by the SRO on duty to manipulate the controls of the reactor. The individual may be the SRO on duty, another SRO or someone without a Senior Reactor Operator License.

reactor safety systems: Reactor safety systems are those systems, including their associated input channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

reactor secured: The reactor is secured when

1. *Either* there is insufficient moderator available in the reactor to attain criticality, all control rods are bottomed, and the console keys are removed,
2. *Or* all fuel pins have been removed from the reactor.

reactor shutdown: The reactor is shutdown if all control rods are bottomed and it is subcritical by at least 1.00 $ in the reference core condition with the reactivity worth of all installed experiments included.

reactivity worth of an experiment: The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended position.

readily available on call: An operator is readily available on call if within 60 minutes normal travel time and 25 miles of the facility and personnel at the facility can readily contact the individual.

reference core condition: The condition of the core when it is at ambient temperature (cold) and the control rods are bottomed.

reportable occurrences

1. Release of radioactivity from the facility above allowed limits;
2. Discovery of loose surface contamination, excluding contamination due to naturally occurring radionuclides such as radon daughters;
3. Operation with actual safety system setting less conservative than the limiting safety system settings;
4. Operation in violation of limiting conditions for operation;
5. Any reactor safety system component malfunction that could render the safety system incapable of performing its intended function;
6. An unanticipated or uncontrolled change in reactivity greater than 60 cents; or
7. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

review and approve: The reviewing group or persons shall carry out a review of the matter in question and may either approve or disapprove it. Before it can be implemented, the matter in question must receive approval from the reviewing group or persons.

safety channel: A channel in the reactor safety system.

scram time: Scram time is the elapsed time between the initiation of a scram signal and indication that the control rod has been at least fully inserted.

secured experiment: A secured experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise because of credible malfunctions.

secured shutdown: The reactor is secured and the facility administrative requirements are met for leaving the facility with no licensed operators present.

senior reactor operator (SRO): An individual who is licensed to direct the activities of reactor operators at the RCF.

shall, should, and may: The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

shutdown margin: Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive rod in the most reactive position, and that the reactor will remain subcritical without further operator action.

shutdown reactivity: The reactivity of the reactor at ambient conditions with all control rods bottomed, including the reactivity of installed experiments.

surveillance frequency: Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. In cases where the elapsed interval has exceeded 100% of the specified interval, the next surveillance interval shall commence at the end of the original specified interval. Allowable surveillance intervals, as defined in ANSI/ANS 15.1 (2007) shall not exceed the following:

1. Annual (interval not to exceed 15 months).
2. Semiannual (interval not to exceed seven and one-half months).
3. Quarterly (interval not to exceed 4).
4. Monthly (interval not to exceed 6 weeks).
5. Daily prior to the first reactor startup of the day.

surveillance interval: The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable.

true value: The true value is the actual value of a parameter.

unknown core: Any core configuration that is not a known core.

unscheduled shutdown: An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or checkout operations.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits – Fuel Pellet Temperature

Applicability

These specifications apply to the maximum temperature reached in any in-core fuel pellet because of either normal operation or transient effects.

Objective

To identify the maximum temperature beyond which material degradation to the fuel and/or its cladding is expected and to define a safety limit below this level.

Specification

Fuel pellet temperature at any point in the core, resulting from normal operation or transient effects, shall be limited to no more than 1000 °C.

Bases

Specific determination of the melting point of the SPERT fuel has not been reported. A safety limit of 1000 °C is well below the listed melting point of UO_2 under a wide variety of conditions. The chosen value is conservative in view of variations that might result because of the presence of small quantities of impurities and the comparatively high vapor pressure of UO_2 at elevated temperatures. The safety limit specified is about 1700 °C below the measured melting point of UO_2 in a helium atmosphere.[1] Additionally, the safety limit of 1000 °C is below the melting point of Stainless Steel 304[2], the cladding material. Therefore, with the conservative assumption that the clad is at the same temperature as the fuel, the cladding integrity would not be compromised.

2.2 Limiting Safety System Settings

Applicability

These specifications apply to the settings that initiate protective action for instruments monitoring parameters associated with the reactor power limits and rate of power level changes.

Objective

To ensure protective action before safety limits are exceeded.

Specification

The limiting safety system settings on reactor power shall be as follows:

1. Maximum Power Level 100 watts
2. Minimum Period 5 seconds

Bases

The maximum power level trip setting of 100 watts on Log Power and Period Channel 2 (PP2) correlates with the operating license limit. The scram set point is used in the safety analysis with the assumption that initial power is at 100 watts indicated power.

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[1] W.A. Duckworth, ed., "Physical Properties of Uranium Dioxide," Uranium Dioxide: Properties and Nuclear Applications, Naval Reactors, Division of Reactor Development, Washington D.C., pp. 173-228 (1961).

[2] E.A. Avallone, T.B. Baumeister, III, ed., Mark's Standard Handbook for Mechanical Engineers, 9th Edition, pp. 6-11, McGraw-Hill, Inc., New York, (1987).

The minimum 5-second period is specified so that the automatic safety system channels have sufficient time to respond in the event of a very rapid positive reactivity insertion. Power and maximum fuel pellet temperature increase subsequent to scram initiation are thereby limited to well below the identified safety limit. This scram is not used in the analysis of the most severe accident since the analysis assumes that the safety channel with a fast rate scram fails concurrent with the reactivity addition.

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

Applicability

These specifications apply to reactivity in the control rods plus the maximum reactivity contained in movable experiments, and reactivity coefficients.

Objective

The purpose of these specifications is to ensure that the reactor is operated within the range of parameters that have been analyzed.

Specifications

1. The excess reactivity of the reactor above cold, critical shall not be greater than 0.60 $.
2. Above 100 °F the isothermal temperature coefficient of reactivity shall be negative. The net positive reactivity insertion from the minimum operating temperature to the temperature at which the coefficient becomes negative shall be less than 0.15 $.
3. The void coefficient of reactivity shall be negative, when the moderator temperature is above 100 °F, within all standard fuel assemblies and have a minimum average negative value of 0.00043 $/cc within the boundaries of the active fuel region.
4. The minimum operating temperature shall be 50 °F.

