5. REACTOR COOLANT SYSTEMS

The maximum design power of 100 watts results in negligible heat up of the 2000 gallons of water in the reactor tank. Therefore, the RCF reactor does not require cooling.

The facility piping diagram is provided in Figure 5.1 for information.

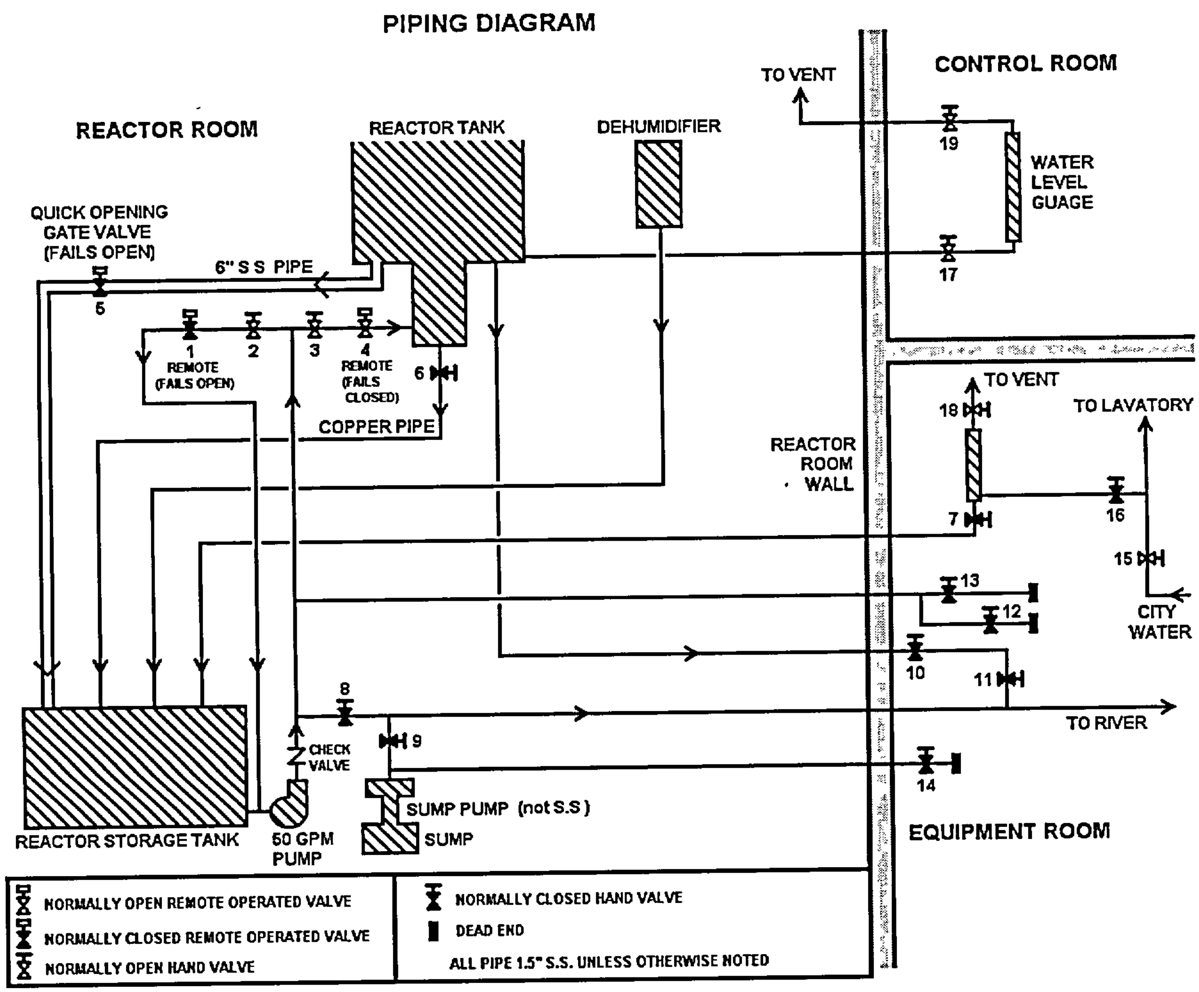


Figure 5.1: RCF Piping Diagram

6. ENGINEERED SAFETY FEATURES

Engineering safety features are not required for the RCF reactor due to the low operating power levels. Fission product inventories are minimal and well-contained within the fuel. A filter is provided on the reactor room ventilation stack to reduce the possibly of fission product release from the facility even further.

7. INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The RCF Reactor is designed for steady-state operation only. All operations are performed from two control panels in the control room. Control Panel 1 (CP-1) contains control rod position indication and movement controls, and all nuclear instrumentation. Control Panel 2 (CP-2) contains reactor tank fill and drain controls, source drive controls and auxiliary equipment controls. All instrumentation is currently analog; however, substantial equipment upgrades are in progress. Linear and log power/period monitors will be replaced with digital instrumentation, and all strip chart recorders will be replaced by digital plasma screen recorders.

7.2 Design of Instrumentation and Control Systems

7.2.1 Design Criteria

The instrumentation and control systems provide numerous functions, including rod position indication and movement control, and reactor power behavior. These systems also provide for automatic shutdown of the reactor if necessary. Redundancy is desired for anticipated possible problems with instrumentation.

7.2.2 Design Basis Requirements

The primary design basis requirement for reactor safety at the RCF is the safety limit on fuel pellet temperature listed in Section 2.1 of the Technical Specifications. Automatic scrams must be designed such that the temperature limit on the fuel is not reached.

7.2.3 System Description

The safety system channels that operate during reactor operation are specified in Section 3.2 of the Technical Specifications (Chapter 14). This indicates each channel's function and range of operation.

7.2.4 System Performance Analysis

I&C system functionality is thoroughly checked before any reactor startup. Scram setpoints and interlocks are also checked to ensure that the Technical Specifications are followed. Some of the instrumentation is very old, though it has been generally reliable. Regardless, an effort is underway to upgrade most of the instrumentation before it fails.

7.2.5 Conclusions

The RCF reactor has been operated successfully for decades with the existing instrumentation, and there is no reason to believe it will not continue to do so. Functionality of the I&C systems is frequently tested, and upgrades are in progress that will greatly improve reliability and precision of the instrumentation.

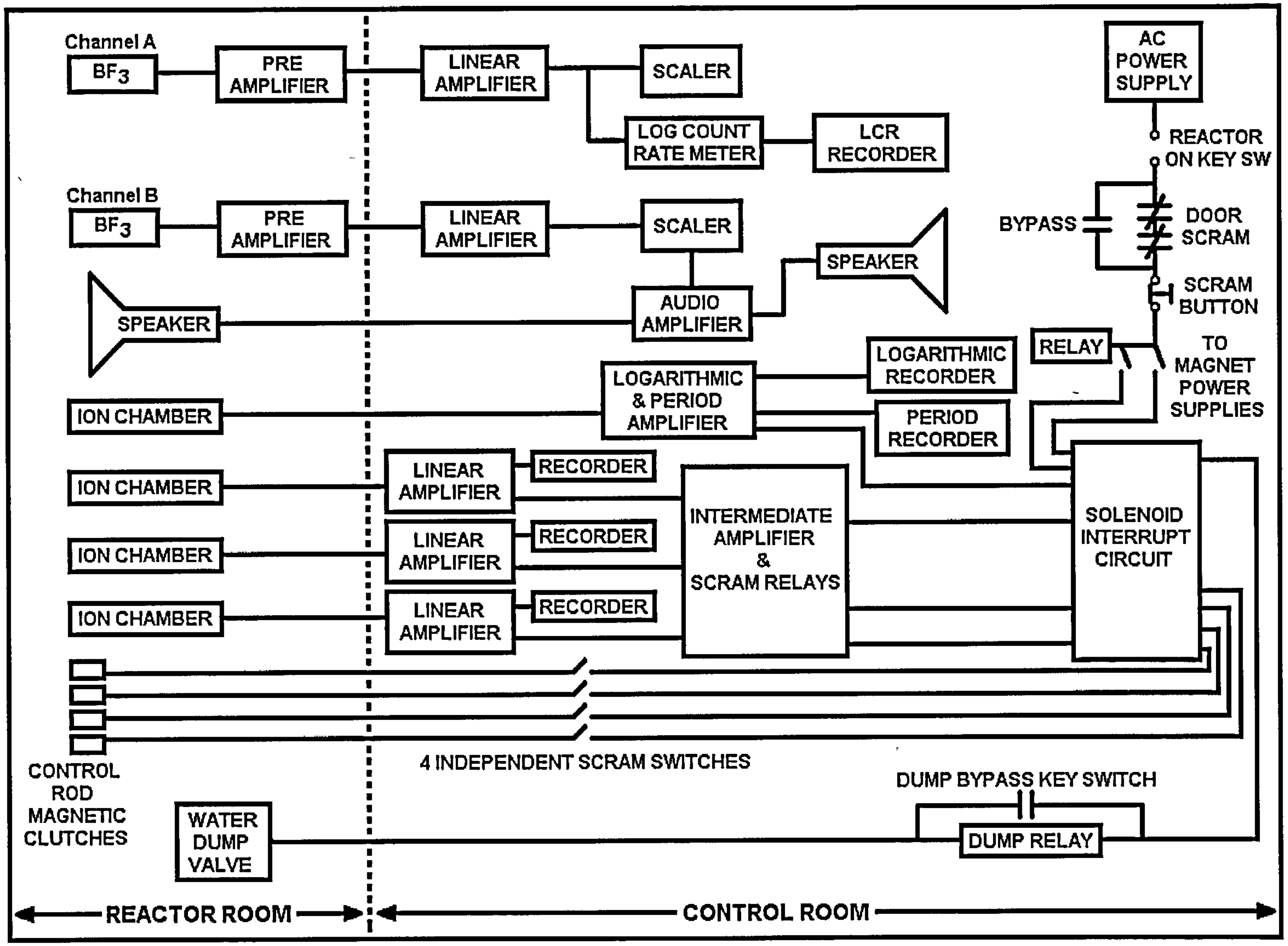
7.3 Reactor Control System

A block diagram of the control instrumentation is shown in Figure 7.1. Control of reactor power level must be performed manually. There is no automatic power level control capability.

