Rensselaer

DEPARTMENT OF MECHANICAL,
AEROSPACE, AND NUCLEAR ENGINEERING

RCF 08-04
July 28, 2008

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

Re: Response to Request for Additional Information

Dear Sir:

This letter provides the information requested by your letter of March 21, 2008, "Request for Additional Information Regarding the Rensselaer Polytechnic Institute (RPI) Application for Renewal of Facility License No. CX-22 for the Rensselaer Polytechnic Institute Reactor Critical Facility".

The requested information is attached in several parts. Specifically:

Responses to the detailed questions

Revised Chapter 12, Safety Analysis Report

Revised Technical Specifications

Explanation of Changes to the Technical Specifications

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

Sincerely;

GWinters

Glenn Winters, Director
L. David Walthousen Laboratory

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

_________ Timothy Wei

Date Dr. Timothy Wei, Interim Dean of Engineering

Rensselaer Polytechnic Institute
110 8th Street | Troy, NY 12180-3590 USA

A020
NRR

cc:

Dr. Michael Podowski, Chair
RPI NSRB

Dr. Achille Messac, Interim Chair
MANE

Dr. Timothy Wei, Interim Dean
School of Engineering

William Kennedy, NRC

Dr. Jeffery Geuther, Operations Supervisor
L. David Walthousen Laboratory

Dr. Peter Caracappa
Radiation Safety Officer

Dr. Timothy Trumbull, Adjunct Professor
MANE

Glenn Winters, Director
L David Walthousen Laboratoy

RPI Responses to RAI dated March 21, 2008

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

RPI Response – The crane in the Reactor Room is operable and will be inspected periodically by a subcontractor to ANSI and OSHA standards. This is planned to commence in Fall 2008. The most recent inspection was 2004. The only restrictions judged necessary are to restrict lifting loads over the reactor tank when the reactor is fueled. No other lifting casualty would endanger equipment critical to reactor safety. Crane operation is under the supervision of the duty Senior Reactor Operator who is responsible to enforce the necessary restrictions.

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

RPI Response – The facility ground floor is 2 feet above the 100-year flood level, but below the 500-year flood level. Facility flooding would potentially damage equipment, but the normal secured shutdown status requires no operable equipment to ensure reactor safety. The fuel stored in the vault is subcritical when fully flooded (see the response to RAI 9.1), as would be the fueled core if fully flooded. There are no plans to attempt to remove the reactor fuel when a flood is pending since it is judged that the removal process would be more hazardous than leaving the fuel in the safe configuration within the facility. The facility would be guarded whenever it was vulnerable, such as if the intrusion detection system was inoperable, to prevent unauthorized access. Further, the absence of radioactive contamination, verified by periodic sampling, means that flooding would not release any radioactive material.

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

RPI Response – The following explanation of the auxiliary reactor scram will be added to the SAR. The addition will be a paragraph at the end of Section 4.3:

"The reactor tank is filled by pumping water from the storage tank with the Reactor Fill Pump. Water may be drained slowly from the tank through the fill line by operating the air-operated drain valve. See Figure 5.1. Water may be quickly drained from the reactor tank by opening the 6-inch butterfly valve on the fast dump line. This valve is also air-operated and fails open on loss of air pressure or loss of site power. When the dump valve is opened water rapidly drains back into the storage tank and ensures the reactor is shutdown regardless of the position of the control rods. The dump valve automatically opens if a reactor scram occurs. A keyed switch on the Auxiliary Control Panel, CP-2, can override this automatic trip when allowed by operating procedures."

Operating procedures allow the automatic trip of the dump valve to be bypassed if the duty Senior Reactor Operator concurs and if the reactor contains a known core. A known core is one for which several core parameters have been measured to verify the core configuration is within the design envelope analyzed for casualties. A specific definition of a known core is contained in the Technical Specifications.

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

RPI Response – The SPERT (F-1) fuel pins provided to RPI were determined to be suitable for low power use by the evaluation reported in NUREG-1281. The conclusion remains valid today since the fuel has continued to be used in the low power RPI reactor. The fuel is stored dry between reactor operations. The typical reactor operation consists of a few minutes at power levels below 100 watts. After operation, the moderator is drained from the reactor tank. The fuel pins are wetted for only a few hours considering time to prepare to operate, operating time, and time to secure the facility to secure the facility. Visual inspection and contamination surveys have not shown any corroded or otherwise defective pins. Surveys of fuel pins show no significant buildup of fission products. Radiation measurements a few days after reactor operation show radiation levels of about 1 millirem per hour at contact on a typical fuel pin.

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

RPI Response – The control rod gantries can be swiveled and extended to change the position of the control rods in the tank. Moving the control rods would require cuts to be made in the lattice plates, and they have not been moved since the core was re-fueled with LEU. Reactor physical configuration is intentionally flexible to allow experiments with varying configurations.

The design boundaries are included in the Technical Specifications. These boundaries specify the number of control rods (four) and shutdown reactivity. Planned deviation from these constraints would require a license revision.

Part of the qualification process of a core given in the technical specifications is the measure of control rod worth to verify that rod worth, drop time, and shutdown margin requirements are met. The critical loading procedure for an unknown core is a very conservative approach requiring that the reactor moderator is drained after every group of pins is loaded (using the inverse multiplication approach) and that the excess reactivity is measured after each group of pins is loaded.

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

RPI Response – The bank control rod worth is approximately $2.00 over a 36" stroke, for an average differential worth of less than 6 cents per inch. The rod bank withdrawal rate is 3 inches per minute. Therefore, the average reactivity insertion rate is approximately 6 cents / inch * 3 inches / minute * 1 minute / 60 seconds = 0.3 cents ($0.003) per second. The technical specifications currently require that the maximum reactivity insertion rate due to bank withdrawal be five cents ($.05) per second when the flux is greater than ten times source level. This would require the maximum differential bank worth to be at least sixteen times greater than the average worth, which is not credible given the sinusoidal shape of the differential bank worth curve. The revised Technical Specification no longer has a limit on control rod worth since control rod differential worth does not have a role in the accident analysis in the SAR.

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

RPI Response - The shielding present consists of the physical structure and the accessibility limitation imposed by it. Typical exposure rates at the nearest accessible points are:

Source stowed in paraffin shield: 12 mrem/hr
Source exposed, tank empty: 23 mrem/hr
Source exposed, tank full: 2.3 mrem/hr

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

RPI Response – The core support structure, piping, and reactor tank are not subject to high temperature or pressure. Therefore, age-related degradation is not expected. The core moderator is not radioactive, and is not heated, such that there is no major negative consequence of a hypothetical moderator leak (see response to 4.7 below). Due to the low probability of age-related degradation and the low impact of the consequences of a moderator leak, no inspections are performed to verify the integrity of these items.

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

RPI Response - The reactor moderator is not monitored on a periodic basis, but is always monitored prior to release. The sensitivity of the procedure to measure gross alpha/beta is on the order of a few dpm per liter, and has not exceeded the minimum detectable activity in memory. This is primarily due to the low operating power level. Therefore, any leak of reactor moderator, even if it were to reach groundwater, would not have a measureable radiological impact.

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

RPI Response - Based upon radiation measurements at approximately 13 W, the maximum anticipated dose at the fence-line boundary at full power (100 W) operation would be 1.3 mrem/hr, which complies with the limitation of no more than 2 mrem/hr to areas accessible to members of the public.

The highest dose rate at an accessible location inside the reactor building (just outside the reactor room door) at 100 W is 27 mrem/hr (sum gamma and neutron).

The revised Technical Specification establishes an annual integrated power limit of 2 kWh, which would limit the maximum possible dose in restricted areas to less than 5 rem/year and in unrestricted areas to less than 100 mrem/year without the need for any further area or access controls during operation.

No other functions, such as buildings or grounds maintenance, are performed at the facility during reactor operation. Personnel enter the reactor room area during operation only for specifically approved procedures, and any new procedures would need to be reviewed for ALARA considerations before being implemented.

4.9 Section 4.5.2, Reactor Core Physics Parameters. Section 4.5.2 does not list any core physics parameters. Temperature and void coefficients are found in Tables 13.2 and

13.3. Shutdown margin is only given as a lower bound (> 0.02) in Table 13.2. Please provide quantitative values for excess reactivity and shutdown margin in Chapter 4 and ensure that these values are consistent with the technical specifications. (See SAR RAI 13.5, TS RAI 1.3.V, and TS RAI 3.2 (D))

RPI Response – – Shutdown reactivity measurements are completed as part of the process of qualifying a known core to verify that the shutdown reactivity meets technical specifications. A typical known core has 10 – 30 cents of excess reactivity from control rod travel beyond the critical position. The limit is 60 cents and includes the reactivity change that may be caused by a movable experiment if one is installed. The control rod bank worth is approximately $2. Therefore, known cores meet the requirement of $1 shutdown reactivity with the full four-rod bank and $0.70 with a stuck control rod, given in TS 1.3 and TS 3.2, respectively. The SAR will be revised wit the next update. A revised Table 13.2 is provided with the response to RAI 13.5 below.

5. REACTOR COOLANT SYSTEMS

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

RPI Response – The water used as moderator is Schenectady city water. No chemistry controls are used since the wetted portions of the reactor water systems are stainless steel and the storage tank and piping to the fill pump are the only portions of the system that are continuously immersed. The reactor tank is normally dry with moderator present for just a few hours each time the reactor is operated. A small pump with a wound-cotton filter is routinely run on circulation with storage tank water to maintain water clarity. If clarity becomes a problem, the storage tank may be sampled to verify no detectable activity and discharged. In practice, tank discharges are infrequent. The most recent discharge was November 2006 in order to prepare for replacement of the fill pump.

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

RPI Response – Operating procedures require water level to be high enough to immerse the control rod buffer pistons on the lower support plate. These pistons act as hydraulic shock absorbers at the end of the control rod stroke. The corresponding reactor tank water level is 19.5 inches. At the upper end, water level is not allowed to be more than 10 inches above the top of the core. This corresponds to 68 inches of water in the reactor tank. The basis for this upper limit is to provide an adequate upper reflector, not flood out instrument dry wells, and minimize the time for fast dump to reduce core reactivity.

Most operations are measurements of reactor parameters when the core is fully flooded, that is, water level is 10 inches above the top of the upper support plate. Some measurements are made with water levels below the top of the upper support plate. These are benchmark measurements to determine the critical water level, that is, the water level for exactly critical when control rods are fully withdrawn.

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

RPI Response – The Startup Checklist forbids control rod movement unless reactor tank water level is above the carrier plate, that is, at least 19.5 inches. There is no interlock to prevent rod motion at lower water levels. The same Startup Checklist requires filling the reactor tank to 68 inches. Again there is no interlock associated with this level and there is no alarm. Operation at some other water level, for example, to determine critical water level, is done using an experiment procedure for the specific measurement being made. The Storage Tank is the normal repository for the moderator at the facility. When the reactor is prepared for operation, moderator is pumped from the Storage Tank into the Reactor Tank. Thus, there is always free volume in the Storage Tank for the water in the Reactor Tank.

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

RPI Response - As stated in the response to SAR RAI 4.7, the maximum activity in moderator water (given the detection limits of the available equipment) is a few dpm/L, which would result in minimal impacts from release to the environment.

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

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

RPI Response – Instrumentation upgrades are complete. The following equipment was replaced with equipment of similar capability:

Picoammeters for three uncompensated ion chambers measure and display current and have output signals to recorders. These replace similar equipment for instrument channels PP2, LP1 and LP2. The replacement equipment was custom manufactured by Circuit Equipment Corporation to RPI specifications.

Paper strip chart recorders were replaced with videographic recorders. Three of the videographic recorders are manufactured by Thermo Westronics. The fourth videographic recorder is manufactured by Honeywell. All recorders are commercially available equipment. One recorder contains alarm relays that are used to implement the control rod outmotion interlock on low startup channel counts and loss of recorder power. The remaining recorders do not provide any control function other than displaying moderator temperature and reactor neutron level as determined from the uncompensated ion chambers and the startup channel BF3 detectors.

The electronics to receive and process signals from the BF3 detectors were replaced with like-kind, commercially available equipment manufactured by Ortec and Canberra. These replacements include high voltage power supplies, counter/scalers, preamplifiers, and discriminators.

The objective of the instrument upgrade was to replace aging and unreliable equipment with commercially available, or with custom-built hardware in cases where commercially available equipment was inadequate. Another objective was to retain all prior control and interlock functions with the new equipment. Both objectives were met.

A water level detector was also procured, but is not yet installed. The intent of this equipment is to display reactor tank water level on a recorder. The future plan for the water level detector is to interlock the reactor fill pump with a high level shutoff and interlock the reactor tank immersion heaters with a low level shutoff. It is also possible that a low water level rod block could be implemented. None of these additional controls based on tank water level affect the severity or likelihood of any plant casualty.

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

RPI Response – Details of the currently installed instrumentation are provided in the response to question 7.2 below.