Bases

Excess reactivity must be limited to ensure any reactivity addition accident is restricted to one that has been analyzed and shown to cause no core damage. The assumption in this analyzed accident is a step insertion of 0.60 $ of reactivity above critical. The minimum absolute value of the temperature coefficient of reactivity is specified to ensure that negative reactivity is inserted when reactor temperature increases above 100 °F. It is of note that even in the worst postulated accident scenarios, such as considered in Section 4 of the Safety Analysis Report (1964) (SAR), reactivity insertion because of temperature change would be negligible. The minimum average negative value of the void coefficient is specified to ensure that the negative reactivity inserted because of void formation is greater than that which was calculated in the SAR. The minimum operating temperature of 50 °F establishes the temperature range for which the net positive reactivity limit can be applied.

3.2 Reactor Control and Safety Systems

Applicability

These specifications apply to all methods of changing core reactivity available to the reactor operator.

Objective

To ensure that available shutdown reactivity is adequate and that positive reactivity insertion rates are within those analyzed in the SAR.

Specifications

1. The maximum reactivity worth of any fuel pin shall be 0.20 $.
2. There shall be a minimum of four operable control rods. The reactor shall be subcritical by more than 0.70 $ with the most reactive control rod fully withdrawn. The minimum shutdown reactivity with all four control rods bottomed shall be 1.00 $.
3. The scram time for each control rod from its fully withdrawn position to its fully inserted position shall be less than or equal to 900 milliseconds. This includes a maximum 50 millisecond magnetic clutch release time.
4. The auxiliary reactor scram (moderator-reflector water dump) shall add negative reactivity within one minute of its activation.
5. The normal moderator-reflector water level shall be established not greater than 10 inches above the top grid of the core.
6. The minimum safety channels that shall be operable during reactor operation are listed in Table 1.
7. One startup channel must be operational during facility evolutions (including but not limited to fuel movement, control rod movement, experimental apparatus insertion) and audible indication must be present in the reactor room.
8. The moderator dump may be bypassed for known cores with the permission of the SRO on duty. After a scram, the moderator dump valve may be re-closed by the SRO on duty if the cause of the scram is known, all control rods are verified to have fully inserted, and the reactor is decreasing in power.
9. The interlocks that shall be operable while the rods are not fully inserted are listed in Table 2.
10. The thermal power level shall be controlled so as not to exceed 100 watts, and the integrated thermal power for any consecutive 365 days shall not exceed 2 kilowatt-hours.

Table 1: Minimum Safety System Channels

Reactor Conditions – Ranges		Channels	Minimum Number	Functions
Startup:	2 cps - 10^4 cps	Log Count Rate	1	Minimum Flux Level Interlock (see Table 2)
Power:	10^{-11} – 10^{-3}amps	Linear Power	2	High Neutron Level Scram
	10^{-14} - 10^{-3} amps +999 – -999 seconds	Log-N; Period	1	High Neutron Level and Period Scram
		Manual Scram[(a)]	2	Reactor Scram
		Control Panel 1 Power	1	Reactor Scram
		Reactor Door Scram[(b)]	1	Reactor Scram

(a) The manual scram shall consist of a regular manual scram at the console and a manual electric switch, which shall disconnect the electrical power of the facility from the scram circuit rectifier, causing a loss of power scram.

(b) The reactor door scram may be bypassed during maintenance checks and radiation surveys with the specific permission of the Operations Supervisor.

Table 2: Interlocks

Interlocks	Action if Interlock Not Satisfied
Reactor Console Keys (2) "On"	Reactor Scram
Reactor Period > 15 sec	Prevents Control Rod Withdrawal
Neutron Flux > 2 cps	Prevents Control Rod Withdrawal
Line Voltage to Recorders > 100 V	Prevents Control Rod Withdrawal
Moderator-Reflector Water Fill "Off"	Prevents Control Rod Withdrawal

Bases

The worth of a single fuel pin varies considerably depending upon where the pin is located. Removal of a pin near the center will increase reactivity for under-moderated configurations while removal of a pin on the periphery will reduce reactivity. A maximum worth is specified to provide additional margin to the limit of 0.60 $ excess

reactivity in any experiment that removes a fuel pin. Limiting worth to 0.20 $ also ensures that the operator will not have difficulty controlling power during the normal operation of measuring reactivity changes by pulling control rods to the top stop and measuring reactor period.

The minimum number of four control rods is specified to ensure that there is adequate shutdown capability even for the stuck control rod condition.

The scram time of less than 900 milliseconds from the fully withdrawn position is specified to ensure that the insertion time does not exceed that assumed when analyzing the consequence of the most severe credible accident.

The auxiliary reactor scram is specified to assure that there is a secondary mode of shutdown available during reactor operations. 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 maximum water height of 10" above the top of the core ensures that the water dump will insert negative reactivity within one minute of activation, provides a large upper reflector to allow consistency between critical position measurements and experiments, and prevents instrument tube flooding that could disable a safety system channel.

If the shielding provided by the moderator is desired (or for any other reason) the moderator may be retained after a scram. Before the auxiliary scram may be terminated by reclosing the moderator dump valve, the rods should be verified to have been fully inserted and the reactor must be decreasing in power. During operation of known cores, the auxiliary scram valve may be bypassed with approval from the SRO on duty to prevent the loss of shielding (or for any other reason) as this core has been proven to meet all other shutdown criteria with the activation of only the primary scram.

The safety system channels listed in Table 1 provide a high degree of redundancy to assure that human or mechanical failures will not endanger the reactor facility or the general public.

The interlock system listed in Table 2 ensures that only authorized personnel can operate the reactor and the proper sequence of operations is performed. It also limits the actions that an operator can take, and assists the operator in safely operating the reactor. The minimum flux level has been established at 2 cps to prevent a source-out startup and provide a positive indication of proper instrument function before any reactor startup. Not requiring the interlocks while the rods are fully inserted but not bottomed allows for approximately one inch of rod travel to verify the operability of these interlocks. Experience has shown that rod worth in the first inch is small.

The annual limit for integrated power is set at 2 kWh to ensure that the maximum dose in any unrestricted area will not exceed 100 mrem per year and the maximum dose in any restricted area (not including the reactor room itself, which should not normally be occupied during operation) will not exceed 5 rem per year.