The core has four control rods located at the periphery of the fuel lattice. Each drive gear box contains a lead screw actuating upper and lower limit switches, normally set for 36 inches of travel, and synchro transmitters for coarse (0-36") and fine (0-1" in increments of 0.01") position indication. The drive switches and synchro receivers are mounted on the control room console. When there is a reactor scram, the rod drives clutch magnet current is interrupted and all rods drop. Additionally, the moderator is dumped when it is not bypassed. The control rods and moderator dump are to operate within the limits of Section 3.2 of the Technical Specifications.

Figure 7.2 shows the interlock system for the RCF reactor. The control rods will not move if any of the conditions shown in the diagram are not met:

- Fill pump off
- Period > 15 sec
- Chart recorder power on
- Source range instrumentation reading > 2 cps
- 400 Hz power on (control rod position indicators)



CONTROL INSTRUMENTATION BLOCK DIAGRAM

Figure 7.1: Control Instrumentation Block Diagram

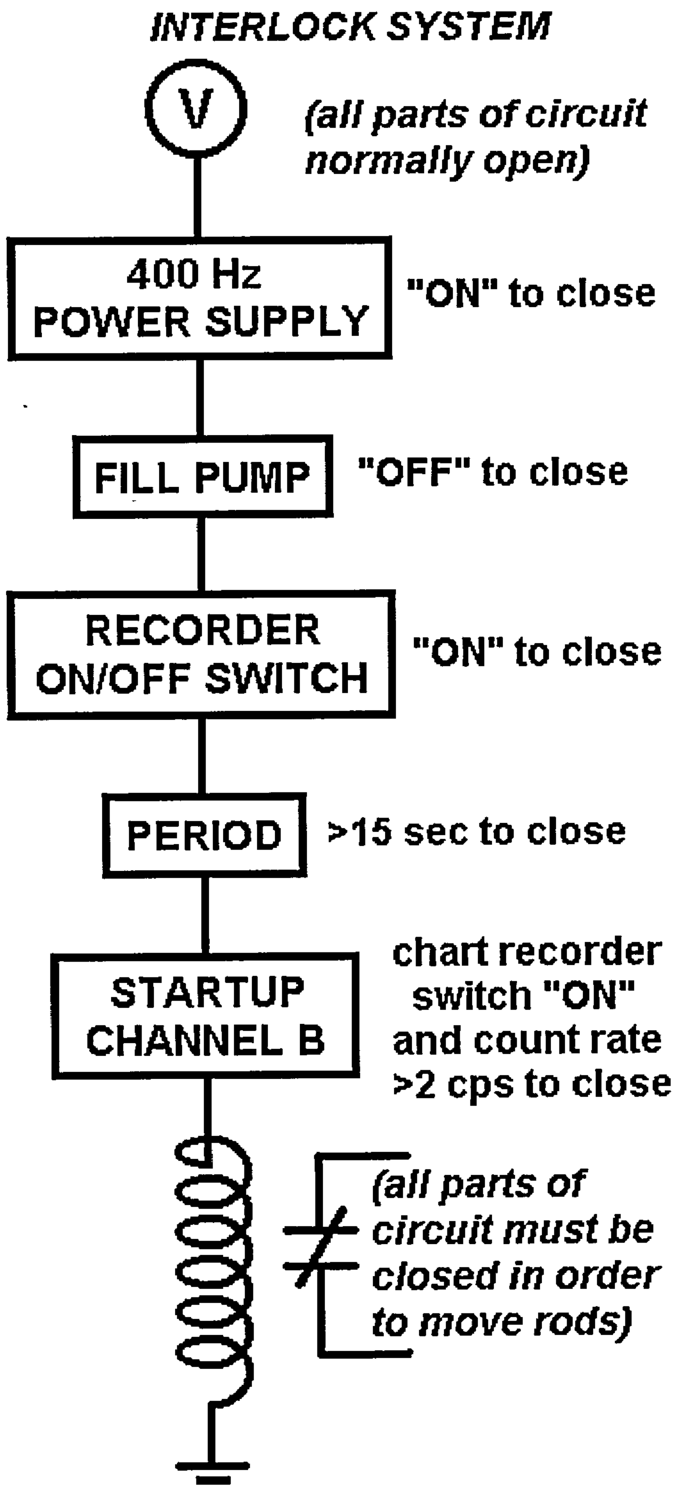


Figure 7.2: Interlock Block Diagram

7.4 Reactor Protection System

The scram circuit for the RCF is shown in Figure 7.3.

The nuclear instrumentation for control of the reactor consists of the following neutron flux detectors: 2 BF_3 counters (source range instrumentation), and 3 uncompensated ion chambers (2 linear amplifiers for intermediate range, 1 log amplifier for "power" range). The linear amplifiers will initiate a scram signal if the reading reaches 90% of the current range, and the log/period amplifier will cause a scram if the period falls below 5 seconds or the log power exceeds 135 W. The bases supporting the scram setpoints are outlined in the Technical Specifications.

There are also several manual scrams:

- Reactor console power (scram circuit power)
- Manual scram button
- Scram circuit key
- Reactor room door

7.5 Engineered Safety Features Actuation Systems

There are no engineered safety features actuation systems.

7.6 Control Console and Display Instruments

The primary control console, CP-1 (pictured in Figure 1.3) performs the functions outlined in Section 7.1: control rod position indication and movement control, as well as providing the indications from the nuclear instrumentation shown in the control block diagram (Figure 7.1). CP-2 contains moderator fill/drain controls, fast dump controls, source drive controls and auxiliary system controls.

The neutron source yields about 10^7 neutrons/second, which is sufficient to maintain the source range rate above the minimum requirement for startup of 2 cps. The source is also sufficient to maintain the logarithmic count rate meter and linear amplifiers on scale at all times when the reactor is subcritical. The linear and logarithmic meters cover all necessary power ranges.