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

RPI Response – The reactor operating status is monitored by two types of neutron detectors. BF3 detectors are the more sensitive and show changes in reactor neutron level when the reactor is shutdown or beginning a startup. Two such detectors are installed and are labeled Startup channels A and B. Uncompensated ion chambers are less sensitive and are used as the reactor approaches operating power levels. Three such detectors are installed. Overlap is achieved by the differing detector sensitivities and by the location of the individual detectors. The arrangement ensures that two or more neutron detectors are always able to determine the neutron level in the reactor.

The Startup Instrumentation detector signals are routed through Ortec Model 142PC preamplifiers, then Ortec Model 590A amplifiers. The amplifiers include single channel analyzers to separate gamma pulses from neutron pulses. A Canberra Model 3125 Dual Voltage Power Supply provides power to both detectors. Processed signals are displayed on individual Ortec Model 449-2 Log/Lin Ratemeters. The instrument suite also includes an Ortec Model 994 Dual Counter/Timer.

Two uncompensated ion chambers comprise linear power channels LP1 and LP2. Each is powered from its own 300 volt battery. The detector signal is processed by a Circuit Equipment Corporation Model 1718 Linear picoammeter. These instruments have 9 ranges, from 1 x 10^{-11} amps to 1 x 10^{-3} amps and display current in amps. The picoammeters have an internal relay with a variable setpoint. The relay provides a high current scram signal to the rod scram circuit.

One uncompensated ion chamber comprises log power channel PP2. The ion chamber is also powered by a 300 volt battery and the signal is processed by a Circuit Equipment Corporation Model 1718 log picoammeter. The instrument has indication from 1 x 10^{-14} amps to 1 x 10^{-3} amps. Current in amps and reactor period in seconds or in decades per minute are displayed on the meter face. An internal relay sends trip signals to the rod scram circuit and has variable setpoints for high current and fast period. A second relay provides a control rod outmotion interlock at fast period. The period setpoints are 5 seconds for a scram and 15 second to block control rod outmotion.

These instruments also have recorder output signals, percent of full scale for LP1 and LP2 and amps for PP2.

Four video graphic recorders are mounted at the main control console. A Thermo Westronics Model SV-100 is used to display moderator temperature measured by J-type thermocouples. Two Thermo Westronics Model SV-180 recorders display LP1, LP2, PP2, SUA, or SUB, as the operator chooses. Typically one would display the two startup channels on one screen and the second would display PP2, or LP1 and LP2, depending upon the operating range. A third recorder, Honeywell MultiTrend Plus, is also used and can display any of the same channels. The operator can have all three sets of instrumentation in view on these three recorders. The recorders also serve as data recorders by writing to 3 ½ inch floppy discs that are then transferred to a computer hard drive for analysis and storage. Data recording is independent of the screen view. All signals sent to the recorder are recorded. Temperature on the SV-100 is recorded once per minute. Ion chamber currents, reactor period and BF3 counts per second are recorded once per second.

One of the Thermo Westronics recorders senses low Startup channel B count rate and implements the rod outmotion interlock for this condition. The setpoint is at 2 counts per second.

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

RPI Response – Instrumentation to monitor reactor power level is described in detail in the response to question 7.2 above.

Other indications available in the Control Room are:

Control Rod Position, inches withdrawn as well as top and bottom lights
Moderator temperature
Reactor Tank water level
Area Monitor (4) radiation levels
Continuous Air Monitor count rate and volumetric air flow
Rod magnetic clutch currents
Dump valve solenoid current
Startup source position
Operating lights for equipment, specifically the air compressor, immersion heaters, reactor fill pump, agitator motor, dump valve position, fill valve, and drain valve.

The operator has alarm lights on the picoammeters showing a scram or a rod block. There is a rod block visual alarm for low startup channel count rate on the videographic recorder that implements that interlock.

Control rod position is derived from optical encoders mechanically linked to the rod drive gearing. The encoders transmit a series of pulses to counters mounted on the control panel. The counters interpret the train of pulses to calculate and display rod position to 0.01 inch resolution. Separate limit switches detect rod position at the top and bottom limits of rod travel. The top limit switches stop outward rod travel and illuminate a top limit light on the control panel. Rod bottom limit switches activate Rod Bottom lights on the control panel.

Moderator temperature is measured at several elevations in the reactor tank by Type J thermocouples. Thermocouple voltage is converted to Fahrenheit degrees by circuitry in the videographic recorders that display temperature. One recorder screen displays three of the available thermocouples. A second recorder displays one thermocouple. No provision is made to measure reactor temperature since it is the same as moderator temperature.

Reactor Tank water level is displayed in a sightglass in the control room.

The four channels of the Area Monitoring system display radiation levels in the Control Room, the Equipment Hall, outside the Fuel Vault, and on the Reactor Tank upper deck. All four channels have visual alerts and audible alarms.

The Continuous Air Monitor samples air above the Reactor Tank. Activity is displayed in the Control Room and the instrument has an audible alarm. Air flow is measured by a mechanical gage located in the Control Room.

The power supply that provides direct current to the rod drive magnetic clutches displays current to each clutch. The same power supply provides power to the solenoid valve that regulates operating air to the dump valve. Current to this solenoid is displayed at the power supply. Power supply input and output voltages are also displayed.

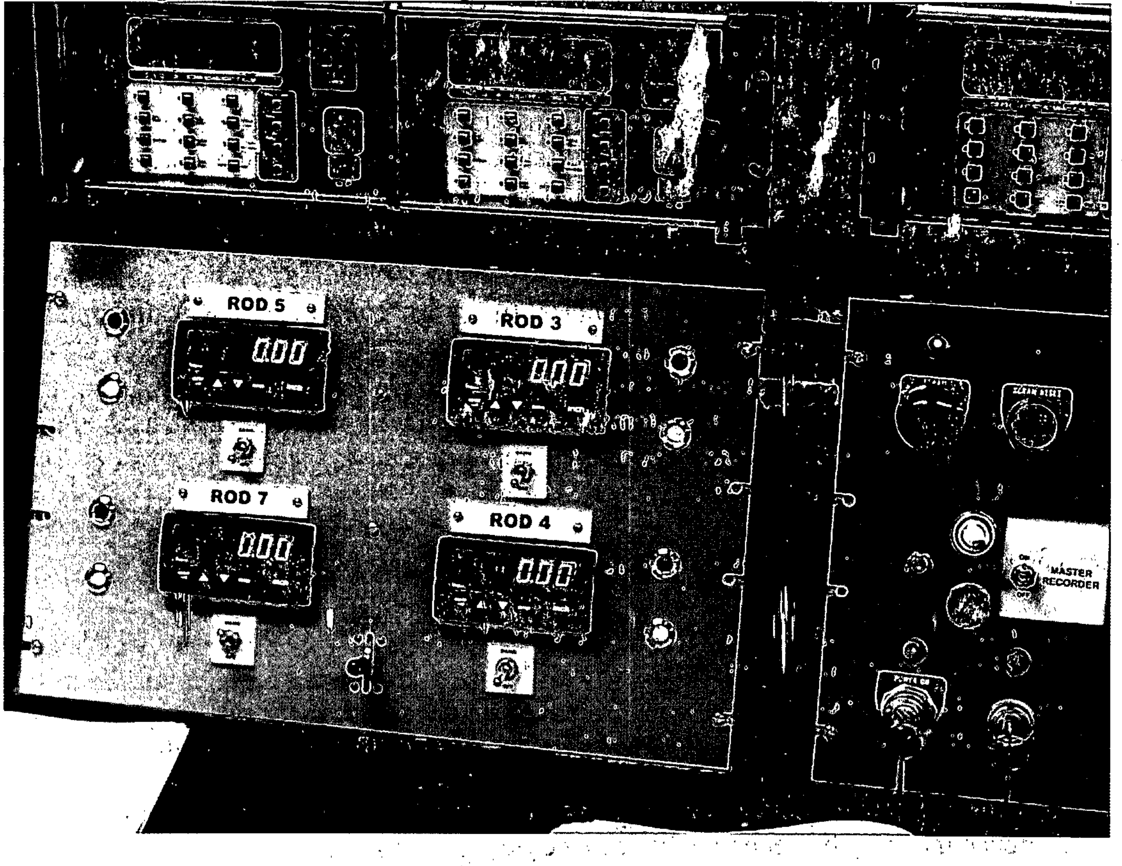
All the operating lights are located on the Auxiliary Control Panel, CP-2, in the Control Room. Position of the startup source is displayed on CP-2. Other operating lights in the Control Room are on the Control Panel, CP-1, and indicate Shutdown activated and Scram initiated. Shutdown is an operating mode that drives all rods inward at the normal operating speed. It is an interlock in that Shutdown mode overrides an outward motion command. The Scram light indicates when a scram has occurred.

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

RPI Response – The block diagram is outdated. Three ion chambers are in use and they are described in responses to questions 7.1 and 7.3 above. All three ion chambers are on the perimeter of the core at approximately the midplane of the fuel. PP2, connected to the log picoammeter is located between Rod 5 and Rod 7. See Figure 4.1. LP1 and LP2 are connected to linear picoammeters are located near Rod 4 and Rod 3 respectively.

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

RPI Response – The main control panel is known as CP-1. Electrical power for CP-1 comes from the building lighting panel. CP-1 displays include the position of the four control rods, including lights indication rods at the top or rods at the bottom. The shim switch for the control rods is located on the same section of panel as the rod position indicators. Each rod also has a switch to select control to the rod shim switch. All or any one of the rods may be selected. Thus rods may be moved in any combination desired from the single shim switch. See the photo below.



An adjacent section of CP-1 contains the keylock switch to energize the panel, a keylock switch to energize the scram circuit, a switch to select shutdown mode, a switch to energize recorders, and a scram switch. Indicating lights are associated with the shutdown switch and the scram switch. This section is shown at the right side of the photo above.

A third section of CP-1 contains selsyn dials for control rod position. This section is inactive since the 400 Hz MG set that provided power for the selsyn units was removed. Above the operating control are the electronics for the neutron detectors. Above that row of instruments are three of the videographic recorders. A separate vertical section of CP-1 contains the Area Radiation Monitoring displays and the fourth videographic recorder. The only operating controls on CP-1 are for control rod motion, including the manual scram. Control of other pumps, heaters and valves are on CP-2 the auxiliary control panel.

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

RPI Response - The purpose of the CAM is to monitor for particulate activity in the reactor room, as a marker of possible fuel element failure, rather than for radiological protection purposes. The filter collects fission products which are detected by the GM detector, so the precise radiological impact is indeterminate, depending upon the collection time and the degree of equilibrium present.

Upon reflection, it seems likely that this system is a remnant of the HUE core that was present at the facility prior to 19[illegible], when fuel elements were clad by only 0.005 in of stainless steel. The failure of one of these elements resulting in the release of contained fission products is assumed to be a more plausible occurrence than the physical failure of one of the SPERT fuel elements currently in use. This equipment will be removed from the SAR and any requirement for operation removed from the Technical Specifications.

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

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

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

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

RPI Response – The normal electrical power system consists of 60-hertz, 480-volt, three-phase power from the utility grid. The incoming service line is rated at 200 amps. The 480 volt supply directly powers the fill pump, air compressor, immersion heaters, building crane, agitator motor and a building air conditioner. A 30 kva transformer reduces incoming voltage to 120 volts and feeds a lighting panel. The lighting panel powers the rod drive motors, source drive motor, lighting circuits and wall outlets, the two control panels CP-1 and CP-2, and the facility boiler house. Instrumentation and recorders are powered from standard 120 volt outlets. The rod drive motors and the source drive motor are the only three-phase loads on the lighting panel.

Facility power usage is highest when preparing for operation due to operating the 2 horsepower fill pump to transfer water into the Reactor Tank from the Storage Tank. The 2 horsepower air compressor cycles periodically during startup preparations and during reactor operations. Unless the immersion heaters are in use, power consumption is very low during operation. The four rod drive motors are rated at 1/20 horsepower and few other loads are in use during operation. When operated the immersion heaters draw 36 kilowatts and the 2 horsepower agitator is operated to keep the moderator temperature uniform. This is by far the largest load needed for reactor operation and is used infrequently. Immersion heaters are operated for experiments that require a change of water temperature, for example, measurement of the moderator temperature coefficient of reactivity. Building capacity exceeds the demands.

All building wiring was installed to codes applicable at the time of installation. No known deviations are present. Repairs to component failures are made with new components. New motor starters were installed for the fill pump, agitator, air compressor, immersion heaters, source drive motor and the rod drive motors in 2006. These are commercially available General Electric motor starters. The main circuit breaker panel and the lighting panel were both replaced in 2006 also.

No backup power is available for the facility. Any power outage will immediately cause a reactor scram due to loss of power to the rod drive electromagnetic clutches. Loss of site power also deenergizes the solenoid that holds the dump valve shut. These actions place the reactor in the normal secure shutdown condition – rods scrammed and moderator drained from the reactor tank. Since the RCF does not generate appreciable fission products during operation there is no decay heat load to dissipate. In a secure shutdown condition there are no discharges of any material from the facility.

Emergency procedures for loss of site power require the operator to remove keys from the reactor control panel, CP-1, and turn off instrumentation to prevent power surge damage if power recovery is erratic. These actions will maintain the secure shutdown condition when power is restored.

Emergency lighting for safe egress is installed in the appropriate areas (installation completed June 2008). Battery powered flashlights are also available.