3.3 Coolant systems – None required

No system is needed specifically to cool the fuel, because the reactor is operated at such low power levels. The roughly 2000 gallons of water used as the moderator is in direct

contact with the fuel and provides enough thermal inertia that no noticeable increase in temperature can be achieved using only energy released through fission.

3.4 Containment or confinement – None required

3.5 Ventilation Systems – None required

3.6 Emergency Power – None required

No emergency power exists at the facility. If building power is lost a passive scram is initiated, bottoming all control rods and draining the moderator regardless of the water dump bypass condition.

3.7 Radiation Monitoring

Applicability

These specifications apply to the minimum radiation monitoring requirements for reactor operations and fuel handling.

Objective

The purpose of these specifications is to ensure that adequate monitoring is available to preclude undetected radiation hazards to facility personnel and the public.

Specifications

1. The minimum complement of radiation monitoring equipment required to be operating for reactor operation shall include:
    a. An area gamma monitoring system that shall have detectors at least in the following locations: (1) control room; (2) reactor room near the fuel vault; (3) reactor room (high level monitor), and; (4) outside the reactor room window.
    b. The radiation monitors required by 3.7.1a may be temporarily removed from service if replaced by an equivalent portable unit.
    c. A calibrated and operational portable survey meter capable of measuring ambient radiation exposure shall be available.
2. During fuel loading or unloading, or during any experiments involving the addition or removal of material from the core (activation foils, etc.) a thin-window GM detector shall be available to check for personnel or area contamination.
3. A criticality detector system that monitors the main fuel storage area is required at all times except while the fuel vault is locked and maintenance on this system is being performed. This system shall have a visible and an audible alarm in the control room. This system may be the same as the area gamma monitor required by 3.7.1a (2).
4. A continuous air monitor (CAM) that draws air from near the surface of the reactor tank shall be operating while the reactor is operating.

Bases

The continuous monitoring of radiation levels in the reactor room and other stations ensures the warning of the existence of any abnormally high radiation levels. The availability of required portable monitors provides assurance that personnel will be able to monitor potential radiation fields before an area is entered.

In all cases, the low power levels encountered in operation of the reactor minimizes the probable existence of high radiation levels.

A CAM will be able to detector large levels of Ar-41 or radioactive airborne particulate, which may indicate accidental activation of the reactor room air or release of radioactive material from fuel rupture or experimental equipment failure.

The criticality monitor may be inoperable temporarily due to maintenance, during which access to the fuel vault is prohibited to minimize the possibility of a criticality accident.

3.8 Experiments

Applicability

These specifications apply to all experiments placed in the reactor tank.

Objective

The objective of these specifications is to define a set of criteria for experiments to ensure the safety of the reactor and personnel.

Specifications

1. No new experiment shall be performed until a written procedure that has been developed to permit good understanding of the safety aspects is reviewed and approved by the Nuclear Safety Review Board (NSRB) and approved by the Operations Supervisor. Experiments that fall in the general category, but with minor deviations from those previously approved, may be approved directly by the Operations Supervisor.
2. No experiment shall be conducted if the associated experimental equipment could interfere with the control rod functions, or with the safety channels.
3. No credible experiment failure shall interfere with another experiment, experimental apparatus or affect fuel cladding.
4. No power transients shall possibly cause an experiment to fail.
5. For movable experiments with an absolute worth greater than 0.35 $, the maximum reactivity change for withdrawal and insertion shall be 0.20 $/sec. Moving parts worth less than 0.35 $ may be oscillated at higher frequencies in the core.
6. The maximum positive step insertion of reactivity that can be caused by an experimental accident or experimental equipment failure of a movable or unsecured experiment shall not exceed 0.60 $.
7. Experiments shall not contain materials that can cause a violent chemical reaction. Unencapsulated experiments shall not contain a material that may

produce significant airborne radioactivity. Encapsulated experiments may contain materials that can cause a minor release of airborne radioactivity, subject to the limits in TS 3.8.10.

8. Experiments containing known explosives or highly flammable materials shall not be installed in the reactor.

9. All experiments that corrode easily and are in contact with the moderator shall be encapsulated within corrosion resistant containers.

10. All experiments containing radioactive material shall be evaluated for their potential release of airborne radioactivity. Limits shall be established for the permissible quantity of radioisotopes in the experiments such that a complete release of all gaseous, volatile, or particulate constituents instantaneously and uniformly distributed in the reactor room air would not exceed twice the associated value of Table 2, Column 1 in Appendix B to 10 CFR Part 20 for that radioisotope and inhalation class.

Bases

The basic experiments to be performed in the reactor programs are described in the SAR. The present programs are oriented toward reactor operator training, the instruction of students, and with such research and development as is permitted under the terms of the facility license. To ensure that all experiments are well planned and evaluated prior to being performed, detailed written procedures for all new experiments must be reviewed by the NSRB and approved by the Operations Supervisor.

Since the control rods enter the core by gravity and are required by other technical specifications to be operable, no equipment should be allowed to interfere with their functions. To ensure that specified power limits are not exceeded, the nuclear instrumentation must be capable of accurately monitoring core parameters.

All new reactor experiments are reviewed and approved prior to their performance to ensure that the experimental techniques and procedures are safe and proper and that the hazards from possible accidents are minimal. A maximum reactivity change is established for the remote positioning and for oscillation of experimental samples and devices during reactor operations to ensure that the reactor controls are readily capable of controlling the reactor.

All experimental apparatus placed in the reactor must be properly secured. In consideration of potential accidents, the reactivity effect of movable apparatus must be limited to the maximum accidental step reactivity insertion analyzed. This corresponds to a 0.60 $ positive step while operating at full power followed by one failure in the reactor safety system.

Restrictions on irradiations of explosives and highly flammable materials are imposed to minimize the possibility of explosion of fires in the vicinity of the reactor.

To minimize the possibility of exposing facility personnel or the public to radioactive materials, no experiment will be performed with materials that could result in a violent chemical reaction, or cause a corrosive attack on the fuel cladding.