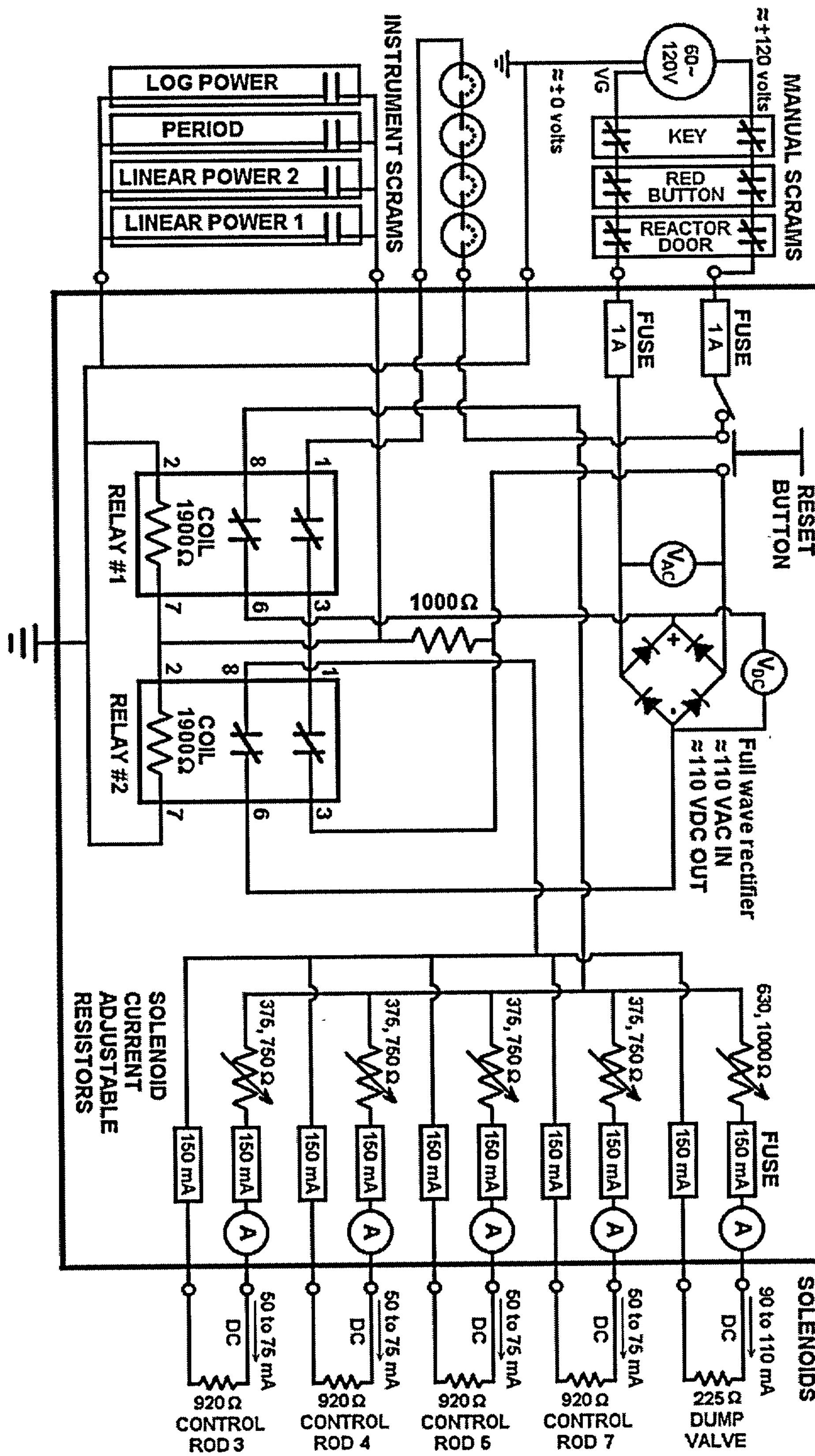


Figure 7.3: Scram Circuit

7.7 Radiation Monitoring System

In accordance with Section 3.3 of the Technical Specifications there is an area gamma monitoring system. Four G-M tubes are used, one at each of the following locations: control room, reactor room near the fuel vault (doubles as vault criticality monitor), reactor deck, and in the equipment hallway. Portable radiation monitors are also available. The area gamma monitors provide visual and audible indications.

The area gamma monitors are found in the following locations and have the following alarm setpoints:

- Control room: 10 mrem/hr
- Equipment hallway: 40 mrem/hr
- Outside vault (also acts as vault criticality monitor): 20 mrem/hr
- Reactor deck: 100 mrem/hr

Whenever the reactor is to be operated, the particulate activity of the reactor room atmosphere is monitored. The air monitor counts the beta-gamma activity on the filter paper through which a continuous 5 cfm sample of air is drawn from the stack duct. It provides audible and visual alarms if the count rate goes above 2000 cpm.

8. ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

Electrical power to the facility is not necessary to keep the reactor safely shutdown. The electrical system at the RCF is similar to that which would be found in any other industrial structure of similar age.

8.2 Emergency Electrical Power Systems

There are no emergency electrical power systems.

9. AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

A stack extends above the reactor room to 50 feet above ground level. It contains a CWS filter for removing the small amount of fission products that might evolve from a maximum credible accident. Air circulation occurs via natural circulation. Forced circulation ventilation is provided in all other rooms in the facility.

Temperature control in the facility is provided by an air conditioning system near the bathroom, and a small boiler house outside the maintenance hallway (which is located immediately outside the reactor room).

9.2 Handling and Storage of Reactor Fuel

Because the RCF reactor operates at such low power levels, it is reasonable to assume there is effectively no depletion in the fuel. Consequently, there are no spent fuel concerns; nor is there ever a need to bring more fuel into the facility. Nuclear material will not need to be removed from the RCF until the facility is decommissioned.

Individual fuel pins are occasionally added or removed from the core. Each fuel pin has a hole built into the top for ease of removal (see Figure 4.5). The tool used to add or remove fuel pins to the core is just a modified wire coat hangar. It is remarkably well-suited for this purpose.

Fuel pins that are not in the core are stored in the locked fuel vault (Figure 1.2). The fuel vault is described in Section 1.3. In compliance with the vault design basis, no more than 15 fuel pins may be stored in any tube.

9.3 Fire Protection Systems and Programs

The fire detection and protection systems in the RCF meet state and local requirements. All walls in the facility are masonry. Fire extinguishers are located in the building and are checked at regular intervals.

9.4 Communication Systems

The RCF has a commercial phone line with phones in the control room and office. A cellular phone is also located in the office.

There is a battery-powered, 2-way wired intercom system between the control room and reactor room.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

Operation of the RCF reactor does not result in production of radioactive byproducts. There are no radioactive materials at the RCF that are used for reactor operation or experiments (other than the PuBe neutron source). There are several small calibration sources in the facility.

9.6 Cover Gas Control in Closed Primary Coolant Systems

This section does not apply to the RCF reactor.

9.7 Other Auxiliary Systems

There are no other auxiliary systems required for safe reactor operation.

10. EXPERIMENTAL FACILITIES AND UTILIZATION

There are currently no experimental facilities at the RCF.

Experiments commonly performed at the RCF are listed in Section 1.6 and do not require specific experimental facilities. For the Core B fuel pin configuration (annular core), it would be possible to modify the spare control rod drive to raise and lower experiments into the center of the core, but there are currently no plans to do this. This system would operate like the control rod drives and would be limited by the maximum experiment reactivity worth of 60 cents found in Section 3.4 of the Technical Specifications.

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.

11. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

11.1.1 Radiation Sources

11.1.1.1 Airborne Radiation Sources

There are normally no airborne sources of radiation at the RCF. In the event of fuel pin clad rupture, the fission product inventory may be released but would be too small to pose a significant health risk.

11.1.1.2 Liquid Radioactive Sources

A small amount of radioactivity exists in the reactor tank water during operation, but this consists of short-lived isotopes and does not pose a health concern.

11.1.1.3 Solid Radioactive Sources

The reactor fuel constitutes a solid radioactive source; though other than short-lived fission product decay, the fuel does not present a significant health concern. In fact, in most cases the fuel can be safely handled minutes after reactor operation.

11.1.2 Radiation Protection Program

RPI has a structured radiation safety program with a staff equipped with radiation detection instrumentation to determine, control, and document occupational radiation exposures at its reactor facility. In addition, the critical facility monitors liquid effluents before release to comply with applicable guidelines and monitors for airborne activity within the reactor room to confirm that all effluents contain insignificant concentrations of radioactive materials.

11.1.3 ALARA Program

The university Provost, in the Radiation Safety Regulations and Procedures, has established formally the policy that operations are to be conducted in a manner to maintain all radiation exposure consistent with the ALARA principle. All proposed experiments and procedures at the reactor are reviewed for ways to decrease the potential exposure of personnel. All unanticipated or unusual reactor-related exposure will be investigated by the Office of Radiation and Nuclear Safety and the operations staff to develop methods to prevent recurrences.

11.1.4 Radiation Monitoring and Surveying

The area gamma monitoring system and air particulate monitor are described in Section 7.7. In addition, a radiation survey is performed in the reactor room as part of the pre-startup procedure when the reactor is to be operated.

The health physics staff participates in experiment planning by reviewing all proposed procedures for methods of minimizing personnel exposure and limiting the generation of radioactive waste. Approved procedures specify the type and degree of radiation safety support required by each activity.

11.1.5 Radiation Exposure Control and Dosimetry

The RPI personnel monitoring program is described in the Radiation Safety Regulations and Procedures Manual. To summarize the program, personnel exposures are measured by the use of thermoluminescent dosimeters (TLDs) assigned to individuals who might be exposed to radiation. In addition, instrument dose rate and time measurements are used to administratively keep occupational exposures well below the applicable limits in 10 CFR 20.

Staff TLDs are checked regularly and consistently show no measurable radiation exposure.

11.1.6 Contamination Control

Monthly contamination surveys are performed to ensure there is no contamination in the facility. These surveys routinely show that there is no detectable contamination.

11.1.7 Environmental Monitoring

The environmental monitoring program consists of several TLDs placed at the exclusion area boundary and at the site boundary. The results indicate about 5 mrem/yr at the site boundary and up to 15 mrem/yr at the exclusion area boundary above that measured at the General Electric Company Guard Station more than 1.6 km away.

11.2 Radioactive Waste Management

The RCF reactor produces insignificant quantities of radioactive waste during normal use because of both its low power level and its limited operating schedule, which are restricted by the Technical Specifications.

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.