Loss of power also disables all monitoring systems at the facility such as fire detection and building security. In this situation standard procedure requires continuous on-site surveillance, normally provided by RPI Public Safety.

Instrumentation cabling is shielded to protect from electromagnetic interference.

9. AUXILIARY SYSTEMS

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

RPI Response – Both are possible. The consequences of placing a pin in a short tube are trivial. The pins will simply extend beyond the end of the tube. Otherwise, the tubes are structurally identical to the long tubes.

Consequences of overloading the tubes (more than the limit of 15 per tube) are non-trivial only in the event of a massive flood in which the entire vault is inundated. Recent Monte Carlo analysis using MCNP [Ref] assumed that that the vault was an infinite array of fuel storage tubes, completely flooded with water. The analysis showed that the maximum reactivity in the arrangement was reached at 53 pins. The maximum infinite multiplication factor, k_{inf}, was less than 0.6900 compared to the 15 pin case where $k_{inf} \approx$ 0.6100.

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

RPI Response – Inventory requirements are derived from the government ownership of the fuel and our requirement to report inventory annually to the Nuclear Materials Management and Safeguards System (NMMSS). It is not a necessary Technical Specification requirement.

10. EXPERIMENTAL FACILITIES AND UTILIZATION

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

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

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

RPI Response – Currently, the primary use of the reactor is to support the Critical Reactor Laboratory class and perform demonstrations. Demonstrations consist of one of the approved experiments demonstrated for a small group of students who are visiting the laboratory as part of a familiarization tour. These are RPI students who may take the formal laboratory course in a subsequent semester. Demonstrations may also be arranged for students from other universities, although that has not occurred in the past several years. The difference between a demonstration and the formal lab course is in the work expected of the students. The lab course requires formal reports and more active student participation in data recording and analysis.

The typical menu of experiments is provided below:

- Source range channel calibration (BF3 detectors),
- Fuel Pin Addition - Approach to Critical using Inverse Subcritical Multiplication Plots,
- Exact Bank Critical Rod Position and Excess Reactivity Measurement
- Bank and Individual Control Rod Worth Measurement,
- Measurement of Individual Fuel Pin Worths,
- Isothermal Moderator Temperature Coefficient of Reactivity,
- Void Coefficient of Reactivity,
- ^{10}B Coefficient of Reactivity,
- Interior/Exterior Radiation Survey at Power,
- Axial and Radial Power Mapping,
- Power Calibration using Gold Foils.

These are pre-approved experiments and are documented in the MANE-4440 Laboratory Manual. No equipment or facilities outside of the RCF are required for these experiments. The RCF has gamma-spectroscopy equipment used to analyze the gold foils and perform gamma-ray scanning of pins, boron-impregnated tape and polystyrene material are used for the boron and void coefficient studies, and electric immersion heaters are used for the moderator density coefficient.

Additional experiments have been conducted, e.g., TLD chip activations, and others are planned. In all cases, the operating procedures are followed depending on whether the new configuration is a "known" or "unknown" core.

Additional tests planned include criticality benchmark testing of partially-reflected core configurations (using Zircaloy, silicon carbide, concrete, aluminum, steel, etc) and Borobond™.

All new proposed experiments and procedures are reviewed by the NSRB, in accordance with Section 6.4 of the TS.

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

RPI Response – RPI will modify Section 10 of the SAR to read, "All new experiments or classes of experiments that raise an unreviewed safety question shall be reviewed and approved by the Nuclear Safety Review Board in accordance with Section 6.3 of the Technical Specifications. 10 CFR 50.59 will be consulted to determine is a license amendment is required."

11. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

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

RPI Response - Dose rates at the RCF during reactor operations at full power are discussed in response to SAR RAI 4.8. This is an extreme case for the facility, since typical operating power levels are far lower than the full license power.

After reactor shutdown, the maximum dose rate in the facility is at the position on the deck above the reactor tank, where dose rates range from about 5 mR/hr shortly after shutdown to about 0.3 mR/hour well past shutdown. Dose rates quickly drop to near background a few meters from the reactor tank.

Specific surveys have not been conducted during fuel handling procedures. However, the quarterly accumulated dose measurement during a recent quarter where a full core unload and reload was performed revealed that no staff member exceeded the minimum detectable dose of 10 mrem for the quarter.

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

RPI Response - Environmental monitors at the site boundary are subject to a minimum detectable quarterly dose of 10 mrem, and have rarely been shown to exceed this value. Environmental monitoring results are reported in the annual operating report, and are consistently shown to be well below regulatory limits. See also the response to SAR RAI 4.8.

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

RPI Response - The origin of the environmental monitoring values stated is unclear, as those values are below the minimum detectable dose for current or previously available environmental monitoring devices used at the facility. The current environmental monitoring data is reported in SAR RAI 11.2. Additional discussion is provided with the response to SAR RAI 4.8.

Compliance with 10 CFT 20.1101(d) is demonstrated through annual review using the COMPLY code and conservative assumptions regarding Ar-41 generation in the target room. Compliance with 10 CFR 20.1301(a)(2) is demonstrated by the values reported in response to SAR RAI 4.8.

The references in TS 5.1 and 5.2 are correct, and the SAR will be updated to reflect them.

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

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

RPI Response – Chapter 12 has been rewritten.

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

RPI Response – See the revised Chapter 12 and Technical Specifications

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

RPI Response – See the revised Chapter 12

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

RPI Response – See the revised Chapter 12 and Technical Specifications

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

RPI Response – See the revised Chapter 12

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

RPI Response – See the revised Chapter 12 and Technical Specifications

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

RPI Response – The ANS Standard gives 30 minutes as an example, not even a recommendation. For RPI, the 30 minute or 15 miles has been an unnecessary burden at times, and no such prompt response is judged to be necessary. Phone contact is adequate for immediate assistance while a second operator arrival up to an hour later is judged adequate response time. The available personnel can place the reactor in secure shutdown in about one minute and wait for arrival of another senior reactor operator. Emergency assistance for fire, injury or security issues is a few minutes away and provided by Schenectady civil authorities or local ambulance services. RPI Campus response such as Public Safety or radiological assistance is about 45 minutes away. The requirement is changed to 60 minutes and 25 miles. See the revised Chapter 12 and Technical Specifications.

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

RPI Response – The reference has been updated.

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

RPI Response – All staff receive training as participants in the RPI Radiation Safety Program, which includes initial training and refresher training at least annually. The training program is in accordance with the New York State Department of Health regulations (State Sanitary Code Part 16), since that is authority which licenses RPI's use of radioactive materials, and which are at least as restrictive as the 10 CFR provisions.

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

10 CFR 20.1101(b) requires an ALARA program. Who is responsible for the ALARA and radiation safety programs? When the RCF is in use, is there a person responsible for radiation safety present at the facility or on call? If on call, does the SRO have sufficient

training in radiation safety to perform those duties until assistance arrives? SAR Section 12.1.5 suggests that RCF staff have responsibility for radiation safety and the campus support is only available for occasional assistance.

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

RPI Response – The radiation safety program for the reactor is under the purview of the RPI Office of Radiation and Nuclear Safety (ORNS), part of the Environment, Health and Safety Department, as part of a radioactive materials license issued by the New York State Department of Health.

The revised Chapter 12 and Technical Specifications have corrected the organizational chart and harmonized the terminology related to the radiation safety program.

The ALARA program is part of the campus radiation safety program, and is described in the campus Radiation Safety Manual. The Radiation and Nuclear Safety Committee is responsible for reviewing the program and ensuring that radiological operations on campus, including at the reactor facility, remain ALARA.

During regular operation, the SRO is responsible for normal radiation safety tasks, such as the examples provided in 12.1.5. A dedicated member of ORNS need not be present at the facility for these types of measurements. ORNS is always on call to respond to emergency situations, and the Radiation Safety Officer provides oversight, assistance, and support for the radiological aspects of facility operations.

A new section 12.3.3 has been added to the SAR discussing the inclusion of radiation protection procedures in the campus Radiation Safety Manual. See the revised SAR Chapter 12.

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

RPI Response – See the revised Chapter 12

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

RPI Response - This question is addressed in TS 6.4.1. This paragraph has been amended to state that the review function of the NSRB is pursuant to 10 CFR 50.59.

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

RPI Response – The TS has been amended to ensure that these areas are addressed as part of the NSRB audit function.

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

RPI Response - Since the RCF has adopted the campus radiation safety program as its radiation safety program, the radiation protection procedures are administered under the structure established by the radioactive materials license with the New York State Department of Health. As such, the procedures are incorporated by reference and not included specifically in Section 12.3. However, we agree that "Radiation Protection" should appear in the bulleted list in section 12.3. See the revised Chapter 12.

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

RPI Response – See the revised Chapter 12

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

RPI Response – See the revised Technical Specification.

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

RPI Response – The proposed changes to TS 6.4 and the description of the experimental program are provided in response to SAR RAIs 10.1, 10.2, & 12.15.

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

RPI Response – The TS has been amended to require that the reactor be shut down in the event of any reportable occurrence.

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

RPI Response – See the revised Chapter 12 and Technical Specifications

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

RPI Response – The Technical Specifications already state that all written reports shall be submitted according to the first paragraph of TS 6.6.

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

RPI Response – See the revised Chapter 12 and Technical Specifications.

13. ACCIDENT ANALYSIS

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

RPI Response – The procedures governing changes to known cores and the training of the SRO's prevents the inadvertent arrangement of fuel to achieve a supercritical configuration from being a credible occurrence. The staffing requirements specify at least two operators, one of which must be an SRO, be physically at the RCF in order to operate. The possibility of both operators agreeing to remove fuel to achieve a supercritical configuration is not judged to be credible.

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

RPI Response – The delayed neutron fraction, β, is a physical property of fissionable material. Several resources provide values of β for various fissionable materials, the most notable being Keepin's work (G. R. Keepin, "Physics of Nuclear Kinetics," 1965). The value of βeff, however, is dependent on the neutron spectrum and fuel system (fissionable material and enrichments) of the reactor. It is essentially the spectrum-adjoint weighted effect of delayed neutrons,

$$\beta_{eff} = \frac{\int d\mathbf{r}^3 \int \Omega' \int dE' \int \chi_d(E')\psi(\mathbf{r},\Omega',E')\nu_d(\mathbf{r},E)\Sigma_f(\mathbf{r},E)\Phi(\mathbf{r},E)dE}{\int d\mathbf{r}^3 \int \Omega' \int dE' \int \chi_t(E')\psi(\mathbf{r},\Omega',E')\nu_t(\mathbf{r},E)\Sigma_f(\mathbf{r},E)\Phi(\mathbf{r},E)dE} \quad (1)$$

where the numerator is the spectrum-adjoint weighted neutron production from delayed neutrons only and the denominator represents the spectrum-adjoint weighted neutron production from prompt and delayed neutrons.

This equation can be solved using deterministic transport, provided the delayed χ and ν are known. These values are generally available from ENDF/B formatted data sets.

Initial diffusion theory analysis of the LEU core resulted in the current βeff = 0.00765. As a check, a new analysis was performed using MCNP5 and the ENDF/B-VI.8 data set. Two runs are necessary to calculate βeff. The first run is a normal iterated source eigenvalue calculation. The second iterated source eigenvalue calculation turns off delayed neutrons. βeff is then given as

$$\beta_{\text{eff}} = \frac{k_{\text{eff,total}} - k_{\text{eff,prompt}}}{k_{\text{eff,total}}}, \qquad (2)$$

where since continuous energy Monte Carlo is used, the adjoint weighting in (1) is approximated with the next-fission probability function. The MCNP runs resulted in a βeff = 0.00813 ± 0.00023. It is known that the ENDF/B-VI.8 delayed neutron fraction for thermal fission in ^{235}U is 0.0069, or approximately 6% higher than Keepin's value. Reducing the calculated value by 6% to match Keepin, the new calculated βeff=0.00766, which is in excellent agreement with the current specified value.

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

RPI Response – The analysis of Section 13.2 was performed at 20 C because that is the temperature that the cross section library was provided. It is not a worst-case for other reasons. However, the overall effect on the postulated accident scenario would be negligible if initiated at 10 C instead of 20 C, as documented in Amendment 7 to the operating license (July 7, 1987), Section 3.2.

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

RPI Response – We assume that the indicated power is 100 watts. To increase the conservatism of the scenario, we then assume that the indicated power could be off by as much as 100% from the true power. Table 13.1 will be updated specifying an initial power of 200 watts.

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

RPI Response – A definition of shutdown margin has been added to the TS.
Table 13.2 will be updated with the following corrections.

Shutdown Margin – remove this value from the table. The intent is to specify shutdown reactivity & shutdown reactivity with most reactive rod stuck.

Shutdown Reactivity	<-$1.00 (LEU Value)	<-$1.00 (TS)
Shutdown Reactivity with most reactive rod stuck	< -$0.65 (LEU value)	< -$0.70 (TS.
Reactivity worth of standard fuel assembly	< $0.039 (LEU value)	< $0.20 (TS.

Other changes to documents:
SAR, 1.3

"...Excess reactivity with all control rods fully withdrawn is typically less than 30 cents. The minimum shutdown reactivity of the reactor is a dollar. A more detailed description of the reactor is given in Chapter 4."