The limitation in TS 3.8.10 is designed to simultaneously ensure that dose limits in

restricted and unrestricted areas are not exceeded in the event of a release of radioactive material contained in an experiment to the surrounding air. Values in Appendix B, Table 2, Column 1 represent the concentration inhaled on a continuous basis resulting in one-half the annual limit of dose to members of the public (50 mrem), with no credit taken for dilution between the point of discharge and the receptor location. The bounding condition occurs when the discharge rate equals the standard persons breathing rate, at 1.2 m^3/hour. When the air is discharged from the reactor room at this rate, the annual average effluent concentration will be no greater than one-fifth the concentrations in Appendix B, Table 2, Column 1, and therefore limits members of the public in unrestricted areas to less than one-tenth the annual dose limit due to this discharge. As no credit is taken for dilution between restricted and unrestricted areas, this limitation will necessarily provide adequate protection to radiation workers in restricted areas as well.

3.9 Facility-specific Limiting Conditions for Operations

Applicability

The limiting conditions for operations presented in this section are applicable at any time the reactor is not secured.

Objective

To prevent inadvertent addition of reactivity to the core and radiation exposure to facility personnel.

Specification

All fuel transfers shall be conducted under the direction of a SRO.

Operating personnel shall be familiar with health physics procedures and monitoring techniques, and shall monitor all operations and evolutions with appropriate radiation instrumentation.

For a completely unknown or untested core, fuel loading shall follow the inverse multiplication approach to criticality and, thereafter, meet TS 4.2.

For a known core, up to a quadrant of fuel pins may be removed from the core or a single fuel pin may be replaced with another pin only under the following conditions:

1. The net change in reactivity has been previously determined by measurement or calculation to be negative or less than 0.20 $;
2. The reactor is subcritical by at least 1.00 $ in reactivity;
3. There is initially only one vacant position within the active fuel lattice;
4. The nuclear instrumentation is on scale;
5. The dump valve is not bypassed; and
6. The critical rod bank position is checked after the operation is complete.

Bases

The Basis for fuel transfers being monitored by a SRO is to ensure that the fuel transfers are performed in accordance with facility specifications. During movement of fuel, the

basis for radiation monitoring is to provide indication of the level of radioactivity in the vicinity of the fuel and core. The basis for limiting the re-arrangement of fuel is to prevent inadvertent insertion of excess reactivity above the 0.60 $ limit and to ensure adequate shutdown reactivity exists.

4. SURVEILLANCE REQUIREMENTS

4.1 Reactor Core Parameters

Applicability

These specifications apply to the verification of shutdown reactivity, reactivity worth of fuel, and reactor power levels that pertain to reactor control.

Objective

The purpose of these specifications is to ensure that the analytical bases are and remain valid and that the reactor is safely operated.

Specifications

The following parameters shall be determined during the initial testing of an unknown or previously untested core configuration:

1. excess reactivity;
2. worth of most reactive fuel pin;
3. reactor power instrument calibration; and
4. shutdown reactivity.

Bases

Measurements of the above parameters are made when a new reactor configuration is assembled. Whenever the core configuration is altered to result in an unknown or untested configuration, the core parameters are evaluated to ensure that they are within the limits of these specifications and the values analyzed in the SAR. During this test period of the reactor, measurements are performed using the approved experimental procedures.

The excess reactivity measurement is made to verify that this configuration is not subject to a reactivity addition accident more severe that that analyzed and described in the SAR, Section 13.2.

This same accident assumes a scram signal at a maximum power level of 100 watts indicated so it is necessary to measure reactor power and make any necessary adjustments to the instrumentation that indicates reactor power. The scram signals are based in detector current while the visual display is in watts. The high current scram must be verified to not exceed an indicated 100 watts.

Lastly, the accident analysis assumes the reactor is shutdown by at least 1.00 $ of reactivity after the high current scram occurs. Shutdown reactivity is also measured to ensure the reactor meets the definition of shutdown when all control rods are bottomed.

4.2 Reactor Control and Safety Systems

Applicability

These specifications apply to the surveillance of the safety and control apparatus and instrumentation of the facility.

Objective

The purpose of these specifications is to ensure that the safety and control equipment is operable and will function as required in TS 3.2.

Specifications

1. The scram time, shall be measured semiannually to verify that the requirements of TS 3.2.3, are met.
2. The moderator-reflector water dump time shall be measured semiannually to verify that the requirement of TS 3.2.4, is met.
3. The startup and power safety channels shall be calibrated annually.
4. A channel test of the safety system channels and a visual inspection of the reactor shall be performed daily prior to reactor startup. The interlock system shall be tested daily prior to reactor startup to satisfy rod drive permit. These systems shall be retested following a secured shutdown.
5. The moderator-reflector water height shall be checked visually prior to reactor startup to verify that the requirements of TS 3.2.5, are met.
6. In addition to the scheduled surveillances, any system shall be tested to prove operability after all modification, maintenance, or repairs have been made to that system.
7. Integral power shall be tallied quarterly as long as the three previous consecutive quarters do not exceed 1.5 kW. If three consecutive quarters exceed a total of 1.5 kW, the surveillance shall be monthly.
8. Requirements 1 thru 6 may be waived when the instrument, component, or system is not required to be operable, but the instrument, component or system shall be tested prior to being declared operable.

Bases

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 TS 3.2.3 and 3.2.4.

Manual scram, CP1 power, and reactor door scrams are except from annual calibration because the signal is it "on" or "off." A check is satisfactory to determining the operability of these safety systems.

Visual inspection of the reactor components, including the control rods, prior to each day's operation, is to ensure that the components have not been damaged and that the core is in the proper condition. Redundant safety channels are provided by having three independent channels provide high current scrams if necessary and by requiring all three channels be operable. The analysis of the most severe accident shows no fuel damage

even if one channel fails. Random failures should not jeopardize the ability of the overall system to perform its required functions. The interlock system for the reactor is designed so that its failure places the system in a safe or non-operating condition. However, to ensure that failures in the safety channels and interlock system are detected as soon as possible, frequent surveillance is desirable and thus specified. All of the above procedures are enumerated in the daily startup checklist.

Past experience has indicated that, in conjunction with the daily check, calibration of the safety channels annually ensures the proper accuracy is maintained.