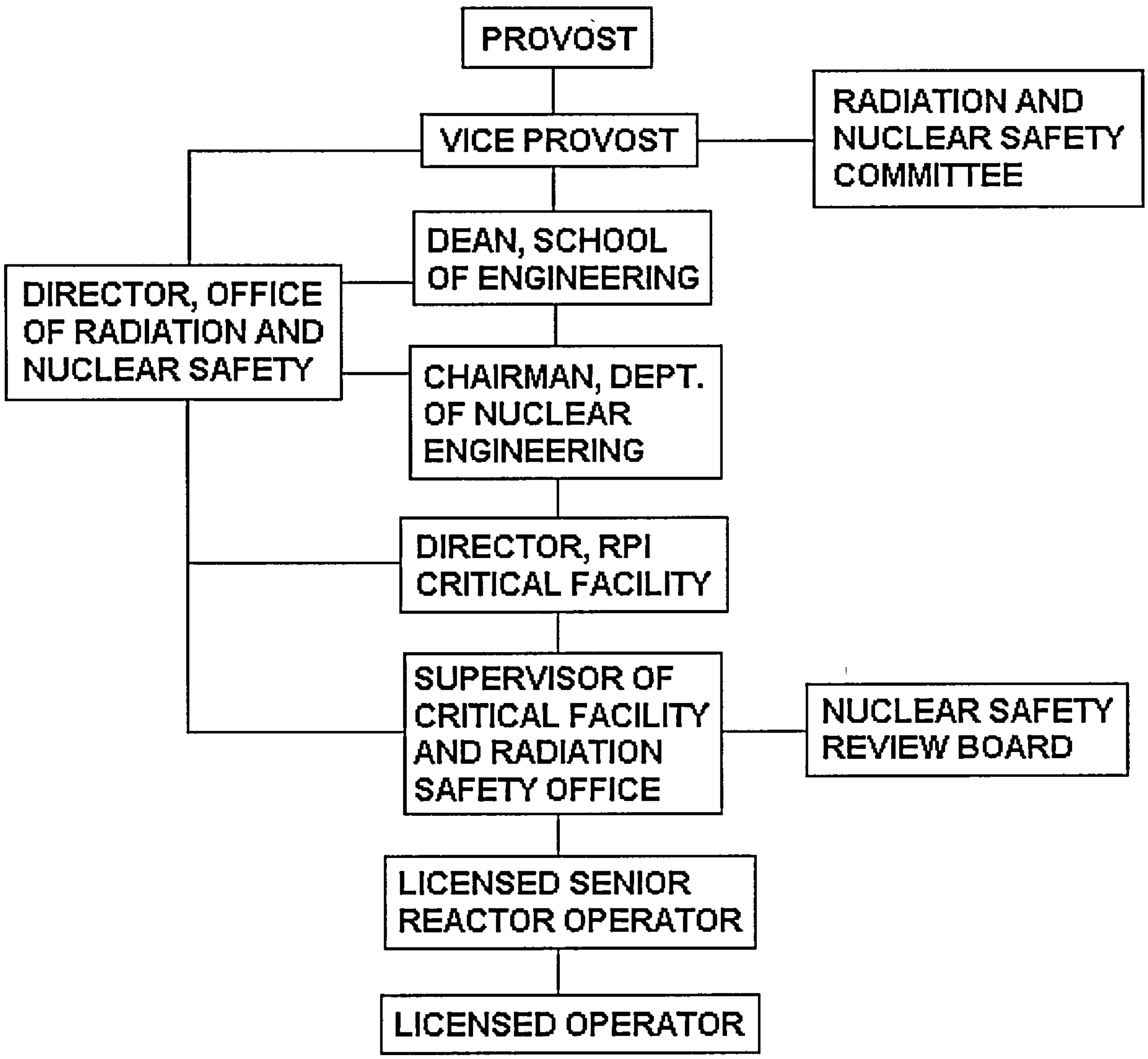


Figure 12.1: RCF Organization

12.1.2 Responsibility

The responsibilities of the individuals in Figure 12.1 are explained in Section 6.1 of the Technical Specifications.

12.1.3 Staffing

Staffing requirements are found in Section 6.1.3 of the Technical Specifications.

12.1.4 Selection and Training of Personnel

New reactor operators are selected from interested students enrolled in classes that take place at the RCF. Most of the training of reactor operators is done by existing RCF personnel. The Operator Requalification Program meets the regulations in 10 CFR 55. *The requalification program is included in the materials submitted for relicensing.*

12.1.5 Radiation Safety

Radiation safety aspects of facility operation are typically performed by members of the RCF staff, including routine radiation and contamination surveys and air sampling. Occasionally, some of these tasks are performed by a member of the campus radiation safety organization.

12.2 Review and Audit Activities

The Nuclear Safety Review Board (NSRB) provides independent review and audits facility activities. The Technical Specifications list the qualifications and provide that alternate members may be appointed by the NSRB Chairman. The NSRB meets at least semiannually. The board must review and approve plans for modifications to the reactor, new experiments, and proposed changes to the license or to procedures. The board also is responsible for conducting audits of reactor facility operations and management, and for reporting the results thereof to the RCF Director.

12.3 Procedures

Written operating procedures are used for the following:

- Reactor Pre-Startup
- Reactor Operations
- Surveillances
- Emergencies

The operating procedures are included in the materials submitted for relicensing.

12.4 Required Actions

Required actions to be taken in the event that a safety limit is exceeded or other reportable occurrence takes place are outlined in Section 6.5 of the Technical Specifications.

12.5 Reports

Reports will be made to the NRC in accordance with Section 6.6 of the Technical Specifications.

12.6 Records

Records for the RCF will be kept in accordance with Section 6.7 of the Technical Specifications.

12.7 Emergency Planning

10 CFR 50.54 requires that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E of 10 CFR 50. The Emergency Plan for the RCF currently in use is dated December 1984. *The Emergency Plan is included in the materials submitted for relicensing.*

The objective of the plan is to establish guidelines for responding to emergency conditions should a radiological emergency occur at the Critical Facility site that may affect the health and safety of workers or the general public.

The 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.

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.

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. The NRC staff has reviewed the Physical Security Plan submitted in 1983 and concluded that the plan met the requirements of 10 CFR 73.67 for special nuclear material of moderate strategic significance. 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). Amendment No. 4 to the facility Operating License CX-22, dated July 27, 1983, incorporated the Physical Security Plan as a condition of the license.

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 are described in Section 12.1.4. *The requalification program is included in the materials submitted for relicensing.*

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

An environmental report is included in the materials submitted for relicensing. 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.

13. ACCIDENT ANALYSIS

13.1 Accident-Initiating Events and Scenarios

Several potentially serious accident scenarios have been evaluated and, even in the worst event sequence considered, no release of a significant quantity of radioactive fission products to the reactor cell would occur. Effects due to natural phenomena, mechanical rearrangement of the fuel, and reactivity insertion were all analyzed.

13.1.1 Maximum Hypothetical Accident

The potentially most severe accident at the RCF is due to reactivity insertion and, hence, this is the limiting case for design purposes. Hypothesizing that an unsecured experiment causes $0.60 reactivity to be instantaneously inserted while the reactor is operating at maximum power, the resultant excursion induces a negligible rise in fuel temperature. This scenario and the details of the analysis are discussed in the next section.

13.1.2 Insertion of Excess Reactivity

The most extreme scenario hypothesized consists of the worst reactivity excursion coincident with a single failure in the reactor protection system.

The worst reactivity excursion results from an unsecured experiment with a reactivity worth equal to the maximum excess reactivity allowed by the Technical Specifications of $0.60. Specifically, this could result from an experiment in which a strip of poison, such as boron, is placed in the core, the control rods pulled all the way out to obtain just critical conditions, thereupon the boron strip falls out of the core, resulting in a step reactivity insertion of the specified amount. A pre-accident power level of 200 watts is assumed, based upon the Technical Specification limit of 100 watts and incorporating a factor of two to account for the cumulative uncertainties associated with instrument calibration. For analytical purposes, the reactivity feedback effects of temperature and void formation are neglected so that control rod insertion is necessarily the terminating event.

The open circuit failure of the ion chamber serving log power and period channel 2 (PP2), coincident with the beginning of the accident, is also assumed. Because this one ion chamber supplies the input to the circuit that provides both the log power (135 watts) and the short period (5 seconds) scram, these scram relays are assumed to be disabled. The failure chosen, then, is the "worst case" single instrument malfunction. Remaining scram protection is provided only by the two linear power channels (LP1, LP2), each of which initiates a scram if its respective meter indication exceeds 90% of the selected scale. Commonly, the operator upscales these meters by factors of three as power increases during a directed power increase to preclude an inadvertent shutdown. For purposes of the accident scenario, LP1 and LP2 are assumed to indicate a value of 10% on the highest selectable scale at the onset of the accident, roughly correlating with 200 watts in-core power (100 watts indicated with factor of two uncertainty). Thus the power must increase by a factor of nine from this pre-accident level to prompt the linear power channel scram activation. Notably, because of the nature of the accident, its severity is not sensitive to variation in initial power. The single insertion of a fixed amount of positive reactivity quickly puts the reactor on a constant positive period, so that both the value of reactor power and its rate of increase when scram is initiated are unrelated to power levels immediately beforehand. Hence selection of a very low power, visible yet well below the point of adding heat, would not have aggravated the results of the analysis.

13.1.3 Loss of Coolant

Loss of coolant does not result in an accident situation at the RCF. In fact, the fast moderator dump is considered an alternate scram mechanism.

13.1.4 Loss of Coolant Flow

This does not apply to the RCF reactor.

13.1.5 Mishandling or Malfunction of Fuel

Mechanical rearrangement of the fuel to obtain a supercritical configuration, inadvertently or with intent, is not a credible occurrence. Physical damage to fuel pins is precluded both by each element's cladding and the stainless steel support plates/stanchions that compose the basic core structure. In the unlikely event that sufficient force to break one or more of the fuel pins was developed, the very low fission product inventory accumulated in the fuel elements would not cause a significant off-site hazard.

13.1.6 Experiment Malfunction

Experiments must be designed such that the maximum possible reactivity effect is 60 cents as limited by the Technical Specifications. Failure of an experiment with this reactivity worth is considered as a possible accident-initiating event and is described in Section 13.1.2.

13.1.7 Loss of Normal Electrical Power

Loss of normal electrical power will cause the reactor to shut down. This does not result in an accident situation.

13.1.8 External Events

Adequate protection against the potential effects of natural phenomena including fires, windstorms, floods, and earthquakes is provided. Radiological hazards to the public from these events are not significant.