SAR 13.2

"...conservatively assumes the instantaneous insertion of $1.000 negative reactivity (the minimum core shutdown reactivity) at 5 seconds after the excursion begins."

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

RPI Response – There are other lattices plates that have been installed and tested in accordance with the TS requirements for new experiments, with the appropriate NSRB approvals. The approval is contingent on an analysis indicating that the current SAR remains valid for the proposed lattice. Previously used lattices include the 0.585" pitch lattice plate referred to in the SAR. The most commonly used lattice plate at this time is the 0.640" pitch lattice. The references to "other approved lattices" will be changed to reflect the revised TS 6.4.

SAR 1.3 changed to read

"...in an octagonal array with a 0.64" pitch (other configurations approved in accordance with section 6.4 of the technical specifications exist as well) with 4 boron flux-trap control rods..."

SAR 4.1 changed to read

"...The most commonly used fuel pin configuration utilizes a 0.640" pitch (other lattice plates approved in accordance with Section 6.4 of the technical specification are available) containing 329-333 fuel pins..."

The proposed lattice plate is analyzed to ensure that the current SAR analyses are valid for the new configuration. This is part of the approval process that is specified in TS 6.4.

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

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

RPI Response - The end of table 13.3 will be changed in the next SAR update to:

Isothermal Temperature Coefficient for LEU Core A:

$\alpha T(°C) = 1.825x10^{-8}T^2 - 4.8x10^{-6}T + 6.932x10^{-5}$

and $\alpha T < 0$ for $T > 16°C$ (61°F)

Isothermal Temperature Coefficient for LEU Core B:

$\alpha T(°C) = 2.113x10^{-8}T^2 - 5.0x10^{-6}T + 1.423x10^{-4}$

and $\alpha T < 0$ for $T > 32°C$ (91°F)

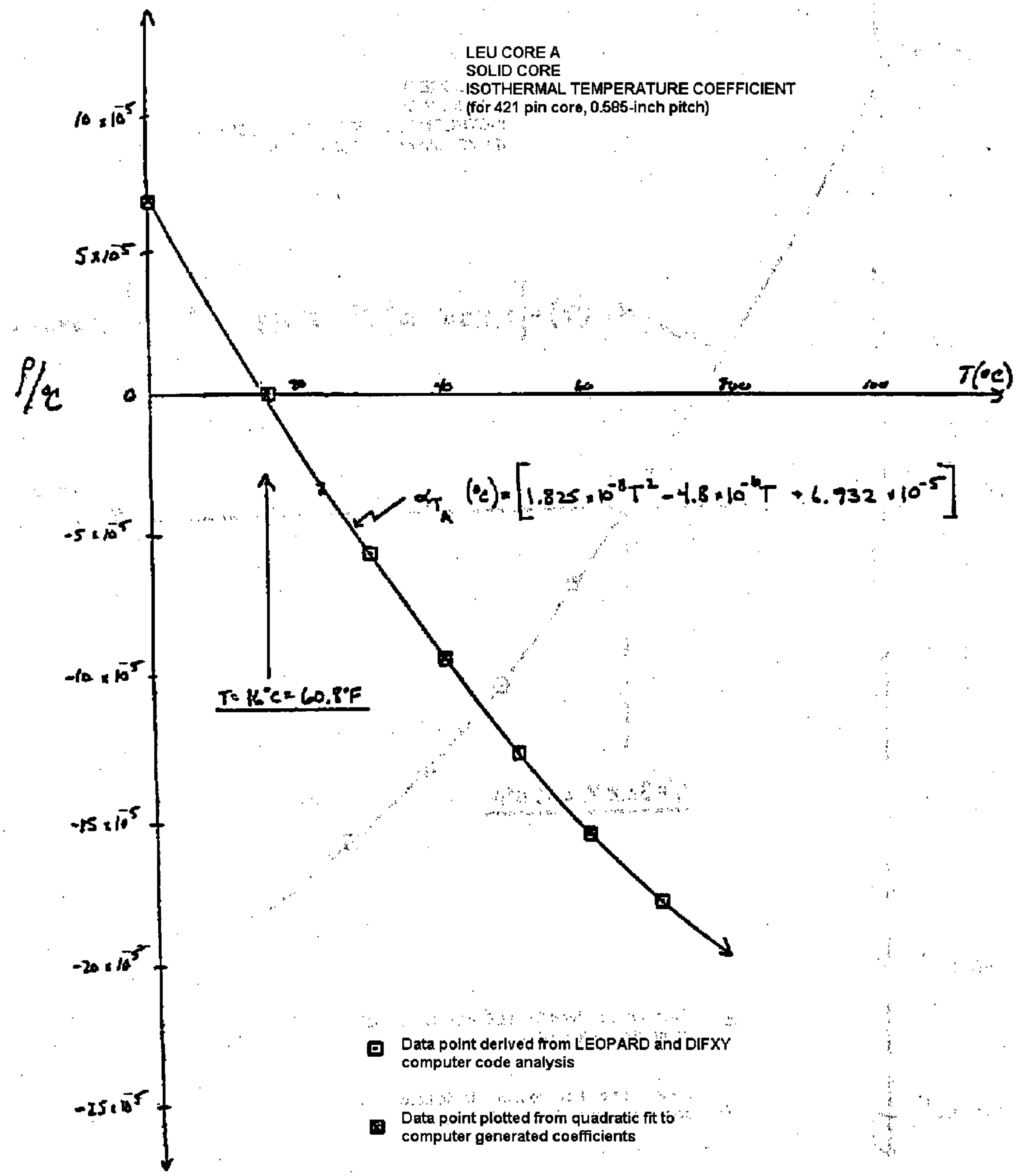
b. On Figures 13.2 and 13.3 the temperature coefficient shows a positive exponent ($\times 10^5$) which is not consistent with the equation. Please correct.

RPI Response - Figure will be changed (see revised Figure 13.2 below) in the next SAR update.

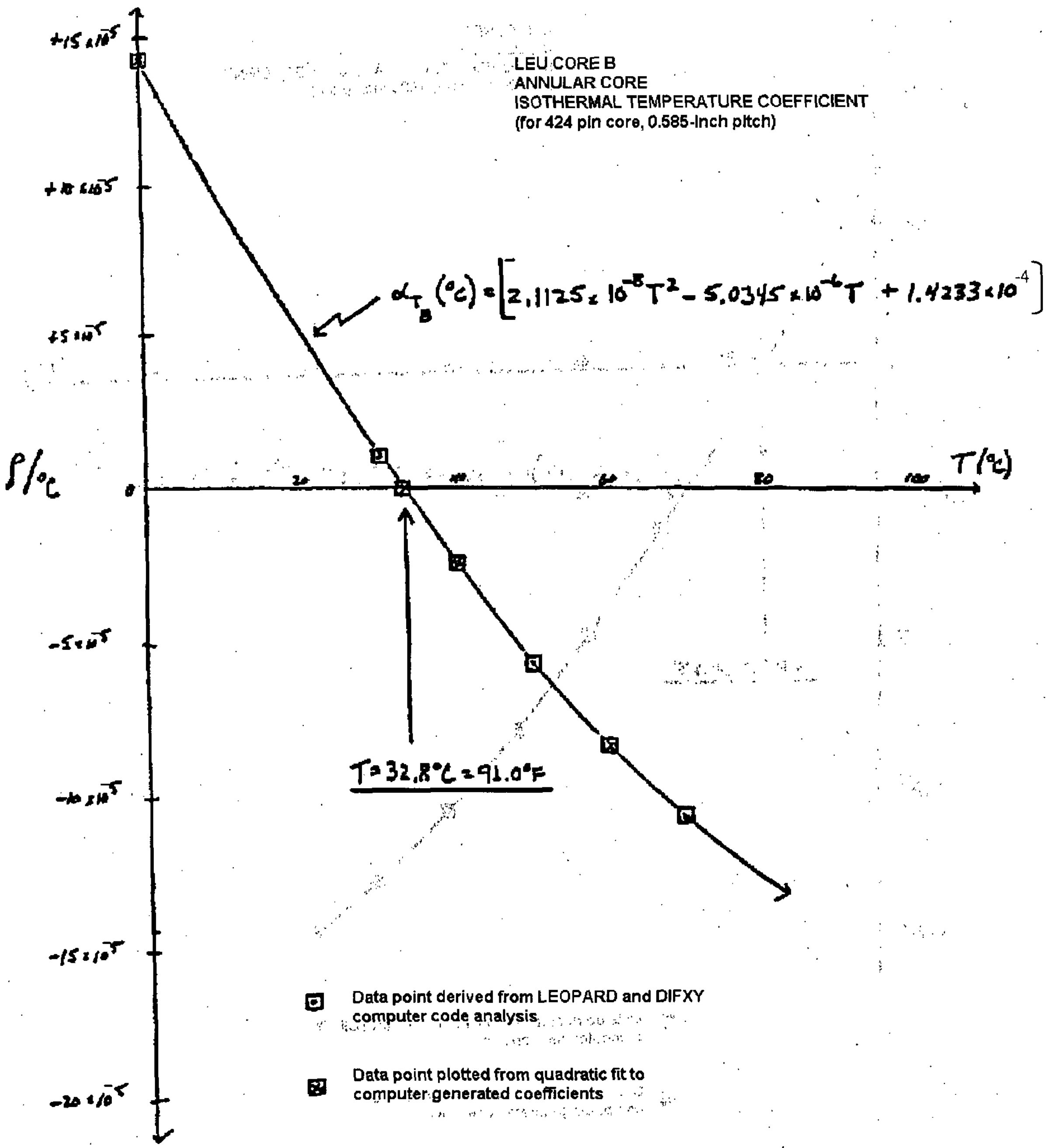
c. On Figure 13.3 the final exponent (-4) of the equation is missing, possibly a photocopying artifact. Please correct.

RPI Response – Figure will be changed (see revised Figure 13.3 below) in the next SAR update.

Revised Fig. 13.2:



Revised Fig. 13.3:



Questions related to the Technical Specifications

General

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

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

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

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

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

General

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

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

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

1.2 Format

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

RPI Response – ANS 15.1-2007 is the consistent reference.

1.3 Definitions

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

RPI Response – These definitions have been added.

1.3.D The definition in the current TSs contains references to EuO3 in a stainless steel cermet, stainless steel, and an alloy of silver-cadmium-indium as possible materials for the control rod absorber sections. The proposed definition does not reference these materials. Confirm that these materials will not be utilized for the control rod absorber sections.

RPI Response – The only poison intended in the control rods is boron. The definition has been revised.

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

RPI Response – The definition has been revised to be consistent with ANS-15.1-2007.

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

RPI Response – Definitions are now provided for reactor operating, reactor shutdown and reactor secured. RPI believes these definitions are mutually exclusive and all inclusive.

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

RPI Response – Done, see item above.

1.3.P The proposed definition specifies the maximum permissible distance and travel time for the Licensed Senior Operator (LSO) on call as 30 miles or 60 minutes. The current definition specifies the maximum permissible distance and travel time for the LSO on call as 15 miles or 30 minutes. Provide justification for the increases in the permissible distance and travel time. (See SAR RAI 12.6)

RPI Response – A justification is provided in the response to RAI 12.6.

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

RPI Response – The definition has been changed to agree with ANS-15.1-2007.

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

RPI Response – Both definitions have been revised to be consistent with ANS-15.1-2007.

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

RPI Response – See response to RAI 1.2 above.

2.0 Safety Limits and Limiting Safety System Settings

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

RPI Response – The original safety limit has been deleted. No safety limit is provided in the new Technical Specifications. This is adequate according to ANS-15.1-2007.

2.1 (B) 10 CFR 50.36c(1)(i)(A) requires safety limits "upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity." TS 2.1 does not adequately address protection of the fuel cladding integrity. Provide analysis that shows that no material degradation of the fuel cladding will occur if the fuel pellet temperature is limited to 2000 °C. Otherwise, revise the safety limit and provide analysis or discussion that shows the new safety limit will reasonably protect the integrity of the fuel and the cladding.

RPI Response – The accident analysis of SAR, Chapter 13, shows that no significant reactor temperature increase occurs for the accident conditions postulated to be the most severe. Thus this accident does not challenge the stainless steel exterior cladding of the fuel pins. As permitted by ANS-15.1-2007, no safety limit has been given in the Technical Specifications. The accident conditions stipulate a maximum excess reactivity and this is restricted by making excess reactivity a limiting condition for operation. In conjunction with the limiting safety system settings, the analyzed accident does not damage the fuel pin.

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

RPI Response – Since the safety limit does not involve fuel temperature, there is no need for this reference.

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

RPI Response – The SAR, Chapter 13, section 132, describes the accident conditions and the consequences.

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

RPI Response – These terms have been removed.

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

RPI Response – Deleted from TS 2.2.

3.0 Limiting Conditions for Operation

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

RPI Response - The moderator temperature was given as 20 ° C in the SAR for reference purposes only. The temperature of the moderator was not an initial condition or variable in the accident analysis, and therefore no TS amendment is needed to specify an LCO for maximum coolant temperature.