If components are not needed, such as during prolong secured periods, components are not used and are not required to be operable. Before reactor operations can resume all requirements must be met including all safety system channels are verified as operable.

4.3 Coolant Systems

No coolant system exists.

4.4 Containment or Confinement – None required

4.5 Ventilation Systems – None required

No surveillances are required for the ventilation system

4.6 Emergency Power – None required

No emergency power system exists

4.7 Radiation Monitoring

Applicability

These specifications apply to the surveillance of the area radiation monitoring equipment and all portable radiation monitoring instruments. These specifications also apply to moderator in the storage tank or reactor tank.

Objective

The purpose of these specifications is to ensure the continued validity of radiation protection standards in the facility.

Specification

The criticality detector system, CAM and area gamma monitors shall be tested with a radiation source at least monthly and daily if the reactor is operated and calibrated semiannually. Portable instruments shall be calibrated annually.

Portable survey meters shall be calibrated at the manufacturer's recommended frequency.

Prior to discharge to the environment the moderator shall be monitored for radioactivity to prove that gross activity levels are lower that maximum levels permitted by 10 CFR 20 Appendix B Table 2.

Bases

Experience has demonstrated that calibration of the criticality detectors, CAM and gamma monitors semiannually is adequate to ensure that significant deterioration in accuracy does not occur. Furthermore, the operability of these radiation monitors is included in the daily pre-startup checklist. If the reactor is not operated for more than a month, the instruments are required to be checked to ensure operability. Portable instruments are calibrated at the manufacturer recommended frequency.

Experience has demonstrated that the moderator does not accumulate radioactive material due to the low operating neutron fluence. Therefore, periodic monitoring is not necessary. Verification is necessary, however, prior to discharge to the environment.

4.8 Experiments – None required

Since experiments may very drastically no general surveillances can defined. However, approved experimental procedures may contain experiment specific surveillances.

4.9 Facility-specific Surveillance Requirements – None required

No facility specific surveillances are required.

5. DESIGN FEATURES

5.1 Site and Facility Description

Applicability

These specifications apply to the design of the RCF and the surrounding site.

Objective

The purpose of these specifications is to provide a layout of the site and the structures that contain the reactor in a means to protect personnel.

Specification

The facility is located on a site situated on the south bank of the Mohawk River in the City of Schenectady. An inner fence of greater than 30 feet radius defines the restricted area. An outer fence and riverbank of greater than 50 feet radius defines the exclusion area.

The reactor is housed in the reactor building. The security of the facility is maintained by the use of two fences, one at the site boundary and the other defining the restricted area around the reactor building itself.

The reactor room is a 12-inch reinforced concrete enclosure with approximate floor dimensions of 40x30 feet. The height from the ground floor to the ceiling shall be about 30 feet. The roof is a steel deck covered by 2 inches of lightweight concrete, five plies of felt and asphalt, with a gravel surface. Access to the reactor room is through a sliding fireproof steel door that also contains a smaller personnel door. Near the center of the room is a pit 14.5 x 19.5 feet wide and 12 feet deep with a floor of 18-inch concrete.

This part contains the 3500-gallon water storage tank and other piping and auxiliary equipment.

Bases

The inner and outer fences provide for the security of the facility. The sliding steel access door provides a means to move equipment into and out of the reactor room. The smaller personnel door permits personnel access without sliding the door out of position. The 3500-gallon water storage tank allows for the storage of approximately 2000 gallons needed to fill the reactor tank for operations with an additional volume to maintain net positive suction head for the reactor fill pump.

5.2 Reactor Coolant System

Applicability

These specifications apply to design of the reactor tank and the methods by which the tank can be dumped or filled.

Objective

The purpose of these specifications is to demonstrate the size of the reactor tank, its connection to the water tank and how the water is to be introduced into or removed from the reactor tank.

Specification

The reactor core is installed in a stainless steel reactor tank that has a capacity of approximately 2000 gallons of water. The tank nominal dimensions are 7 feet in diameter and 7 feet high. The tank is supported at floor level above the reactor room by 8-inch steel I-beams. There are no side penetrations in the reactor tank.

The reactor tank is connected to the water storage tank via a six-inch quick dump line. Therefore, it is required that the storage tank be vented to the atmosphere such that its freeboard volume can always contain all water in the primary system. The water handling system allows remote filling and emptying of the reactor tank. It provides for a water dump by means of a failsafe butterfly-type gate valve when a reactor scram is initiated. The filling system shall be controlled by the operator, who must satisfy the sequential interlock system before adding water to the tank. A pump is provided to add the moderator-reflector water from the storage dump tank into the reactor tank. A nominal six-inch valve is installed in the dump line and has the capability of emptying the reactor tank on demand of the operator or when a reactor scram is initiated, unless bypassed with the approval of the licensed senior operator on duty. A valve is installed in the bottom drain line of the reactor tank to provide for completely emptying the reactor tank.

Bases

The capacity of the reactor tank is adequate to contain the core support structure, lattice plates, detectors, control rods, immersion heaters, and agitator, while still providing adequate moderation and reflector savings for the core. The 6-inch dump line and fail-safe butterfly valve provide for rapidly draining the moderator from the reactor tank to the storage tank in the event of a scram. The fill rate of approximately 50 gpm allows for

completing the reactor tank fill in a reasonable amount of time. The sequential interlock system prevents the simultaneous addition of moderator with control rod withdrawal.

5.3 Reactor Core and Fuel

Applicability

These specifications apply to the makeup of the fuel pellets and the support structure that contains the fuel.

Objective

The purpose of these specifications is to provide a detailed makeup of the fuel composition and to give the fuel pin design and configuration with support structures.

Specification

The reactor core shall consist of uranium fuel in the form of 4.8 weight percent or less enriched UO_2 pellets in metal cladding, arranged in roughly a cylindrical fashion with four control rods placed symmetrically about the core periphery. The total core configuration and the arrangement of individual fuel pins, including any experiment, shall comply with the requirements of TS 3.1 and 3.2. Core fuel pins to be utilized are 4.8 weight percent enriched SPERT (F-1) fuel rods. Each fuel rod is made up of sintered UO_2 pellets, encased in a stainless steel tube, capped on both ends with a stainless steel cap and held in place with a chromium nickel spring. Gas gaps to accommodate fuel expansion are also provided at both the upper end and around the fuel pellets. NUREG-1281 describes these fuel pins in additional detail.