An event such as an airplane impact or terrorist bomb would likely destroy the facility, but since the fission product inventory of the RCF is small, the health effects would be minimal (similar to a fuel malfunction incident as described in Section 13.1.5).

13.1.9 Mishandling or Malfunction of Equipment

No equipment malfunction scenarios are envisioned that would result in a serious accident scenario.

13.2 Accident Analysis and Determination of Consequences

With the reactor operating initially at 200 watts, the insertion of \$0.60 positive reactivity causes power to promptly jump to 600 watts and then increase on a period of 3.0 seconds to 1800 watts, at which point LP1 and/or LP2 generate a scram signal. Allowing 1.5 seconds thereafter for the rods to be bottomed (Technical Specification is 900 msec), analysis conservatively assumes the instantaneous insertion of \$1.000 negative reactivity (less than the core shutdown margin) at 5 seconds after the excursion begins.

Maximum power reached during the transient is slightly below 3050 watts, depositing about 10 kJ of energy in the core and inducing a fuel temperature rise of less than 0.1°C above an initial value of 20°C. This energy deposition is roughly a factor of 10^3 less than the core safety limit identified in the Technical Specifications. Figure 13.1 portrays changes in power for the stated reactivity insertion transient, annotated with pertinent events. Clearly the integrity of the fuel is not in question. Additionally, while feedback effects are intentionally disregarded in the analysis, the very small temperature change encountered would make their cumulative effect negligible. This conclusion is valid for both the Core A and Core B pin arrangements.

The supporting transient analyses conducted employed the "FRKGB" computer code model[2], developed at RPI specifically for low power pool reactors. The model utilized Runge-Kutta time stepping methods to derive numerical solutions. The program was initially benchmarked against a set of Gaussian, Nordheim-Fuchs, and SPERT type bursts.

Tables 13.1 through 13.3 list pertinent nuclear and physical characteristics of the core configuration used in the analysis that are relevant to safe operations.

The core physics design and fuel vault criticality calculations were carried out using the LEOPARD[3] code with ENDF/B-4 based data) to compute few group diffusion constants, the PLATAB[4] code to compute equivalent few group diffusion constants for strong absorbers (this code used detailed flux spectra from LEOPARD), and the DIFXY[5] code to apply few group diffusion code theory in X-Y geometry.

Figures 13.2 and 13.3 display graphs of the temperature coefficient of reactivity for the solid (Core A) and annular (Core B) core fuel pin arrangements, respectively. The curves portray data derived from the computer codes referenced above.

13.3 Summary and Conclusions

The most severe hypothetical accident at the RCF involves a reactivity insertion transient. However, none of the accidents postulated would release significant fission products from the fuel. No credible accidents at the RCF pose a significant risk to public health and safety.

Table 13.1: Nuclear and Physical Characteristics of the RPI LEU Core

Effective Delayed Neutron Fraction,	$\beta_{eff} = 0.00765$
Effective Neutron Lifetime,	$l^* = 12.2 \times 10^{-6}$ sec

Delayed Neutron Data

Group No.	β_i/β_{eff}	Decay Constant(1)
1	0.041	3.01
2	0.115	1.14
3	0.396	0.301
4	0.196	0.111
5	0.219	0.0305
6	0.033	0.0124

Reactor Power,	P = 100 watts
Axial Power Shape	Chopped Sine
Coolant Temperature,	T = 20°C

(1) G.R. Keepin, "Physics of Nuclear Kinetics", Addison Wesley, 1965.

Table 13.2: Kinetics Parameters of RPI LEU Core and Technical Specifications

Kinetics Parameter	LEU Core Value	Technical Specification
Excess Reactivity at 68°F	0.00468	< 0.00468
Reactivity with One Stuck Rod	< -0.005	< -0.005
Shutdown Margin	> 0.02	> 0.02
Core Average Isothermal Temperature Coeff. Of Reactivity	< 0 for T > 91°F(1)	< 0 for T > 100°F
Core Average Void Coefficient of Reactivity(1)	$-9.99\text{x}10^{-3}/cm^3$ at 57°F	$< -3.3\text{x}10^{-6}/cm^3$
Integrated Reactivity Due to Temperature Change, 50°F-T(α_T=0)(1)	$1.073\text{x}10^{-3}$	$< 1.148\text{x}10^{-3}$
Reactivity Worth of Standard Fuel Assembly(2)	< 0.039	< 0.039

(1) Value cited is for the Core B arrangement. Values for Core A are less restrictive.

(2) Note: A "standard fuel assembly" consists of a single fuel pin in the RPI LEU Core.

Table 13.3: Calculated Feedback Coefficients for RPI LEU Core

Core Average Void Coefficient of Reactivity = 0.7647 pcm/cm^3

Radial[(1)] Values of the Average Void Coefficient of Reactivity:

Distance from Core Center (cm)	Average Void Coefficient (pcm/cm^3)
0.00	-1.2795
2.97	-1.26078
5.94	-1.14822
8.92	-0.97842
11.89	-0.77206
14.86	-0.56474
17.83	-0.27250

(1) Values cited along a radial from the core center outward toward a control rod with symmetry assumed.

Isothermal Temperature Coefficient for LEU Core A:

$$\alpha_T(°C) = 1.825\text{x}10^{-8}T^2 - 4.8\text{x}10^{-6}T + 6.932\text{x}10^{-5}$$

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

Isothermal Temperature Coefficient for LEU Core B:

$$\alpha_T(°C) = 2.113\text{x}10^{-8}T^2 - 5.0\text{x}10^{-6}T + 1.423\text{x}10^{-4}$$