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

RPI Response - Boiling of reactor moderator is not a credible scenario given the reactor power (< 100 W) and volume (2000 gallons). The most severe accident does not demonstrate that voids form, nor is any void coefficient assumed in the analysis. Therefore the void coefficient of reactivity is not limited by the technical specifications.

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

RPI Response – Technical Specifications were changed.

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

RPI Response - The footnote has been removed. With the current instrumentation, the interlock can't be bypassed.

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

RPI Response - Table 2 of the current Technical Specification is correct with Amendment 11.

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

RPI Response - The proposed TS requires a shutdown reactivity of > $1 with all four control rods fully inserted. See Section 3.2. The accident analysis in the SAR has been updated to use this value.

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

RPI Response - A paragraph will be added to the SAR, section 13.2, on the next update stating that maximum reactivity worth of a clean fuel pin are set to prohibit the possibility of exceeding the excess reactivity limitation. TS 3.2.1 has been amended to reference that section.

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

RPI Response – The confusion here is due to omission of a $. See the revised Table 13.2 entry provided with the response to SAR RAI 13.5. The maximum allowable reactivity worth of a fuel pin is $0.20, as written in TS 3.2.1. The SAR will be updated.

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

RPI Response – The limit on maximum control rod reactivity rate has been deleted as unnecessary. The response to RAI 4.4 discusses why a high rate is not credible. Further, the analyzed accident does not initiate from control rod motion, but from a step insertion of reactivity far greater than would occur from control rod motion. Therefore, no limit on control rod reactivity insertion rate is needed.

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

RPI Response - The accident analysis in the SAR will be updated to include the safety system response time. The magnet release time of 50 ms does not include the safety system response time. The Technical Specification has been revised.

Note that the accident scenario of SAR, Chapter 13, provides 1.5 seconds for control rod scram, but allows a full 5 seconds before reactivity is inserted. The revised Technical Specification allows 900 milliseconds for rod drop time, initiated by a manual scram signal, and 600 milliseconds for instrument response. Magnet release is measured as part of the rod drop time. The revised Technical Specification does not require a separate measurement of magnet release time.

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

RPI Response - The auxiliary scram function of the moderator dump valve is not included in the accident analysis in the SAR. However, it is easy to maintain, and provides protection in the extremely unlikely event of multiple stuck control rods. Therefore, it is being maintained as a required safety system in the TS. The requirement that negative reactivity be added within one minute of the dump valve opening provides a means of periodically assessing the viability of the dump valve as a secondary scram mechanism.

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

RPI Response - A paragraph has been added to the TS explaining the basis for the limit on water height. The requirement that the water level be no greater than 10 inches above the top grid of the core is a means of ensuring that the time taken to insert negative reactivity via the secondary scram is not greater than the time measured during surveillances. The SAR will be revised on the next update.

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

RPI Response - The reactor is under-moderated and operates at low thermal power (< 100 W), and therefore having a low moderator height is not a reactor safety concern. Measurements with water level lowered are performed with a specific experimental procedure.

3.2.8 See TS RAI 3.2.5.

RPI Response – See the response to 3.2.8.

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

RPI Response - The table has been updated to include more specific information on the interlocks:

Line Voltage to Recorders < 100 V
Reactor Period <= 15 seconds
Source Range Counts <= 2 counts per second

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

RPI Response - A gauge located in the reactor control room provides indication of the water height. The Startup Procedures required that water height in the reactor tank be visually verified after the fill pump is turned OFF, and prior to the operation of the reactor, to verify that the water level is at the desired height. Therefore, it is not credible for the water height to exceed 10" above the core grid without the knowledge of the operator. An interlock is not necessary. Further, a lower water level does not exacerbate the analyzed accident.

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

RPI Response - The safety conclusions in the SAR do not take credit for a thermal power level of less than 100 watts. The administrative limit of 15 watts is in place because the reactor experiments do not require a higher power level, and the lower power limit helps keep exposure to personnel handling fuel and working in the control room and control room hallway ALARA.

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

RPI Response - The integrated annual thermal power limit of 200 kWh is not used in the safety analysis. This limit is used to ensure that the annual public exposure does not exceed expectations due to abnormal power and duration of operation for a given year. Note that the proposed Technical Specification has changed this limit.

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

RPI Response - TS 3.3.1.c has been amended to remove the requirement for particulate air monitoring. See RAI 7.6 for justification.

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

RPI Response - This section will be updated to state that at a minimum:

- During normal operation, a calibrated and operational portable survey meter capable of measuring ambient radiation exposure will be available
- 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 will be available to check for personnel or area contamination.

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

RPI Response – See the Technical Specifications.

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

RPI Response - The limit on the maximum reactivity insertion rate due to an oscillating experiment is not a safety limit, and therefore is not treated in the SAR. This limit is meant to maintain controllability of the reactor during the performance of experiments. The bases has been revised.

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

RPI Response - The specification given by TS 3.4.5 is a blanket statement that pertains to all experiments, whether singly-, doubly-, or un-encapsulated. TS 3.4.8 establishs specific limits for encapsulated experiments. For clarity, 3.8.5 of the revised Technical Specification has been amended to read the following:

> Experiments shall not contain materials which 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 Technical Specifications 3.8.8.

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

RPI Response -RPI does not see a need to specify a universal limit for maximum activity of radioactive material that can be placed in the core for any experiment. The NSRB must review all experiments prior to implementation. This review considers failure of any experiment containing radioactive material and ensures that failure will not compromise regulatory exposure limits. There is also no need to treat singly encapsulated and doubly encapsulated experiments differently, so TS 3.4.8 and TS 3.4.9 (renumbered as 3.8.8 in the revised Technical Specification) have been combined into a single condition.

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

RPI Response – see response to 3.4.8(A) above

3.4.9 (A) See TS RAI 3.4.8 (A)

RPI Response – see response to 3.4.8(A) above

3.4.9 (B) See TS RAI 3.4.8 (B)

RPI Response – see response to 3.4.8(A) above

4. Surveillance Requirements

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

RPI Response – RPI judges that no Technical Specification surveillance is necessary. Discussion will be included in the SAR on the next update.

The fuel is stored dry and only wetted infrequently, usually once a week for a few hours. The fuel pins do not show any sign of corrosion after 20 years of operation. Pins are removed from service if they become bowed because the pins are then difficult to align with the upper and lower support plates. Frequent radiological surveys for loose surface contamination would detect a leaking fuel pin. RPI notes that twenty years of use of these pins has never detected any leaking pins. If a pin were severely damaged during handling, the event would be treated as a radiological casualty and surveys taken to determine if the pin were now leaking.

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

RPI Response – The term known core is defined in the Technical Specification.

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

RPI Response – See the revised Technical Specification.

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

RPI Response – See the revised Technical Specification. The experimental procedures developed for the facility are those also used to determine that the reactor parameters of a new core configuration meet the requirements of Section 4.1. The SAR will be updated.

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

RPI Response – The correct word is "bank". The typographical error has been corrected

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

RPI Response – See the responses with the referenced questions.

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

RPI Response – See the revised Technical Specification.

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

RPI Response – The statement has been corrected to refer only to items corresponding to the discussion of bases.

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

RPI Response – The bases are poorly worded and have been revised to show how redundancy is achieved. There are three instrument channels that use ion chambers and all three have a high current scram. One of these, the Log-N channel also has a fast period scram. The Log count rate channel is driven by a BF3 detector and provides the rod outmotion interlock on low counts. Redundancy is claimed based on the ability to still provide a high current scram even if any one of the ion chamber channels fail. Note that the SAR analysis assumes the failed channel is the Log-N channel so both a high current and a fast rate scram are unavailable. The high current scram from either linear power channel terminates the accident. The accident is independent of power level so failure of the low source counts instrument does not increase the severity of the accident. One could argue that the log count rate channel is not a safety channel, however it is listed here as a necessary channel for operation to assist the operator as the reactor approaches criticality.

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

RPI Response – The reference has been changed to 3.2, Item 5. Renumbering occurred in this section.

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

RPI Response – Periodic instrument calibration requirements have been added to the Technical Specifications.

5. Design Features

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

RPI Response – This typographical error has been corrected. The tank is stainless steel, not lined with stainless steel.

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

RPI Response – The text describing fuel other than SPERT (F-1) pins has been removed. No other fuel is at the facility.

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

RPI Response – The correct word is "one". The typographical error has been corrected. The correct reference is to 5.4.4 vice 5.4.4.

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

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

RPI Response – The correct word is "on". The typographical error has been corrected. The correct reference is to 5.6 vice 5.5.

6. Administrative Controls

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

12 Conduct of Operations

12.1 Organization

12.1.1 Structure - Responsibility for the safe operation of the reactor facility is vested within the chain of command shown in Figure 12.1.

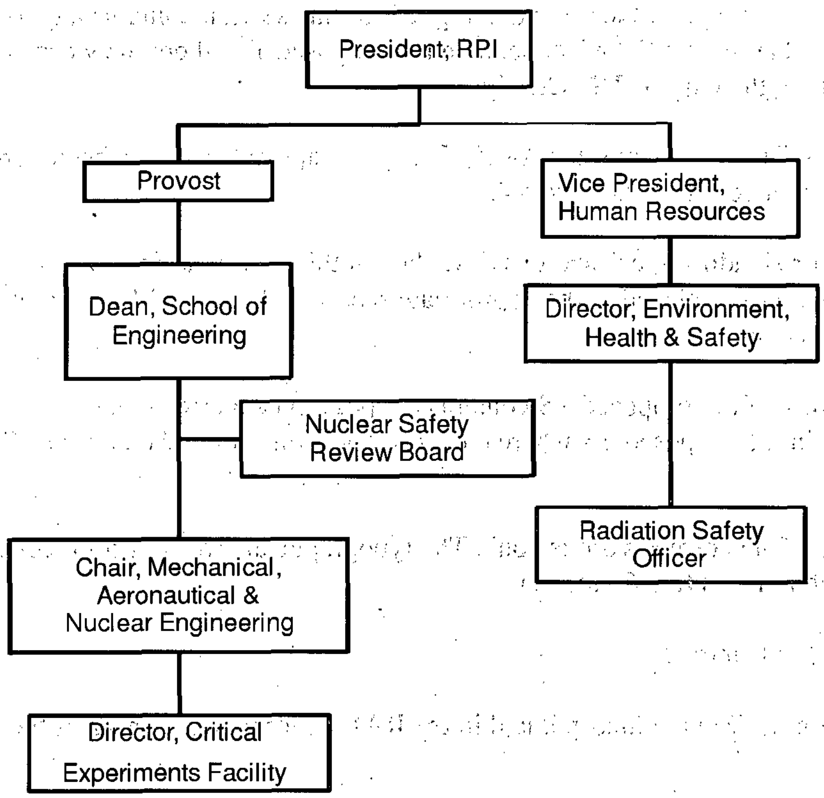


Figure 12.1: Facility Organization

12.1.2 <u>Responsibility</u>

12.1.2.1 Level 1: The Dean, School of Engineering, is responsible for the facility license and appoints the Chair, Nuclear Safety Review Board.

12.1.2.2 Level 2: The Facility Director is responsible for facility administration and safety. The Facility Director reports to the Chair, Mechanical, Aerospace and Nuclear Engineering for administrative purposes.

12.1.2.3 Level 3: The Operations Supervisor is responsible for the day-to-day operation of the facility and reports to the Facility Director

12.1.2.4 Level 4: Licensed operators and senior operators are the operating staff and report to the Facility Director for administrative purposes.

12.1.2.5 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.

12.1.3 Staffing

12.1.3.1 The minimal staffing when the reactor is not shutdown shall be:

12.1.3.1.1 A senior reactor operator licensed pursuant to 10 CFR 55 present at the controls.

12.1.3.1.2 One other person in the control room certified by the Operations Supervisor as qualified to activate manual scram and initiate emergency procedures. A second senior reactor operator or a reactor operator licensed pursuant to 10 CFR 55 fulfills this requirement.

12.1.3.1.3 A licensed senior reactor operator shall be present or readily available on call. This is defined as being with 60 minutes normal travel time, or 25 miles, whichever is more limiting. The time for the on call operator to arrive is based on reasonable response to potential needs that can't be satisfied by phone. It is considered unlikely that a second operator would actually need to arrive that quickly since the reactor can be placed in the safe shutdown mode in less than a minute.

12.1.3.2 The identity of and method for rapidly contacting the licensed senior operator on duty shall be known to the operator.

12.1.3.3 No staffing is required when the reactor is in secure shutdown.

12.1.3.4 The minimal staffing when the reactor is shutdown but not secure shutdown shall be a senior reactor operator at the facility and a second senior reactor operator present or readily available on call.

12.1.3.5 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 management personnel, radiation protection personnel and other RCF Staff.

12.1.3.6 Events requiring the direction of the Operations Supervisor:

12.1.3.6.1 All fuel or control rod relocations within the reactor core.

12.1.3.6.2 Recovery from unplanned or unscheduled shutdown.

12.1.4 Selection and Training of Personnel

New reactor operators and senior reactor operators are selected from interested students enrolled in classes that take place at the RCF. The Operations Supervisor is responsible for the operator training assisted by other RCF staff. The Operator Requalification Program meets the regulations in 10 CFR 55.