The fuel pins are supported and positioned on a fuel pin support plate, drilled with holes to accept tips on the end of each pin. The support plate rests on a carrier plate, which forms the base of a three-tiered overall core support structure. An upper fuel lattice plate rests on the top plate, and both are drilled through with holes with the prescribed arrangement to accommodate the upper ends of the fuel pins. The lower fuel pin support plate, a middle plate, and the upper fuel pin lattice plate are secured with tie rods and bolts. The entire core structure is supported vertically and anchored by four posts set in the floor of the reactor tank.

Four control rod assemblies are installed, spaced 90 degrees apart at the core periphery. Each rod consists of a 6.99-cm square stainless steel tube, which passes through the core and rests on a hydraulic buffer on the bottom carrier plate of the support structure.

Housed in each of these "baskets" are two neutron-absorber sections, one positioned above the other.

Bases

The basis for the fuel pin specifications comes from the SPERT fuel pin description in NUREG-1281. The support structure and lattice plates are designed to support a nominal core load of fuel pins and the four perimeter control rods. The control rod absorber sections are arranged such that the combination of the four rods satisfy the requirements, with regard to reactivity with one stuck rod and shutdown reactivity.

The total core configuration and the arrangement of individual fuel pins, including any experiment, shall comply with the requirements of TS 3.1 and 3.2. The core shall consist

of all SPERT (F-1) fuel. All core components are composed of stainless steel, eliminating the risk of corrosion. Fuel pins have been qualified by the DOE and NRC in accordance with their standards details of compositions. The design criteria of the fuel pins was set to minimize the risk of fission product release. The enriched boron absorber sections are strategically positioned one above the other. In the end, each of the four rods has approximately the same reactivity effect.

5.4 Fissionable Material Storage

Applicability

These specifications apply to the storage of fuel not loaded in the reactor core.

Objective

The purpose of these specifications is to define the storage of fuel when it is not needed in the reactor core and what precautions are taken to keep the stored fuel from becoming critical.

Specification

When not in use, the SPERT (F-1) fuel shall be stored within the storage vault located in the reactor room. The vault shall be closed by a locked door and shall be provided with a criticality monitor near the vault door. The fuel shall be stored in cadmium clad steel tubes with a minimum center-to-center separation of 8.5 inches and with no more than 15 SPERT (F-1) fuel pins per tube mounted on a steel wall rack. The center-to-center spacing of the storage tubes, together with the cadmium clad steel tubes, ensures that the infinite multiplication factor is less than 0.9 when the vault is fully flooded with water.

Bases

Fuel not loaded in the reactor is stored in the fuel vault for security and for criticality safety. The spacing of the tubes, the limit of 15 pins per storage tube, and the cadmium sheet wrapped on the storage tube ensure conditions in the vault remain subcritical in the event of a complete flood of the vault. The criticality monitor provides for indication of an inadvertent criticality in the fuel vault.

6. ADMINISTRATIVE CONTROLS

6.1 Organization

Structure

The organization for the management and operation of the reactor facility shall include the structure indicated in Figure 1.

Level 1:	Dean, School of Engineering
Level 2:	Facility Director
	Chair, Nuclear Safety and Review Board (NSRB)
Level 3:	Operations Supervisor
Level 4:	RO's and SRO's

Responsibility

The Dean, School of Engineering, is responsible for the facility license and appoints the Chair, NSRB. The Facility Director is responsible for facility administration and safety. The NSRB serves as an independent review and auditing body, this board is described in further detail in TS 6.2. The Operations Supervisor is responsible for the day-to-day safety and operation of the facility, as well as coordinating the training of new RO's and SRO's. The Operations Supervisor has the primary responsibility to ensure surveillances and maintenance are performed when necessary and operator proficiency is maintained. The RO's and SRO's are responsible for conducting day-to-day operations and maintenance in accordance with facility procedures.

The RPI Radiation Safety Officer (RSO) who is organizationally independent of the reactor operations group shall provide advice as required by the Facility Director and the Operations Supervisor in matters concerning radiological safety. The RSO also has interdiction responsibility and authority.

Personnel at the various management levels, in addition to the duties and responsibilities outlined above, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and technical specifications.

Staffing

1. The minimal staffing when the reactor is not shutdown as described in these specifications shall be:
    a. A RO or SRO licensed pursuant to 10 CFR 55 present at the controls.
    b. One other person in the control room certified by the SRO on duty as qualified to activate the manual scram and initiate emergency procedures.
    c. A SRO shall be present or readily available on call. The identity of and method for rapidly contacting the SRO on call shall be known to the operator.
2. The minimal staffing when the reactor is shutdown, but not secured is a SRO on duty in the control room and a second SRO present or readily available on call.
3. A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator. The list must include:
    a. Management personnel.
    b. Radiation safety personnel.
    c. Other operations personnel.
4. Events requiring the direction of the Operations Supervisor:
    a. All fuel or control rod relocations within the reactor core unless the activity is part of an approved experiment.

b. Recovery from unplanned or unscheduled shutdown.

5. Responsibility of any level may be delegated to either a designated alternate or by a member of a higher administrative level, conditional on all appropriate qualifications are met by the alternate.

Selection and Training of Personnel

The selection, training and requalification of operations personnel shall meet or exceed the requirements of American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Sections 4-6.

Additionally, the minimum requirements for the Operations Supervisor are at least four years of reactor operating experience and possession of a Senior Reactor Operator License for the RPI Critical Facility. Years spent in baccalaureate or graduate study in a nuclear engineering discipline or in the US Navy Nuclear Power School may be substituted for operating experience on a one-for-one basis up to a maximum of two years.

Level 1 and 2 personnel are not required to have operating licenses and will be appointed by the appropriate bodies at RPI. The minimum qualification for the Facility Director is an advanced degree in nuclear science or nuclear engineering. Six years of nuclear experience during reactor operation and a bachelor's degree in engineering may be substituted for an advanced degree.