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

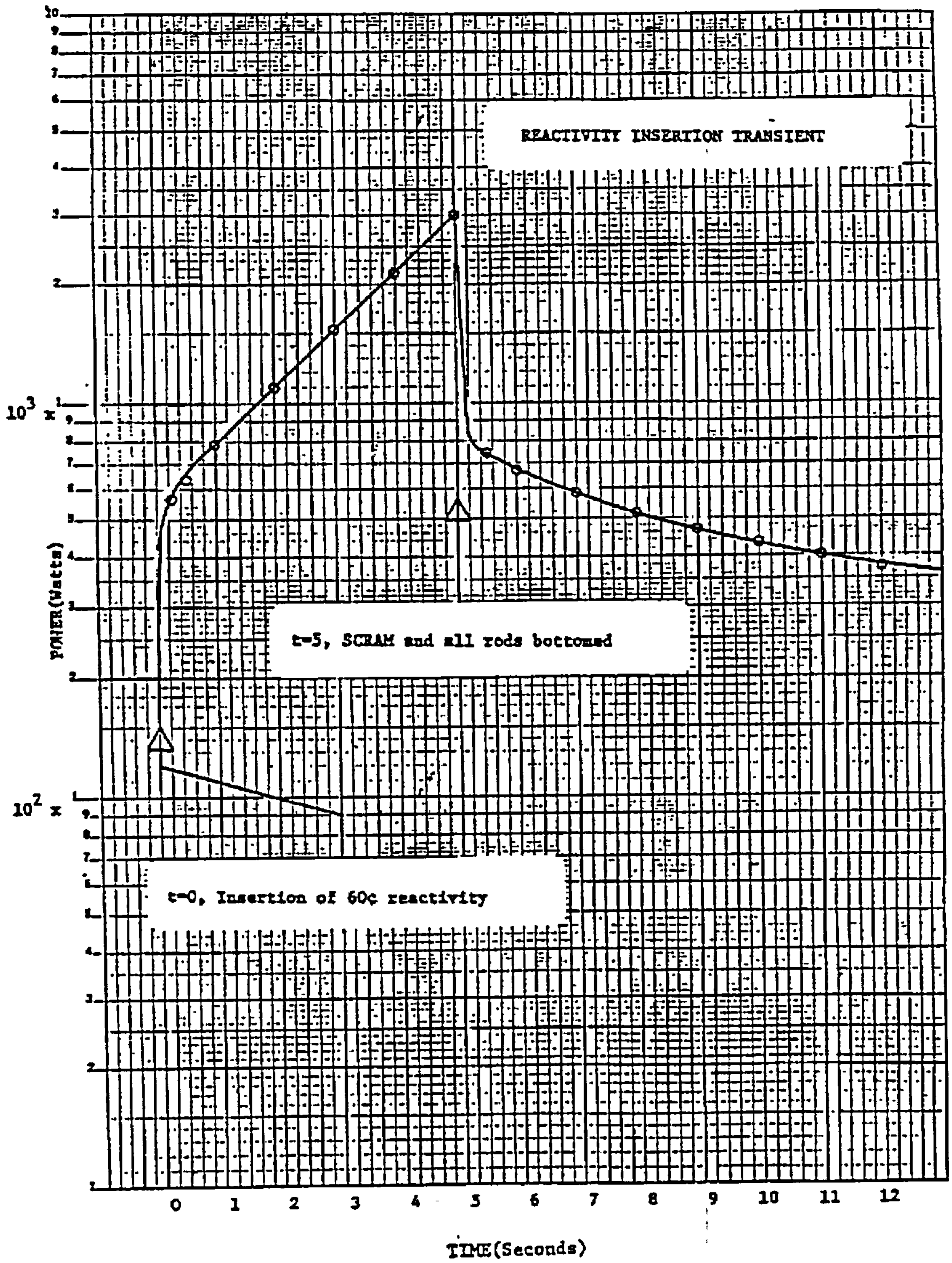


Figure 13.1: Reactivity Insertion Transient

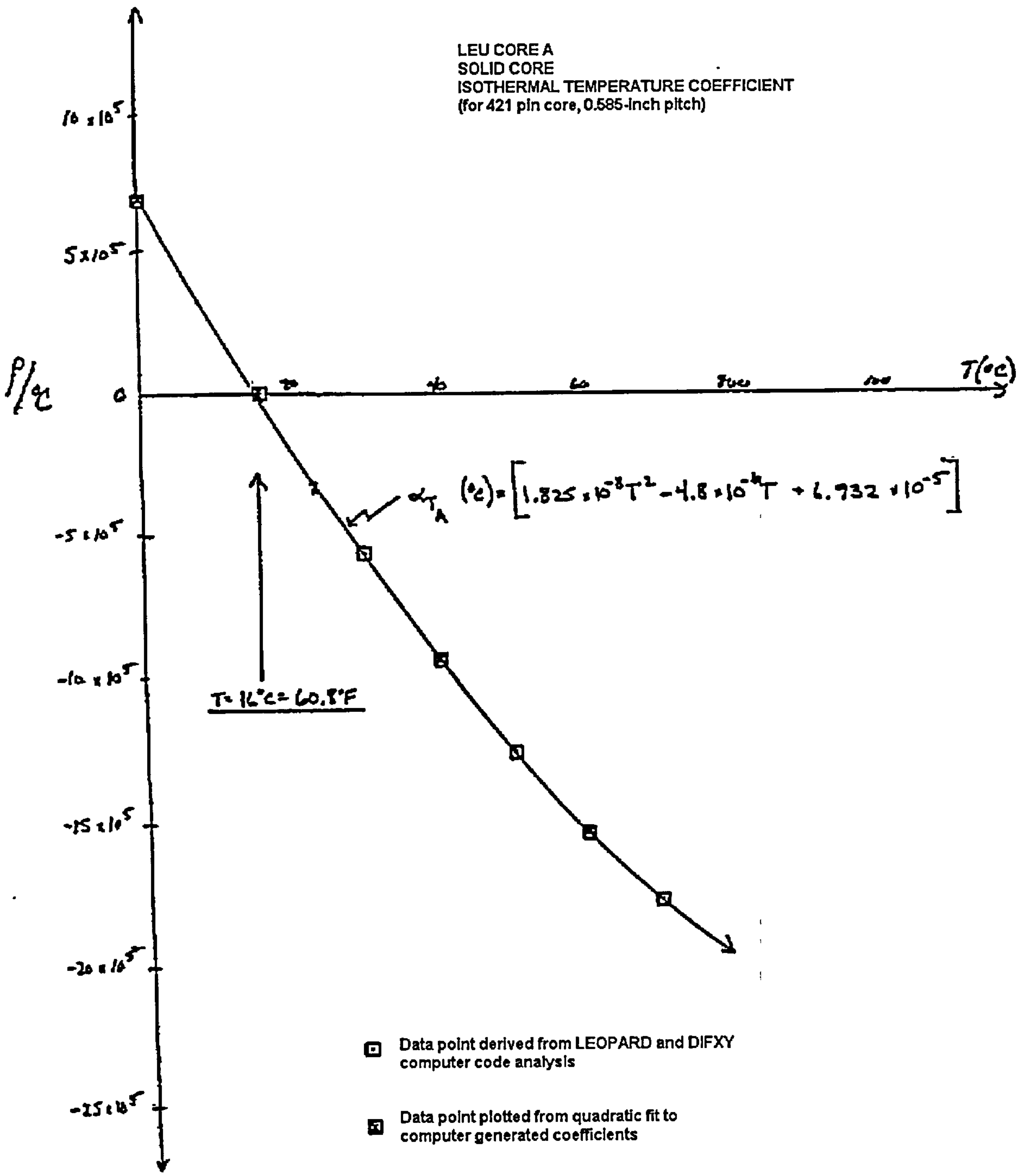


Figure 13.2: Core A, Isothermal Temperature Coefficient

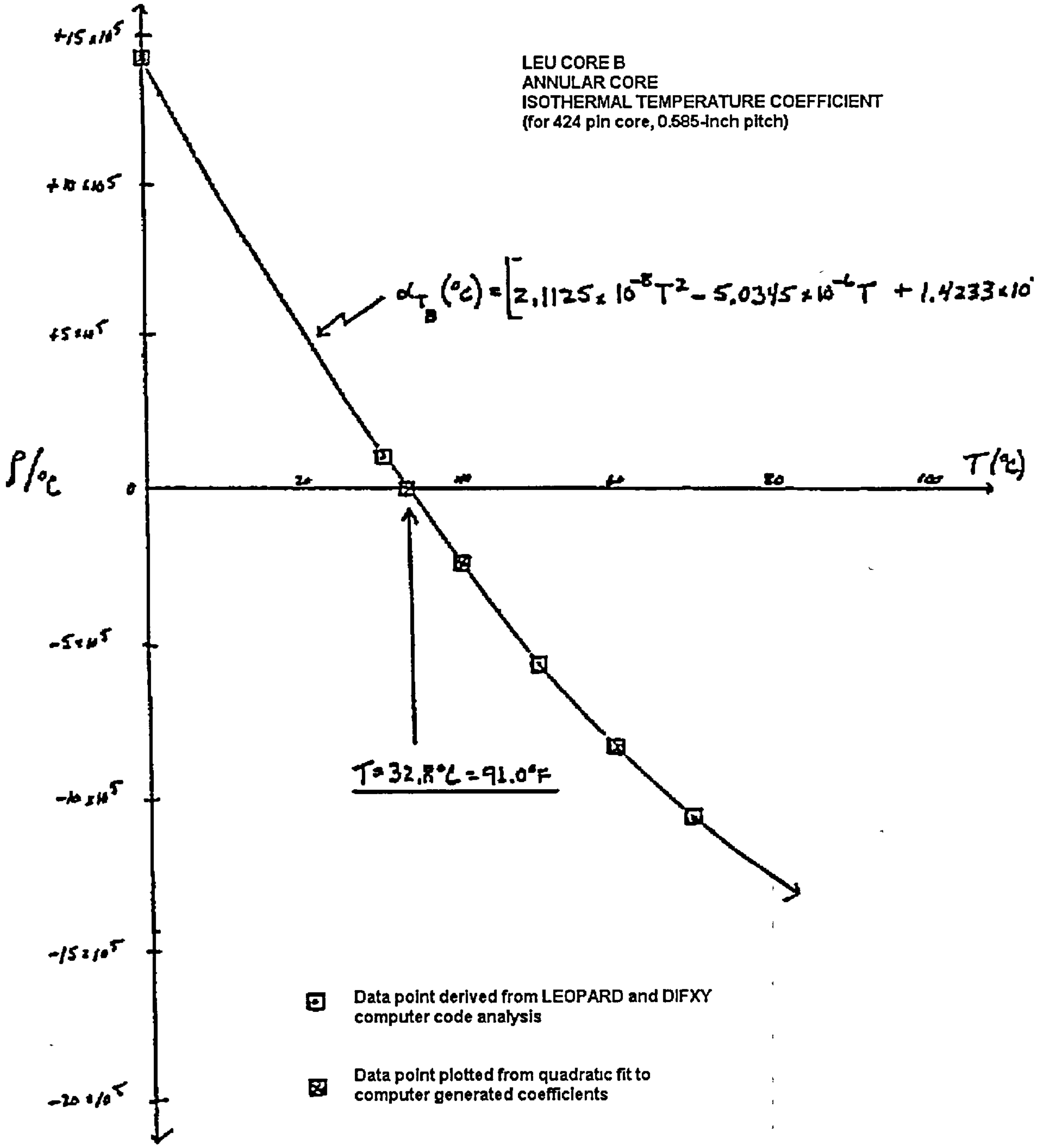


Figure 13.3: Core B, Isothermal Temperature Coefficient

13.4 References

1. D.R. Harris and F. Wicks, "Rensselaer Polytechnic Institute Critical Facility Safety Analysis Report." Docket No. 50-225. License No. CX-22. January 1983.

2. D.R. Harris, O.C. Jones, F.E. Wicks, A.B. Harris, F. Rodriguez-Vera, and C.F. Chuang, "Design Basis Transient Analysis for Low Power Research Reactors", Proc. Of Int. Symposium on Use and Development of Low and Medium Flux Research Reactors, Cambridge, Mass., Oct. 16-19, 1983, Atomkemenergie, Kerntechnik, 44, 450 (1983).

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4. D.R. Harris, "PLATAB, a Code for Computation of Equivalent Diffusion Theory Parameters for Strong Absorbers," Tech. Apl. Associates, TAA-1, 1986.

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14. TECHNICAL SPECIFICATIONS

The proposed Technical Specifications for the RCF reactor are attached as Appendix A to the SAR. They have been updated from the Technical Specifications currently in force (submitted in 1983) to reflect changes to the facility. The format remains essentially unchanged from the previous version.

Normal reactor operation within the limits of these Technical Specifications will not result in offsite exposures in excess of 10 CFR 20 limits. In addition, the limiting conditions for operation and surveillance requirements will limit reduce the probability of malfunctions and mitigate the consequences to the public of accident events.

15. FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

This section does not apply to the RCF relicensing process.

15.2 Financial Ability to Operate a Non-Power Reactor

The RCF has an exceptionally low annual budget; typically below $50,000. This number has been somewhat higher over the last few years due to the substantial equipment upgrades underway.

Total grants for fiscal year 2002 were $43,798, received entirely from DOE toward the facility equipment upgrades. RPI contributed an additional $20,000 for this purpose. Purchases for other equipment and supplies totaled $11,061. Gas and electric bills totaled $9,548.

Salaries for RCF personnel are included in the standard $50,000 annual budget. Currently, there are no full-time staff members at the facility.

With such low operating costs, it is not expected that funding for RCF operations will be a problem in the foreseeable future.

15.3 Financial Ability to Decommission the Facility

Decommissioning cost estimates vary depending upon the degree of work to be completed. If the only objective is to remove all fissionable material (i.e. the fuel) from the facility, decommissioning costs are estimated to be about $50,000. This relatively low cost does not pose a problem for the institute. A complete decommissioning, including removal of all hazardous waste and asbestos, and clean-up of the facility grounds (presumably contaminated from former ALCO plant operations), would cost at least 10 times that amount, or $500,000.

APPENDIX A: Proposed Technical Specifications

1. INTRODUCTION

1.1 Scope

The following constitutes the proposed Technical Specifications for the RPI Reactor Critical Facility, as required by 10 CFR 50.36.

1.2 Format

Content and section numbering are in accordance with section 1.2.2 of ANSI/ANS 15.1.

1.3 Definitions

The terms Safety Limit (SL), Limiting Safety System Setting (LSSS), and Limiting Condition for Operation (LCO), and Surveillance Requirements are as defined in 50.36 of 10 CFR Part 50.

A. Channel Calibration - The correlation of channel outputs to known input signals and other known parameters. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip.

B. Channel Check - Qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

C. Channel Test - The injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm and/or initiating action.

D. Control Rod Assembly - A control mechanism consisting of a stainless steel basket that houses two absorber sections, one above the other. These absorber sections contain enriched boron in iron. All absorber sections are 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.

E. Excess Reactivity - The available reactivity above a cold, clean critical configuration which may be added by manipulation of controls.

F. Experiment - (1) An apparatus, device, or material placed in the reactor vessel, and/or (2) any operation designed to measure reactor characteristics.

G. Measuring Channel - The combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.

H. Measured Value - The value of the process variable as it appears on the output of a measuring channel.

I. Movable Experiment - A movable experiment is one in which material may be inserted, removed, or manipulated while the reactor is critical.

J. Operable - A system or component is capable of performing its intended function in its required manner.

K. Operating - A system or component is performing its intended function in its required manner.

L. Reactor Safety System - Combination of safety channels and associated circuitry that forms the automatic protective system for the reactor or provides information that requires manual protective action to be initiated.

M. Reactor Scram - A gravity drop of the control rods accompanied by the opening of the moderator dump valve. The scram can be initiated either manually or automatically by the safety system.

N. Reactor Secured - (1) The full insertion of all control rods has been verified, (2) the console key is removed, and (3) no operation is in progress that involves moving fuel pins in the reactor vessel, the insertion or removal of experiments from the reactor vessel, or control rod maintenance.

O. Reactor Shutdown - The control rods are fully inserted and the reactor is shutdown by at least 1.00$. The reactor is considered to be operating whenever this condition is not met and more than 60% of the total number of fuel pins required for criticality in a given configuration have been loaded in the core.

P. Readily Available on Call - The Licensed Senior Operator (LSO) on duty shall remain within a 30 mile radius or 60 minutes travel time of the facility, whichever is closer, and the operator-on-duty shall know the exact location and telephone number of the LSO on duty.

Q. Reportable Occurrence - The occurrence of any facility condition that:

1. Causes a Limiting Safety System Setting to exceed the setting established in Section 2 of the Technical Specifications;

2. Exceeds a Limiting Condition for Operations as established in Section 3 of the Technical Specifications;

3. Causes any uncontrolled or unplanned release of radioactive material from the restricted area of the facility;

4. Results in safety system component failures which could, or threaten to, render the system incapable of performing its intended safety function as defined in the Technical Specifications or SAR;

5. Results in abnormal degradation of one of the several boundaries which are designed to contain the radioactive materials resulting from the fission processes;

6. Results in uncontrolled or unanticipated changes in reactivity of greater than 0.60$.

7. Causes conditions arising from natural or offsite manmade events that affect or threaten to affect safe operation of the facility, or;

8. Results in observed inadequacies in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the facility.

R. Review and Approve - The reviewing group or person 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 person.

S. Safety Channel - A measuring channel in the reactor safety system.

T. Secured Experiment - Any experiment, experimental facility, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor. The restraining forces must be equal to or greater than those that hold the fuel pins themselves in the reactor core.

U. Secured Shutdown - The reactor is secured and the facility administrative requirements are met for leaving the facility with no licensed operators present.

V. Shutdown Reactivity - The reactivity of the reactor at ambient conditions with all control rods fully inserted; including the reactivity of installed experiments.

W. Source - A neutron-emitting radioactive material, other than reactor fuel, which is positioned in or near the assembly to provide an external source of neutrons.

X. 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 (1982) shall not exceed the following:

1. Five-year (interval not to exceed six years).
2. Two-year (interval not to exceed two and one-half years).
3. Annual (interval not to exceed 15 months).
4. Semiannual (interval not to exceed seven and one-half months).
5. Quarterly (interval not to exceed four months).
6. Monthly (interval not to exceed six weeks).
7. Weekly (interval not to exceed ten days).
8. Daily (must be done during the calendar day).

Y. 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.

Z. True Value - The actual value at any instant of a process variable.

AA. Unsecured Experiment - Any experiment, experimental facility, or component or an experiment is deemed to be unsecured if it is not and when it is not secured. Moving parts of experiments are deemed to be unsecured when they are in motion.

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits - Fuel Pellet Temperature

Applicability

Applies to the maximum temperature reached in any in-core fuel pellet as a result of either normal operation or transient effects.

Objective

To identify the maximum temperature beyond which material degradation of the fuel and/or its cladding is expected.

Specification

Fuel pellet temperature at any point in the core, resulting from normal operation or transient effects, shall be limited to no more than 2000°C.

Bases

Specific determination of the melting point of the SPERT fuel has not been reported. A safety limit of 2000°C is below the listed melting point of UO_2 under a wide variety of conditions. The chosen value is conservative in view of variations that might result due to the presence of small quantities of impurities and the comparatively high vapor pressure of UO_2 at elevated temperatures. The safety limit specified is about 700°C below the measured melting point of UO_2 in a helium atmosphere.

2.2 Limiting Safety System Settings - Reactor Power

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	135 watts
Minimum Flux Level	2.0 counts/sec.
Minimum Period	5 seconds

Bases

The maximum power level trip setting of 135 watts on Log Power and Period Channel 2 (PP2) correlates with a reading of not greater than 90% on the highest scale of either of the two Linear Power Channels (LP1, LP2) as established by activation techniques. These scram setpoints ensure reactor shutdown and prevent significant energy deposition or enthalpy rise in the core in the event of any credible accident scenario.

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 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.

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Parameters

Applicability

These specifications apply to core parameters and reactivity coefficients.

Objective

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

Specifications

1. Above 100°F the isothermal temperature coefficient of reactivity shall be negative. The net positive reactivity insertion from the minimum operating temperature to the temperature at which the coefficient becomes negative shall be less than 0.15$.

2. The void coefficient of reactivity shall be negative, when the moderator temperature is above 100°F, within all standard fuel assemblies and have a minimum average negative value of 0.00043$/cc within the boundaries of the active fuel region.

3. The minimum operating temperature shall be 50°F.

Bases

The minimum absolute value of the temperature coefficient of reactivity is specified to ensure that negative reactivity is inserted when reactor temperature increases above 100°F. It is of note that even in the worst postulated accident scenarios, such as considered in Chapter 13 of the SAR, reactivity insertion because of temperature change would be negligible. The minimum average negative value of the void coefficient is specified to ensure that the negative reactivity inserted because of void formation is greater than that which was calculated in the SAR. The minimum operating temperature of 50°F establishes the temperature range for which the net positive reactivity limit can be applied.

3.2 Reactor Control and Safety Systems

Applicability

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

Objective

To assure that available shutdown method is adequate and that positive reactivity insertion rates are within those analyzed in the Hazards Summary Report (hereinafter safety analysis report).

Specifications

1. The excess reactivity of the reactor core above cold, clean critical shall not be greater than 0.60$. The maximum reactivity worth of any clean fuel pin shall be 0.20$.

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.

3. The maximum control rod reactivity rate shall be less than 0.12$/sec up to 10 times source level and 0.05$/sec at all higher levels.

4. 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 900 milliseconds. This time shall include a maximum magnet release time of 50 milliseconds.

5. The auxiliary reactor scram (moderator-reflector water dump) shall add negative reactivity within one minute of its activation.

6. The normal moderator-reflector water level shall be established not greater than 10 inches above the top grid of the core.

7. The minimum safety channels that shall be operating during the reactor operation are listed in Table 1.

8. 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.