12.1.5 Radiation Safety

Radiation safety aspects of routine facility operation are typically performed by members of the RCF staff who receive training from the RSO to perform those tasks. Thus radiation surveys to verify normal radiation levels during reactor operation, fuel handling, or experiments will be conducted by the RCF Staff. The RSO is available for assistance if needed. The RSO also conducts periodic contamination surveys and maintains and monitors personnel exposure records.

12.2 Review and Audit Activities

The Nuclear Safety Review Board (NSRB) provides independent review and audits facility activities. The Dean, School of Engineering, appoints the NSRB Chair. Some members of the NSRB are appointed by virtue of their position, the Facility Director and RSO are examples of this. Other members of the NSRB are appointed by their management. The NSRB then reports to the Dean, School of Engineering. The NSRB Charter provides additional details.

12.3 Procedures

12.3.1 Written operating procedures are used for the following:

- 12.3.1.1.1 Reactor Pre-Startup
- 12.3.1.1.2 Reactor Operations, including conduct of experiments
- 12.3.1.1.3 Surveillances
- 12.3.1.1.4 Emergencies
- 12.3.1.1.5 Radiation Protection

12.3.2 Procedures are developed by the RCF Staff in response to a planned need for a new or revised procedure. Existing procedures are consulted and revised if possible to meet the need for a new procedure. This process is supervised by the Operations Supervisor. A proposed new procedure is reviewed by the Operations Supervisor and the Facility Director to determine the need for NSRB review and approval. Minor changes that do not affect the safe operation of the reactor may be approved for use by the Operations Supervisor. Procedures which do not meet these criteria are presented to the NSRB for approval. Approved procedures are put into use after updating the list of approved procedures. This list informs the operators which procedures, by name and version, are approved for use. The list is updated and approved by the Operations Supervisor and is posted at or near the reactor operating console.

12.3.3 Radiation protection procedures are maintained by the Office of Radiation and Nuclear Safety, with the approval of the Radiation and Nuclear Safety Committee. The Radiation Safety Manual addresses the Program, Policy, and Organization of the radiation safety program, the Radiation Safety Training program, radioactive waste management, dosimetry and radiation monitoring, instrumentation calibration, and the ALARA program. The Radiation Safety Officer ensures that the Radiation Safety Manual addresses each of the recommendations in ANSI/ANS-15.11-1993, and distributing updates to the Critical Facility Director and Nuclear Safety Review Board.

12.4 Required Actions

12.4.1 Action to be taken in Case of Safety Limit Violations – No action steps are provided for this since there is no identified safety limit. Safety limits are not required for reactors without engineered safety systems provided that the accident analysis shows that there is no damage to the fuel cladding. Chapter 13 analyzes the potential accidents to the RCF and concludes that no fuel clad damage will occur for the most severe accident.

12.4.2 Action to be Taken in the Event of an Reportable Occurrence

12.4.2.1 The reactor shall be shut down. Operations shall not be resumed unless authorized by the Chair, NSRB.

12.4.2.2 Occurrence shall be reported to the Facility Director or designated alternate, the NSRB and to the Nuclear Regulatory Commission not later than the following working day by telephone and confirmed in writing to licensing authorities, to be followed by a written report that describes the circumstances of the event within 14 days of the event.

12.4.2.3 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.

12.5 Reports

Reports include annual operating reports that describe the activities for the previous year and non-periodic reports that describe important changes in the facility or facility management. NUREG-1537, Part 1, and ANS-15.1-2007 provide guidance for the details of the information that should be reported. This guidance has been incorporated in the Technical Specifications.

12.6 Records

Facility records are required to be maintained for specific periods of time depending upon the type of record. NUREG-1537, Part 1, and ANS-15.1-2007 provide guidance for the details of the retention time. This guidance has been incorporated in the Technical Specifications.

12.7 Emergency Planning

12.7.1 The RCF Emergency Plan describes the Critical Facility emergency organization and includes the responsibilities and authority with a line of succession for key members of the emergency organization. The emergency organization described in the plan ensures that emergency management will be available to meet any foreseeable emergency at the research reactor. Additionally, the plan describes the criteria for the termination of an emergency, authorization for reentry, and establishes limits of exposure to radiation in excess of normal occupational limits for emergency team members for life saving and corrective actions to mitigate the consequences of an accident.

12.7.2 Two emergency classes are described for the Critical Facility. These classes are based upon credible accidents associated with the reactor operations and other emergency situations that are non-reactor related but could affect routine reactor operations. The emergency classes are Personnel Emergency and Emergency Alert. Each class is associated with specific Emergency Action Levels (EALs) for activating the emergency organization and initiating protective actions appropriate for the emergency event in process. The Emergency Planning Zone (EPZ) is the area within the Critical Facility building. Predetermined protective actions for the EPZ include radiation surveys to locate areas and levels of radioactive contamination, personnel evacuation should this become necessary and personnel accountability.

12.7.3 The emergency facilities and equipment available for emergency response include a designated Emergency Support Center, radiological monitoring systems, instruments and laboratory facilities for continually assessing the course of an accident, first aid and medical facilities and communications equipment. The provisions for maintaining emergency preparedness include programs for training, retraining, drills, plan review and updates, and equipment inventory and calibrations.

12.8 Security Planning

The RCF has established and maintains a program to protect the reactor and fuel and to ensure its security. Both the physical security plan and the staff's evaluation are withheld from public disclosure under 10 CFR 2.790(d)(1) and 10 CFR 9.5(a)(4). The current Security Plan was last revised in 2006.

12.9 Quality Assurance

Quality Assurance is achieved via extensive documentation and periodic interaction with the Nuclear Safety Review Board (NSRB). All operations and experiments must follow written procedures that have been approved by the NSRB.

12.10 Operator Training and Requalification

Operator training and requalification programs have been approved by the Nuclear Regulatory Commission.

12.11 Startup Plan

A startup plan is not necessary for facility license renewal. The facility is not undergoing any changes that would require such a plan.

12.12 Environmental Reports

The facility has existed up to the present without having any significant effect on the environment. No future changes to the facility are anticipated that would result in an increased effect on the environment. The facility has no off-site environmental monitoring requirements. The annual operating report includes data on facility discharges and radiation monitoring data from site exclusion boundary dosimetry.

TECHNICAL SPECIFICATIONS

CRITICAL EXPERIMENTS FACILITY

RENSSELAER POLYTECHNIC INSTITUTE

July 2008

Approved: Michael Z. Podowski

Dr. Michael Podowski, Chair
Nuclear Safety Review Board

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Technical Specifications

1. INTRODUCTION

1.1 Scope

The following constitute the Technical Specifications for the 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-5.1-2007.

1.3 Definitions

certificate or charter: See license.

certified: See licensed.

Class A reactor operator: See senior reactor operator.

Class B reactor operator; See reactor operator.

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 out-put 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. The absorbers, when fully inserted, shall extend above the top and to within one inch of the bottom of the active core.

core configuration: The core configuration includes the number, type, or arrangement of fuel elements, reflector 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 ($k_{eff} = 1$) at reference core conditions or at a specified set of conditions.

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.

facility-specific definitions: Facility-specific definitions are those definitions unique to a specific facility.

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 inserted and one rod stuck in the full out position
(3) reactivity worth of most reactive fuel pin

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

licensed: See licensee.

licensee: An individual or organization holding a license.

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.

owner or operator: See licensee.

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.

permit: See license.

protective action: Protective action is 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 on the bottom.

reactor operator: An individual who is licensed to manipulate the controls of a reactor.

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

reactor secured: A reactor is secured when

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

reactor shutdown: The reactor is shut down if all control rods are inserted and it is subcritical by at least one dollar 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 on the bottom.

research reactor: A research reactor is defined as a device designed to support a self-sustaining neutron chain reaction for research, development, educational, training, or experimental purposes and that may have provisions for the production of radioisotopes.

research reactor facility: Includes all areas within which the owner or operator directs authorized activities associated with the reactor.

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 unless prompt remedial action is taken;
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

responsible authority: A governmental or other entity with the authority to issue licenses, charters, permits, or certificates.

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 bottomed.

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 as a result 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: An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

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 the nonscramable rods in their most reactive positions 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 fully inserted, including the reactivity of installed experiments.

supervisory reactor operator: See senior reactor operator

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:

- Annual (interval not to exceed 15 months).
- Semiannual (interval not to exceed seven and one-half months).
- 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 – None

Bases

The Safety Analysis Report (SAR) evaluates all potential accidents and identifies an unplanned or uncontrolled reactivity addition as the most severe. Analysis of this type of accident has been performed for an addition of 60 cents and acceptable core performance was demonstrated. See SAR Section 13.2.

2.2 Limiting Safety System Settings

Applicability

Applies to the settings to initiate protective action for instruments monitoring parameters associated with the reactor power limits.

Objective

To assure protective action before safety limits are exceeded.

Specification

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

Maximum Power Level	100 watts
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 setpoint is used in the safety analysis with the assumption that initial power is at 100 watts indicated power.

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 increase and energy deposition 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.

Objective

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

Specifications

The excess reactivity of the reactor above cold, clean critical shall not be greater than 0.60$.

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 60 cents of reactivity above critical.

3.2 Reactor Control and Safety Systems

Applicability

Applies to all methods of changing core reactivity available to the reactor operator.

Objective

To assure 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 clean 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 inserted shall be $1.00.
3. The total control rod drop time for each control rod from its fully withdrawn position to its fully inserted position shall be less than or equal to 1.5 seconds. This time shall include a maximum instrument response time of 600 milliseconds. Instrument response may be measured separately from rod drop time if desired. If the total time is measured and is less than required, then instrument response time need not be separately measured to determine if the 600 millisecond time is met.
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 operating during the reactor operation are listed in Table 1.
7. After a scram, the moderator dump valve may be re-closed by a senior reactor operator if the cause of the scram is known, all control rods are verified to have

scrammed and it is deemed wise to retain the moderator shielding in the reactor tank.

8. The interlocks that shall be operable during reactor operations are listed in Table 2.

9. 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
Start-up:	2 cps - 10^4 cps	Log Count Rate	1	Minimum Flux Level
Power:	10^{-3} – 10^{-11} amps	Linear Power	2	High Neutron Level Scram
	10^{-3} - 10^{-14} amps +999 – -999 seconds	Log-N; Period	1	High Neutron Level and Period Scram
		Manual Scram[a]	2	Reactor Scram
		Building 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 reactor, 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 provided that no other scram channels are bypassed.

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 On	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 60 cents excess reactivity in any experiment that removes a fuel pin. Limiting worth to 20 cents 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 insertion time of less than 1.5 seconds for each control rod from its 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. Experience shows that rod drop time of less than 900 milliseconds is typical, therefore 600 milliseconds of the total 1.5 second drop time is allocated to instrument response.

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.

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.

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

3.4 Containment or confinement – None required

3.5 Ventilation Systems – None required

3.6 Emergency Power – None required

3.7 Radiation Monitoring

Applicability

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

Objective

The purpose of these specifications is to ensure that adequate monitoring is available to preclude undetected radiation hazards or uncontrolled release of radioactive material.

Specifications

1. The minimum complement of radiation monitoring equipment required to be operating for reactor operation shall include:
   a. A criticality detector system that monitors the main fuel storage area and also functions as an area monitor. This system shall have a visible and an audible alarm in the control room.
   b. 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.
   c. The radiation monitors required by 3.3.1 a and b, may be temporarily removed from service if replaced by an equivalent portable unit.
2. During normal operation, a calibrated and operational portable survey meter capable of measuring ambient radiation exposure will be available.
3. 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 will be available to check for personnel or area contamination.

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 instruments to measure the amount of particulate activity in the reactor room air ensures continued compliance with the requirements of 10 CFR Part 20. The availability of required portable monitors provides assurance that personnel will be able to monitor potential radiation fields before an area is entered and during fuel handling.

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

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 and approved by the Operations Supervisor. Experiments that fall in the general category, but with minor deviations from those previously performed, 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 functions of the nuclear instrumentation.

3. For movable experiments with an absolute worth greater than $.35, the maximum reactivity change for withdrawal and insertion shall be $.20/sec. Moving parts worth less than $.35 may be oscillated at higher frequencies in the core.

4. 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 $.60.

5. Experiments shall not contain materials which 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 Technical Specifications 3.8.8.

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

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

8. All experiments containing radioactive material shall be evaluated for their potential release of airborne radioactivity and limits shall be established for the permissible concentration of radioisotopes in the experiments such that a complete release of all gaseous, volatile, or particulate constituents to the reactor room air would not exceed the limitations for exposure of individuals in restricted or unrestricted areas.

Bases

The basic experiments to be performed in the reactor programs are described in the Safety Analysis Report (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, produce airborne activity, or cause a corrosive attack on the fuel cladding or primary coolant system.

Specification 8 will ensure that the quantities of radioactive materials contained in experiments will be so limited that their failure will not result in exposures to individuals in restricted or unrestricted areas to exceed the maximum allowable exposures stated in 10 CFR 20. The restricted area maximum is defined in 10 CFR 20.1201 through 10 CFR 20.1204. The unrestricted area maximum is defined in 10 CFR 20.1301 and 10 CFR 20.1302.