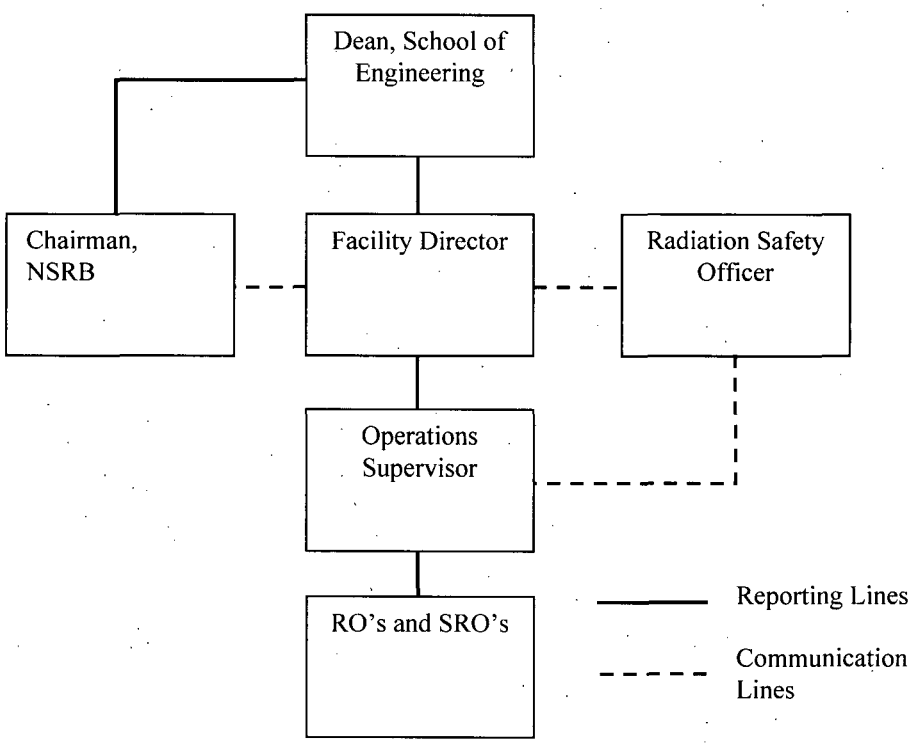

Figure 1: RCF Management Organization

6.2 Review and Audit

A NSRB shall review and audit reactor operations and advise the Facility Director in matters relating to the health and safety of the public and the safety of facility operations.

Composition and Qualifications

The NSRB shall be appointed by the Dean, School of Engineering, in accordance with the NSRB Charter. The NSRB shall consist of a minimum of 3 persons. The Chair will be appointed by the Dean.

Charter and Rules

The NSRB Charter shall describe the composition of the board. The NSRB shall function under the following rules:

1. The NSRB shall meet at least semiannually.
2. The quorum shall consist of not less than a majority of the full NSRB and shall include the Chairman or his designated alternate. In addition, the majority of the quorum shall not be composed of operating staff (administrative levels 3 and 4).
3. Minutes of each NSRB meeting shall be distributed, reviewed, and approved by the Chairman and NSRB members, and such others as the Chairman may designate.

Review Function

The following items shall be reviewed and approved by the NSRB before implementation:

1. Proposed experiments and tests utilizing the reactor facility that are significantly different from tests and experiments previously performed at the facility;
2. Proposed changes in equipment, systems, tests, experiments, or procedures do not require a license amendment, as described in 10 CFR 50.59;
3. Proposed changes in reactor facility equipment or system having safety significance;
4. Reportable occurrences;
5. Proposed changes to the TS and proposed amendments to the facility license;
6. Operating, Emergency and Surveillance procedures;
7. Audit reports.

Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Where necessary, discussions with cognizant personnel shall take place. In no case shall the individual immediately responsible for the area audit in the area. The following areas shall be audited at least annually.

1. Reactor operations and reactor operational records for compliance with

internal rules, regulations, procedures, and with license provisions;

2. Existing operating procedures for adequacy and to ensure that they achieve their intended purpose in light of any changes since their implementation;
3. Plant equipment performance with particular attention to operating anomalies, abnormal occurrences, and the steps taken to identify and correct their use;
4. Facility emergency plan and implementing procedures.

The case of that any deficiency is identified during the audit, the auditing group shall report, in writing, directly to the Dean, School of Engineering.

6.3 Radiation Safety

The Radiation and Nuclear Safety Committee and the Radiation Safety Officer shall be responsible for the implementation of the Radiation Safety Program for the RCF. The primary purpose of the program is to assure radiological safety for all University personnel and the surrounding community.

AS LOW AS IS REASONABLY ACHIEVABLE (ALARA) PROGRAM

Control of ionizing radiation exposure is based on the assumption that any exposure involves some risk. However, occupational exposure within accepted limits represents a very small risk compared to the other risks voluntarily encountered in other work environments.

The policy of RPI is to maintain occupational exposures of individuals to be well within allowable limits as are defined in the appropriate regulations. The individual and collective dose to workers is maintained as low as reasonably achievable (ALARA).

ALARA is a part of the normal work process where people are working with ionizing radiation. Management at all levels, as well as each individual worker, must take an active role in minimizing this radiation exposure.

Exposures at the facility are routinely reviewed by the Radiation Safety Officer and Radiation and Nuclear Safety Committee to ensure that proper radiation safety procedures are in place and ALARA is maintained.

6.4 Procedures

Written procedures shall be prepared, reviewed and approved prior to initiating any of the activities listed in this section. The procedures, including applicable checklists, shall be reviewed by the NSRB and followed for the following operations:

1. Startup, operation and shut down of the reactor.
2. Installation and removal of fuel pins, control rods, experiments, and experimental facilities.
3. Corrective actions to be taken to correct specific and foreseen malfunctions such as for power failures, reactor scrams, radiation emergency, responses to alarms, moderator leaks and abnormal reactivity changes.

4. Periodic surveillance of reactor instrumentation and safety systems, area monitors, and continuous air monitors.
5. Procedures for implementing the approved facility emergency plan and facility security plan.
6. Maintenance procedures that could have an effect on reactor safety.
7. Use, receipt, and transfer of byproduct material.