3.9 Facility-specific Limiting Conditions for Operations

All fuel transfers shall be conducted under the direction of a licensed senior reactor operator.

Operating personnel shall be familiar with health physics procedures and monitoring techniques, and shall monitor the operation with appropriate radiation instrumentation.

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

For a known core, up to a quadrant of fuel pins may be removed from the core or a single stationary fuel pin be replaced with another stationary 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 and the dump valve is not bypassed.
5. The critical rod bank position is checked after the operation is complete.

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:

a. excess reactivity;

b. worth of most reactive fuel pin;

c. reactor power measurement;

d. 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 Safety Analysis Report, 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 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 on the bottom.

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 Specification 3.2.

Specifications

1. The total control rod drop time, including instrument response time shall be measured semiannually to verify that the requirements of Specification 3.2, Item 3, are met.
2. The moderator-reflector water dump time shall be measured semiannually to verify that the requirement of Specification 3.2, Item 4, is met.
3. All safety system channels shall be calibrated annually.
4. A channel test of the safety system channels (intermediate, and power range instruments) and a visual inspection of the reactor shall be performed daily prior to reactor startup. The interlock system shall be checked daily prior to reactor startup to satisfy rod drive permit. These systems shall be rechecked following a shutdown in excess of 8 hours.
5. The moderator-reflector water height shall be checked visually before reactor startup to verify that the requirements of Specification 3.2, Item 5, are met.
6. These tests 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 Specification 3.2, Items 3 and 4.

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.

4.3 Coolant Systems

Applicability

These specifications 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

Analyze moderator for radioactivity prior to discharge to the environment.

Bases

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.

4.4 Containment or Confinement – None required

4.5 Ventilation Systems – None required

4.6 Emergency Power – None required

4.7 Radiation Monitoring

Applicability

These specifications apply to the surveillance of the area radiation monitoring equipment and all portable radiation monitoring instruments.

Objective

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

Specification

The criticality detector system, 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.

Bases

Experience has demonstrated that calibration of the criticality detectors 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.

4.8 Experiments – None required

4.9 Facility-specific Surveillance Requirements – None required

5. DESIGN FEATURES

5.1 Site and Facility Description

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 facility 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.

5.2 Reactor Coolant System

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 fail safe 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 fast fill rate of about 50 gpm is provided. 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.

5.3 Reactor Core and Fuel

The reactor core shall consist of uranium fuel in the form of 4.81 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. The core shall consist of all SPERT (F-1) fuel described in 5.4.3.

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.

Core fuel pins to be utilized are 4.81 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. Figure 4.5 of the SAR depicts a single fuel pin and its pertinent dimensions. NUREG-1281 describes these fuel pins in additional detail.

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 as depicted in Figure 4.6 of the SAR. The combination of the four rods must meet the values given in Table 13.2 of the SAR, with regard to reactivity with one stuck rod and shutdown reactivity.

5.4 Fissionable Material Storage

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 no more than 1 kg fuel per tube mounted on a steel wall rack. A storage tube in the storage vault cannot contain more than 15 SPERT (F-1) fuel pins at any time. 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.

6. ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The organization for the management and operation of the reactor facility shall include the structure indicated in Figure 6.1.

Level 1: The Dean, School of Engineering, appoints the Chair, Nuclear Safety Review Board.

Level 2: The Facility Director reports to the Chair, Mechanical, Aerospace and Nuclear Engineering for administrative purposes.

Level 3: The Operations Supervisor reports to the Facility Director.

Level 4: Licensed operators and senior operators are the operating staff and report to the Facility Director for administrative purposes.

6.1.2 Responsibility

The Dean, School of Engineering, is responsible for the facility license and appoints the Chair, Nuclear Safety Review Board. The Facility Director is responsible for facility administration and safety. The Operations Supervisor is responsible for the day-to-day safety and operation of the facility.

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.

6.1.3 Staffing

(a) The minimal staffing when the reactor is not shutdown as described in these specifications shall be:

1) An operator or senior operator licensed pursuant to 10 CFR 55 present at the controls.
2) One other person in the control room certified by the Reactor Supervisor as qualified to activate manual scram and initiate emergency procedures. This person is not required if an operator and a senior operator are in the control room.
3) A licensed senior operator shall be present or readily available on call. The identity of and method for rapidly contacting the licensed senior operator on duty shall be known to the operator.

(b) The minimal staffing when the reactor is shutdown, but not in safe shutdown is a senior reactor operator in the control room and a second senior reactor operator present or readily available on call.

(c) 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:
1) Management personnel.
2) Radiation safety personnel.
3) Other operations personnel.

(d) Events requiring the direction of the Operations Supervisor:
1) All fuel or control rod relocations within the reactor core unless the activity is part of an approved experiment.
2) Recovery from unplanned or unscheduled shutdown.

6.1.4 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 Operator License for the RPI Critical Facility. Years spent in baccalaureate or graduate study may be substituted for operating experience on a one-for-one basis up to a maximum of two years.

6.2 Review and Audit

A Nuclear Safety Review Board (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.

6.2.1 Composition and Qualifications

The NSRB shall be appointed by the Dean School of Engineering in accordance with the NSRB Charter.

6.2.2 Charter and Rules

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

(a) The Board shall meet at least semiannually.
(b) The quorum shall consist of not less than a majority of the full Board and shall include the Chairman or his designated alternate.

(c) Minutes of each Board meeting shall be distributed to the Dean, NSRB members, and such others as the Chairman may designate.

6.2.3 Review Function

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

(a) Proposed experiments and tests utilizing the reactor facility that are significantly different from tests and experiments previously performed at the facility.
(b) Reportable occurrences.
(c) Proposed changes to the Technical Specifications and proposed amendments to facility license.
(d) Operating, Emergency and Surveillance procedures.

6.2.4 Audit Function

(a) 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.
(b) Reactor operations and reactor operational records for compliance with internal rules, regulations, procedures, and with licensed provisions;
(c) Existing operating procedures for adequacy and to ensure that they achieve their intended purpose in light of any changes since their implementation;
(d) Plant equipment performance with particular attention to operating anomalies, abnormal occurrences, and the steps taken to identify and correct their use.

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.

6.3.1 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 Rensselaer Polytechnic Institute 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 in 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 check lists, 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) Implementation of the facility security plan.
6) Implementation of facility emergency plan in accordance with 10 CFR 50, Appendix E.
7) Maintenance procedures that could have an effect on reactor safety.

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 and subsequently reviewed by the Nuclear Safety Review Board.

6.5 Experiment Review and Approval

1) All new experiments or classes of experiments that might involve an unreviewed safety question shall be reviewed by the Nuclear Safety Review Board. NSRB approval shall ensure that compliance with the requirements of the license technical specifications and 10 CFR50.59 shall be documented. A licensee shall obtain a license amendment pursuant to Sec. 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

   (a) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the final safety analysis report;
   (b) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the final safety analysis report;
   (c) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report;
   (d) Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report;
   (e) Create a possibility for an accident of a different and potentially more severe than any previously evaluated in the final safety analysis report;

(f) Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the final safety analysis report;
(g) Result in a design basis limit for a fission product barrier as described in the SAR being exceeded or altered; or
(h) Result in a departure from a method of evaluation described in the SAR used in establishing the design bases or in the safety analyses.

2) Substantive changes to previously approved experiments shall be made only after review and approval in writing by NSRB. 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 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 Nuclear Regulatory Commission not later than the following working day by telephone and confirmed in writing to licensing authorities, 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.

6.7 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.

6.7.1 Operating Reports

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

(a) Operations Summary. A summary of operating experience occurring during the reporting period that relates to the safe operation of the facility, including:
1) Changes in facility design;
2) Performance characteristics (e.g., equipment and fuel performance);
3) Changes in operating procedures that relate to the safety of facility operations;

4) Results of surveillance tests and inspections required by these Technical Specifications;
5) A brief summary of those changes, tests, and experiments that require authorization from the Commission pursuant to 10 CFR 50.59, and;
6) Changes in the plant operating staff serving in the following positions:
    a) Facility Director;
    b) Operations Supervisor;
    c) RSO;
    d) Nuclear Safety Review Board Members.

(b) Power Generation. A tabulation of the integrated thermal power during the reporting period.

(c) 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.

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

(e) 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 Commission approval pursuant to the requirements of 10 CFR Part 50.59.

(f) 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.

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

(h) 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.

6.7.2 Special Reports

(a) Reportable Operational Occurrence Reports. Notification shall be made within 24 hours by telephone in accordance with 10CFR50.36(c)(7) followed by a written report in accordance with 10CFR50.36(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.

(b) Unusual events. A written report in accordance with 10CFR50.36(c)(5) shall be submitted as specified in 10CFR 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 Safety Analysis Report or in the bases for the Technical Specifications.

(c) Key changes in Organization. A written report in accordance with 10CFR50.36(c)(5) submitted as specified in 10CFR 50.4 shall be provided for any change in Level 1 or Level 2 personnel.

6.8 Operating Records

6.8.1 The following records and logs shall be maintained at the Facility or at Rensselaer Polytechnic Institute for at least five years.

(a) Normal facility operation (except retain checklists for one year) and principal maintenance operations;
(b) reportable occurrences;
(c) tests, checks, and measurements documenting compliance with surveillance requirements;
(d) experiments performed with the reactor;
(e) fuel shipments, inventories, and receipts;
(f) reactor facility radiation and contamination surveys;
(g) approved changes to operating procedures;
(h) records of NSRB meetings and audits

6.8.2 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.

6.8.3 The following records and logs shall be maintained at the Facility or at Rensselaer for the life of the Facility.

(a) gaseous and liquid radioactive releases from the facility;
(b) TLD environmental monitoring systems;
(c) radiation exposures for all RPI Critical Facility personnel (students and experimenters) and visitors;
(d) the present as-built facility drawings and new updated or corrected versions.

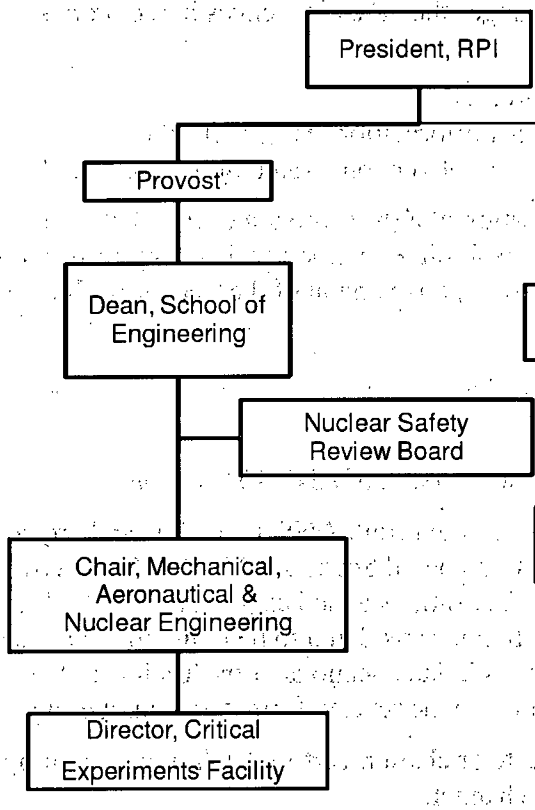

Figure 6.1: RCF Management Organization

Explanation of Changes to Technical Specifications

The changes explained herein are relative to the current Technical Specification, valid for Amendment 11 to the license. The changes then are differences in the proposed Technical Specification.

References used in the explanations are:

1. NRC Request For Additional Information, March 21, 2008
2. Technical Specifications (current version), dated September 2004

Note that Ref. 1 uses text and organization of a third version of the Technical Specifications, specifically the version submitted in 2002 with an application for relicensing. Except for responses to Ref. 1, this version of the Technical Specifications is no longer in use.

The changes relative to Ref. 2 are:

1. Section 1.1 added.
2. Section 1.2 added to show which version of ANS-15.1 was used.
3. Section 1.3 Definitions were imported from ANS-15.1-2007 and are not numbered as in Ref. 2. This is numbered Section 1.0 in Ref. 2. Specific definition changes are presented below. Definitions added as part of ANS-15.1-2007 are not discussed and no justification is considered to be required. Note that references to specific definitions in later sections of the Technical Specification have been revised to accommodate the revised format and numbering.
4. Definitions added for certificate or charter, certified, Class A reactor operator, Class B reactor operator and channel.
5. Definition of channel calibration changed to agree with ANS-15.1-2007
6. Definition of channel check changed to agree with ANS-15.1-2007.
7. Definition of channel test changed to agree with ANS-15.1-2007.
8. Definition of control rod assembly (Section 1.0.D) changed to control rod and reference to materials not in use was removed from the definition. There is no plan to use control rods other than those in the definition.
9. Definition added for core configuration.
10. Definition of excess reactivity changed to agree with ANS-15.1-2007.
11. Definition of experiment changed to agree with ANS-15.1-2007.
12. Added definitions for facility-specific definitions, known core, license, licensed, and licensee.
13. Removed definition of measuring channel (Section 1.0.G) as redundant with the definition of channel.
14. Definition of measured value changed to agree with ANS-15.1-2007.
15. Definition of movable experiment changed to agree with ANS-15.1-2007.
16. Added definition of owner or operator.

17. Definition of operable changed to agree with ANS-15.1-2007.

18. Definition of operating changed to agree with ANS-15.1-2007.

19. Added definition of permit.

20. Added definitions of protective action, reactor operating and reactor operator. Reference to xenon in the ANS-15.1-2007 definition of reactor operating was removed since the RPI reactor does not generate detectable xenon.

21. Definition of reactor safety systems changed to agree with ANS-15.1-2007.

22. Definition of reactor secured changed to agree more closely with ANS-15.1-2007 and reflect facility-specific conditions.

23. Definition of reactor shutdown changed to agree with ANS-15.1-2007 and to reflect facility-specific conditions..

24. Added definition of reactivity worth of an experiment.

25. The definition for readily available on call was revised to 60 minutes travel time or 25 miles. The previous definition was more restrictive, 30 minutes and 15 miles. The justification for this change was provided with the response to the Request for Additional Information, item 12.6 and is discussed in the Safety Analysis Report, Chapter 12.

26. Added definition of reference core condition.

27. Added definitions of research reactor and research reactor facility.

28. Added definition of responsible authority.

29. Definition of reportable occurrence revised to blend together the definition in ANS-15.1-2007 and the definition in Ref. 2.

30. Modified definition of safety channel to be consistent with replacement of the term measuring channel with the term channel.

31. Added definition of scram time.

32. Definition of secured experiment changed to agree with ANS-15.1-2007.

33. Added definition of senior reactor operator.

34. Added definition of shall, should and may.

35. Added definition of shutdown margin and supervisory reactor operator.

36. Changed definition of surveillance frequency to agree with and to reference ANS-15.1-2007. The surveillance intervals that are not used at the RPI reactor were removed from the list.

37. Definition of true value changed to agree with ANS-15.1-2007.

38. Definition of unknown core added.

39. Added definition of unscheduled shutdown.

40. Section 2.1 revised to state that no safety limit is required. This is consistent with ANS-15.1-2007 for reactors without engineered safety systems. The text for Applicability, Objective and Specification was deleted. A basis for not requiring a safety limit replaces the text explaining the basis for the current safety limit.

41. The limiting safety system setting for reactor power in Section 2.2 has been reduced to 100 watts, a more conservative setting than the 135 watts in the previous Technical Specification. Instrumentation changes make this possible since high current setpoints are based on specific current values rather than 90% of instrument full scale ranges. The value of 100 watts was chosen to agree with the license limit. The explanation of bases was changed for consistency with the reduced safety system setting.

42. The limiting safety system setting of 2.0 counts/sec was deleted. This instrument setting is unrelated to safety and is retained as an interlock shown later in the Technical Specifications. The associated discussion paragraph in Bases was also removed.

43. Sections 3.1 and 3.2 were renumbered as 3.2 and 3.1 respectively to agree with the organization of ANS-15.1-2007 and the constraint on excess reactivity moved to the new 3.1 from paragraph 3.1.1 of Ref. 2. Text for Applicability, Objective, and Bases of this specification was added.

44. Paragraph 3.1.3, Ref. 2 was deleted as unnecessary. The analyzed accident is independent of control rod reactivity rate. Further, responses to Ref. 1 show that the rate limited by Ref. 2 is not achievable. Reviewing changes to core configuration include a consideration for accidents more severe than that analyzed and this would include reactivity additions by control rods. Note that deletion of this item renumbered all following items.

45. Control rod drop time has been increased from that specified in paragraph 3.1.4 of Ref. 2. The most severe accident assumes a 1.5 second delay to insert the control rods. The current specification is shorter than required and may not accommodate instrument response. Further, there is no part of the accident analysis that relates to a magnet release time so this specification is eliminated. Associated discussion in Ref. 2, section 3.1 Bases was revised to agree with the revised drop time limitation.

46. Instrument ranges for Linear Power and Log N power on Ref. 2, section 3.1, Table 1 were changed to show the ranges of the current equipment. A range for Log N Period was added to this Table.

47. The Functions in Table 1 for Building Power changed to Reactor Scram. This better defines the action that occurs if Building Power fails.

48. Footnotes (a) and (b) have been deleted from Ref. 2, section 3.1, Table 1, as unnecessary and the remaining two footnotes, currently (c) and (d), renumbered as (a) and (b) respectively. Older instrumentation had bypass capability. The current instrumentation does not.

49. Footnote (a) has been deleted from Ref. 2, section 3.1, Table 2, as unnecessary. Older instrumentation had bypass capability. The current instrumentation does

not. In addition "less than" signs (<) were added to the specification for Reactor Period value and Neutron Flux value for completeness. An explanation for minimum neutron flux was added to section 3.1 Bases.

50. The interlock for Failure of Line Voltage to Recorders in section 3.1, Table 2, was reworded and a minimum value established as proposed by Ref. 1, 3.2.9(A). The value of 100 volts is based on the equipment operating manual.

51. The interlock for Reactor Tank Water Level section 3.1, Table 2, was deleted. This is not an automatic interlock, but an administrative control established by operating procedures. Explanation in Bases for moderator water-level has been revised.

52. A new paragraph was added to the start of Ref. 2, section 3.1 Bases to discuss the limitation on fuel pin worth. This paragraph was placed first to agree with the order the constraints are listed.

53. The specification for integrated thermal power was lowered to 2 kilowatt-hours. This change is a result of evaluating radiation survey values inside and outside of the reactor building. See the discussion in Ref. 1, 4.8. While facility environment monitors show less than minimum detectable radiation levels inside and outside, this is a result of operating power levels and schedule. The reduced integrated power value will limit the maximum possible dose in restricted areas to less than 5 rem per year and in unrestricted areas to less than 100 millirem per year if the facility operates at licensed power for the total 2 kilowatt-hours.

54. The last four paragraphs of Ref. 2, section 3.1 Bases were removed as unnecessary since the specification discussed in those paragraphs is discussed elsewhere (excess reactivity, pin worth and reactor power) or deleted (control rod worth).

55. Material in section 3.2 of Ref. 2 has been deleted and replaced by the new Section 3.1. The reactor parameters of Ref. 2 are not parameters used in the accident analysis and need not be stated as limits or constraints. The accident analysis makes no assumption about temperature or void coefficients of reactivity or about initial temperature, other than to use cross-sections based on 20C. Nor does the progress of the accident generate significant temperature changes or voids. The new core parameter of excess reactivity is the necessary parameter to define the accident magnitude, along with the anticipated response from the reactor safety system.

56. Ref. 2 sections 3.3 and 3.4 are renumbered 3.7 and section 3.4 and 3.8 respectively. New sections 3.3 – 3.6 have been added. This format change is in accordance with ANS-15.1-2007.

57. Paragraph 3.3.1c in Ref. 2 has been deleted as unnecessary. As explained in the response to Ref. 1 7.6 the instrument is not considered to be necessary for adequate radiological protection at the RPI reactor facility. The following paragraph was renumbered.

58. Requirements for portable radiological instruments have been added as paragraphs 3.7.2 and 3.7.3 and text added to the Bases to explain the purpose.

This addition was proposed by Ref. 1, 3.3.2. In Ref. 2 this is 3.3.2 and is less specific as to what instruments are required.

59. Specification 5 from Ref. 2, section 3.4 has added text for clarity. See response to Ref. 1, 3.4.5.

60. Specifications 8 and 9 from Ref. 2, section 3.4 combined into a single specification. See Response to Ref. 1, 3.4.8(A). The associated text in Bases (last paragraph) has been revised to reference a single specification and provide correct references to 10CFR20.

61. New section 3.9 has been added and material from section 5.6 of Ref. 2 was moved to the new section as proposed by Ref. 1, 5.6(A).

62. Sections 4.1 and 4.2 of were renumbered as 4.2 and 4.1 respectively and the titles revised to agree with the format of ANS-15.1-2007.

63. The Applicability and Specifications of section 4.2 of the Ref. 2 were changed to measure those parameters of an unknown core that are important to determine if the unknown core meets the requirements of section 3 and to calibrate the safety channel instruments. Core configuration changes can perturb the neutron flux at the ion chambers. Unless a power calibration is performed, the limiting safety system setting of 100 watts may not be met.

64. Ref. 2, Section 4.2 Objective revised to refer to Section 3.2 vice Specification 3.1.

65. Ref. 2, section 4.2, Specifications 1, 2 and 5, revised to reference the correct section and specification.

66. Ref. 2, section 4.2, specification 3, revised to require annual calibration of only the safety system channels. Calibration of all instrumentation, as is now required, is unnecessary since only the safety system channels provide reactor protection.

67. Ref. 2, section 4.2, Bases, first paragraph, revised to refer to the correct section and specification as proposed in Ref. 1, 4.2(B). The second paragraph was rewritten for clarity as discussed in response to Ref. 1, 4.2(C).

68. New Section 4.3 through 4.6 added and Section 4.3 renumbered as 4.7 to agree with ANS-15.1-2007 format.

69. Ref. 2, section 4.3 Specification revised to remove the mobile particulate gamma monitor as discussed in the response to Ref. 1, item 7.6. Further the periodic testing of the area monitors and criticality detector was reworded. The current requirement to check the channel is now a test using a radiation source. At least monthly test are required, even if the reactor is not operating. The associated basis was reworded for clarity.

70. Annual calibration of portable instruments added as a surveillance and explanation added to Bases.

71. Ref. 2, Section 5, was reorganized to agree with the organization and section titling of ANS-15.1-2007. Specifically, paragraphs 5.1, 5.2, and 5.3 combined into a new 5.1. Paragraph 5.4.1 and 5.5 combined as new 5.2 with minor

rewording in the first paragraph. Paragraphs 5.4.2, 5.4.3 and 5.4.4 were combined in new 5.3. The first paragraph of 5.6 is renumbered as 5.4 and the remaining paragraphs were moved to the new Section 3.9 as discussed previously. The last sentence of the new 5.4 was slightly revised.

72. Ref. 2, Section 6 was rewritten in conjunction with a revised Safety Analysis Report, Chapter 12.

73. Ref. 2, paragraph 6.1.1 was revised to agree with the structure recommended by ANS-15.1-2007. Some wording changes were also made to show reporting relationships and the responsibilities were moved to 6.2.2. The paragraph describing the health physicist's relationship with the facility organization was reworded to include the Facility Director, moved to 6.2.2, and the position identified as Radiation Safety Officer.

74. Responsibility of the Facility Director and the Operations Supervisor were reworded.

75. Ref. 2, paragraph 6.1.3(a)3) and 4) combined.

76. Ref. 2, paragraph 6.1.3(c)1) revised to allow fuel and control rod relocations in accordance with an approved experiment without direction from the Operations Supervisor.

77. Minimal staffing requirement added to Ref. 2, paragraph 6.3.1 for the condition of reactor shutdown, but not secured, as proposed by Ref. 1, 12.5. This inserted paragraph pushed the following paragraphs back one step.

78. Updated a reference in Ref. 2, paragraph 6.1.4 as proposed by Ref. 1, 12.7.

79. Ref. 2, paragraph 6.1.5 renumbered to section 6.2. This causes renumbering of subsequent sections.

80. Ref. 2, paragraphs 6.1.5.1 and 6.1.5.2 revised to refer to the NSRB Charter for its composition.

81. Ref. 2, paragraph 6.1.5.3 revised by adding item (d) for specific procedures requiring approval.

82. Added a new section 6.4, Radiation Safety.

83. Ref. 2, sections 6.2 through 6.6 renumbered as 6.4 through 6.8 respectively.

84. Ref. 2, section 6.3 1) completely rewritten as proposed by Ref. 1, 12.15.

85. Ref. 2, Section 6.4.1 was deleted since there is no safety limit. Section 6.4.2 was moved up to become section 6.4, then renumbered at 6.6.

86. Ref. 2, section 6.4.2(a) (now renumbered to 6.6 1)) revised to require reactor shutdown as discussed in response to Ref. 1, 12.17.

87. References to 10CFR Part 50.59 in Ref. 2, section 6.5.1 revised as proposed in Ref. 1, 12.18.

88. Health Physicist changed to RSO in Ref. 2, 6.5.1 (a) 6) c) for consistency with other parts of the Technical Specifications.

89. Ref. 2, 6.5.1 (d) revised to include preventative maintenance as proposed by Ref. 1, 12.18.

90. Ref. 2, 6.5.1(e) revised to clarify that license modifications are required only of a more severe accident is identified.

91. Ref. 2, 6.5.1(h) revised to include visitors as proposed by Ref. 1, 12.18.

92. Reports in Ref. 2, 6.5.2 reworded as proposed by Ref. 1, 12.19 and a report added for changes to level 1 or Level 2 personnel. Reference to telegraph reports was deleted since phone reports to the NRC Operations Center can be made at any time.

93. Ref. 2, section 6.6 completely revised to agree with ANS-15.1-2007 in types of records to be retained and minimum retention time as proposed by Ref. 1, 12.20.

94. Ref 2, section 6.8.1 revised to show the entire name of the university.