Substantive changes to the above procedures shall be made only with the prior approval of the NSRB.

Temporary changes to the procedures that do not change their original intent may be made with the approval of the Operations Supervisor. All such temporary changes to the procedures shall be documented, reported to the Facility Director within 24 hours and subsequently reviewed by the NSRB.

6.5 Experiment Review and Approval

1. All new experiments or classes of experiments shall be reviewed by the NSRB. NSRB approval shall ensure compliance with the requirements of the license, TS and 10 CFR 50.59, and shall be documented. This includes NSRB review of determinations that proposed changes in tests and experiments do not require a license amendment, as described in 10 CFR 50.59.
2. Substantive changes to previously approved experiments shall be made only after review and approval in writing by NSRB and the Facility Director. Minor changes that do not significantly alter the experiment may be approved by the Operations Supervisor.
3. Approved experiments shall be carried out in accordance with established approved procedures.
4. Prior to review, an experiment plan or proposal shall be prepared describing the experiment, including any safety considerations.
5. Review comments of the NSRB setting forth any conditions and/or limitations shall be documented in committee minutes and submitted to the Facility Director.

6.6 Required Actions in the Event of a Safety Limit Violation

1. The reactor shall be shutdown and reactor operations shall not be resumed until authorized by the NRC.
2. The safety limit violation shall be promptly reported to Facility Director or designated alternates and to the NSRB.
3. The safety limit violation shall be reported to the NRC in accordance with TS 6.8 Special Reports Item 2.
4. A safety limit violation report shall be prepared. The report shall describe the following:

a. Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public.

c. Corrective action to be taken to prevent reoccurrence.

The report shall be reviewed by the NSRB and any follow-up report shall be submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation of the reactor.

6.7 Required Actions in the Event of a Reportable Occurrence

1. The reactor shall be shut down. Operations shall not be resumed unless authorized by the Chair, NSRB.
2. Occurrence shall be reported to the Facility Director or designated alternate, the NSRB and to the NRC not later than the following working day by telephone and confirmed in writing to the NRC, to be followed by a written report that describes the circumstances of the event within 14 days of the event.
3. All such conditions, including action taken to prevent or reduce the probability of a recurrence, shall be reviewed by the NSRB. The NSRB shall concur with corrective actions.

6.8 Reports

In addition to the requirements of applicable regulations, and in no way substituting therefore, all written reports shall be sent to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555, with a copy to the Region I Administrator.

Operating Reports

A written report covering the previous year shall be submitted by March 1 of each year. It shall include the following:

1. Operations Summary. A summary of operating experience occurring during the reporting period that relates to the safe operation of the facility, including:

   a. Changes in facility design;

   b. Performance characteristics (e.g., equipment and fuel performance);

   c. Changes in operating procedures that relate to the safety of facility operations;

   d. Results of surveillance tests and inspections required by these Technical Specifications;

   e. A brief summary of those changes, tests, and experiments that require authorization from the NRC pursuant to 10 CFR 50.59, and;

   f. Changes in the plant operating staff serving in the following

positions:

i. Facility Director;

ii. Operations Supervisor;

iii. RSO;

iv. NSRB Members.

2. Power Generation. A tabulation of the integrated thermal power during the reporting period.

3. Shutdowns. A listing of unscheduled shutdowns that have occurred during the reporting period, tabulated according to cause, and a brief description of the preventive action taken to prevent recurrence.

4. Maintenance. A tabulation of corrective maintenance (including major preventative maintenance) performed during the reporting period on safety related systems and components.

5. Changes, Tests and Experiments. A brief description and a summary of the safety evaluation for all changes, tests, and experiments that were carried out without prior NRC approval pursuant to the requirements of 10 CFR 50.59.

6. A summary of the nature, amount and maximum concentrations of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

7. Radioactive Monitoring. A summary of the TLD dose rates taken at the exclusion area boundary and the site boundary during the reporting period..

8. Occupational Personnel Radiation Exposure. A summary of radiation exposures greater than 25% of the values allowed by 10 CFR 20 received during the reporting period by facility personnel (faculty, students or experimenters) and visitors.

Special Reports

1. Reportable Operational Occurrence Reports. Notification shall be made within 24 hours by telephone in accordance with 10 CFR 50.36(c)(7) followed by a written report in accordance with 10 CFR 50.36(c)(5) within 10 days in the event of a reportable operational occurrence as defined in Section 1.3. The written report on these reportable operational occurrences, and to the extent possible, the preliminary telephone and e-mail notification shall: (1) describe, analyze, and evaluate safety implications; (2) outline the measures taken to ensure that the cause of the condition is determined; (3) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems; and (4) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

2. Unusual events. A written report in accordance with 10 CFR 50.36(c)(5) shall be submitted as specified in 10 CFR 50.4 within 30 days in the event of discovery of any substantial errors in the transient or accident analyses or in the methods used for such analyses, as described in the SAR or in the bases for the TS.
3. Key changes in Organization. A written report in accordance with 10 CFR 50.36(c)(5) submitted as specified in 10 CFR 50.4 shall be provided for any change in Level 1 or Level 2 personnel.

6.9 Operating Records

The following records and logs shall be maintained at the RCF or at RPI for at least five years:

1. Normal facility operation (except retain checklists for one year) and principal maintenance operations;
2. reportable occurrences;
3. tests, checks, and measurements documenting compliance with surveillance requirements;
4. experiments performed with the reactor;
5. fuel shipments, inventories, and receipts;
6. reactor facility radiation and contamination surveys;
7. approved changes to operating procedures;
8. records of NSRB meetings and audits.

Records to be retained for at least one certification cycle:

Records of training or retraining of certified operations personnel shall be maintained at all times the individual is employed or until the certification is renewed.

The following records and logs shall be maintained at the RCF or at RPI for the life of the RCF:

1. gaseous and liquid radioactive releases from the facility;
2. TLD environmental monitoring systems;
3. radiation exposures for all RPI Critical Facility personnel (students and experimenters) and visitors;
4. records of the results of each review of exceeding the safety limit, the automatic safety system not functioning as required by the limiting safety system settings, or any limiting condition for operation not being met;
5. the present as-built facility drawings and new updated or corrected versions.