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

10. The thermal power level shall be controlled so as not to exceed 100 watts, and the integrated thermal power for any consecutive 365 days shall not exceed 200 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[(a)]	1	Minimum Flux Level
Power:	10^{-4} - 150%	Linear Power	2	High Neutron Level Scram
	10^{-3} - 300%	Log-N; Period[(b)]	1	High Neutron Level and Period Scram
		Manual Scram[(c)]	2	Reactor Scram
		Building Power	1	Loss of Power
		Reactor Door Scram[(d)]	1	Reactor Scram

(a)May be bypassed when linear power channels are reading greater than $3x10^{-10}$ amps.
(b)During steady-state operation, this safety channel may be bypassed with the permission of the Operations Supervisor.
(c)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.
(d)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[a]	Prevents Control Rod Withdrawal
Neutron Flux 2 cps	Prevents Control Rod Withdrawal
Failure of 400 Cycle Synchro Power Supply	Prevents Control Rod Withdrawal
Failure of Line Voltage to Recorders	Prevents Control Rod Withdrawal
Moderator-Reflector Water Fill On	Prevents Control Rod Withdrawal

(a)These interlocks are available on only 1 of the 2 Log-N period Amplifiers and, therefore, may be bypassed with the permission of the Operations Supervisor if that one amplifier is out of service.

Bases

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 900 milliseconds for each control rod from its fully withdrawn position is specified to ensure that the insertion time does not exceed that assumed when establishing the minimum period of Specification 2.2 as a limiting safety system setting.

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 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 him in safely operating the reactor.

Limitations imposed on core reactivity, control rod worth, and reactor power preclude conditions that could allow the development of a potentially damaging accident. The limitations are conservative in view of core energy deposition, yet permit adequate flexibility in the research and instruction for which the facility is intended.

3.3 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. Instruments to continuously sample and measure the particulate activity in the reactor room atmosphere shall be operating whenever the reactor is to be operated.

   d. The radiation monitors required by 3.3.1 a, b, and c, may be temporarily removed from service if replaced by an equivalent portable unit.

2. Portable detection and survey instruments shall be provided.

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.

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

3.4 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 a material that may produce a violent chemical reaction and/or significant airborne radioactivity.

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 reactor coolant shall be encapsulated within corrosion resistant containers.

8. The radioactive material content of any singly encapsulated experiment shall be limited such that the complete release of all gaseous, particulate or volatile components directly to the reactor room will not result in exposures in excess of 10% of the equivalent annual exposures stated in 10 CFR 20 for persons

remaining in unrestricted areas for two hours or in restricted areas during the length of time required to evacuate the restricted area.

9. The radioactive material content of any doubly encapsulated experiment shall be limited such that the postulated complete release from the encapsulation or confining boundary of the experiment could not result in exposure in excess of applicable limits in 10 CFR 20 of any person occupying an unrestricted area continuously for a period of two hours from the time of release, or an exposure in excess of applicable limits in 10 CFR 20 for persons located within the restricted area during the length of time required to evacuate the restricted area.

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 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.

Specifications 8 and 9 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 20.101 and 10 CFR 20.103. The unrestricted area maximum is defined in 10 CFR 20.105 and 10 CFR 20.106.

4. SURVEILLANCE REQUIREMENTS

4.1 Reactor Parameters

Applicability

These specifications apply to the verification of control rod reactivity worths, temperature and void coefficients of reactivity, 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. control rod back reactivity worth;

b. temperature and void coefficients of reactivity;

c. reactor power measurement;

d. shutdown margin.

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 the initial test period of the reactor, measurements and calculations of core parameters will be for standard assemblies that are to be utilized in the reactor's operational program.

4.2 Reactor Control and Safety

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 and magnet release time shall be measured semiannually to verify that the requirements of Specification 3.2, Item 4, are met.

2. The moderator-reflector water dump time shall be measured semiannually to verify that the requirement of Specification 3.2, Item 5, is met.

3. All instrument channels, including 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 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, 4, and 5.

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. Since redundancy of all safety channels is provided, 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 Radiation Monitoring

Applicability

These specifications apply to the surveillance of the area and air radiation monitoring equipment.

Objective

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

Specification

The criticality detector system, area gamma monitors, and the mobile particulate air monitor shall be checked daily if the reactor is operated, tested monthly, and calibrated semiannually.

Bases

Experience has demonstrated that calibration of the criticality detectors, air gamma monitors, and the mobile air monitoring instrument 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.

5. DESIGN FEATURES

5.1 Site

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.

5.2 Facility

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.

5.3 Reactor Room

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.

5.4 Reactor

5.4.1 Reactor Tank

The stainless steel lined reactor tank 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.

5.4.2 Reactor Core

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) or approximately half of SPERT (F-1) fuel with the remainder (experiment) being made up of low enriched (4.81 w/o) uranium light water reactor type fuel of typical power reactor design and arrangement.

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.

5.4.3 Fuel Pins

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.

Any fuel pins used in an experiment shall consist of uranium fuel in the form of 4.81 weight percent or less enriched UO_2 pellets encapsulated in metal cladding.

5.4.4 Control Rod Assemblies

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, on 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 margin.

5.5 Water Handling 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.

5.6 Fuel Storage and Transfer

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 flooded with water.

Experimental fuel, when not in use, shall be stored in an approved sealed shipping container in the reactor room. Criticality and radiation analyses shall have been performed for this fuel in the shipping containers before delivery.

All fuel transfers shall be conducted under the direction of a licensed senior 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 system, 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 system, 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 one scale and the dump valve is not bypassed.

5. The critical rod bank position is checked after the operation is complete.

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 A.1.

Level 1: The Facility Director is responsible for the facility license and site administration.

Level 2: The Operations Supervisor is responsible for the reactor facility operation and management.

Level 3: Licensed senior operators are responsible for daily reactor operations.

Level 4: Licensed operators are the operating staff.

A health physicist who is organizationally independent of RPI operations group shall provide advice as required by the RPI Operations Supervisor in matters concerning radiological safety. The health physicist also has interdiction responsibility and authority.

6.1.2 Responsibility

The Operations Supervisor of the Rensselaer Polytechnic Institute Critical Experiment Facility shall be responsible for the safe operation of the facility. He shall be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, including these technical specifications.

In all matters pertaining to the operation of the reactor and these technical specifications, the Operations Supervisor shall report to and be directly responsible to the Facility Director.

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 be 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.

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

(b) 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.

(c) Events requiring the direction of the Operations Supervisor:

1) All fuel or control rod relocations within the reactor core.

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-15.4-1977, 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 Qualification

The NSRB shall have at least four members of whom no more than the minority shall be from the line organization shown in Figure A.1. The board shall be made up of senior personnel who shall collectively provide a broad spectrum of expertise in reactor technology. Qualified and approved alternates may serve in the absence of regular members.

6.2.2 Charter and Rules

The Review Board shall function under the following rules:

(a) The Chairman of the NSRB shall be approved by the Facility Director.

(b) The Board shall meet at least semiannually.

(c) The quorum shall consist of not less than a majority of the full Board and shall include the Chairman or his designated alternate.

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

6.2.3 Review and Approval Function

The following items shall be reviewed and approved 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.

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 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 shutdown 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.

6.4 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 shall be documented.

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.

6.5 Required Actions

6.5.1 Action to be taken in Case of Safety Limit Violations

(a) The reactor shall be shutdown, and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

(b) The safety limit violation shall be promptly reported to the level one authority or designated alternates and to the NSRB.

(c) The safety limit violation shall be reported to the Nuclear Regulatory Commission in accordance with Section 6.5.3.

(d) A safety limit violation report shall be prepared. The report shall describe the following:

1) Applicable circumstances leading to the violation, including, when known, the cause and contribution factors.

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

3) Corrective action to be taken to prevent recurrence.

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

6.5.2 Action to be Taken in the Event of an Occurrence of the Type Identified in Section 1.0 Q (Reportable Occurrence)

(a) Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the Facility Director or designated alternate.

(b) Occurrence shall be reported to the Facility Director or designated alternates and to the Commission as required.

(c) All such conditions, including action taken to prevent or reduce the probability of a recurrence, shall be reviewed by the NSRB.

6.6 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.

6.6.1 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(a), and;

6) Changes in the plant operating staff serving in the following positions:

a) Facility Director;

b) Operations Supervisor;

c) Health Physicist;

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 (excluding 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(b).

(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).

6.6.2 Non-Routine Reports

(a) Reportable Operational Occurrence Reports. Notification shall be made within 24 hours by telephone and e-mail to the Administrator of Region I, followed by a written report within 10 days in the event of a reportable operational occurrence as defined in Section 1.0. 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 shall be forwarded within 30 days to the Administrator of Region I in the event of: (1) 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.

6.7 Operating Records

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

(a) Normal facility operation and maintenance.

(b) Reportable operational occurrences.

(c) Tests, checks, and measurements documenting compliance with surveillance requirements.

(d) Records of experiments performed.

(e) Records of radioactive shipments.

6.7.2 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).

(d) Fuel inventories, offsite transfers and in-house transfers if they are not returned to their original core or vault location during the experimental program in which the original transfer was made.

(e) Facility radiation and contamination surveys.

(f) The present as-built facility drawings and new updated or corrected versions.

(g) Minutes of Nuclear Safety Review Board meetings.

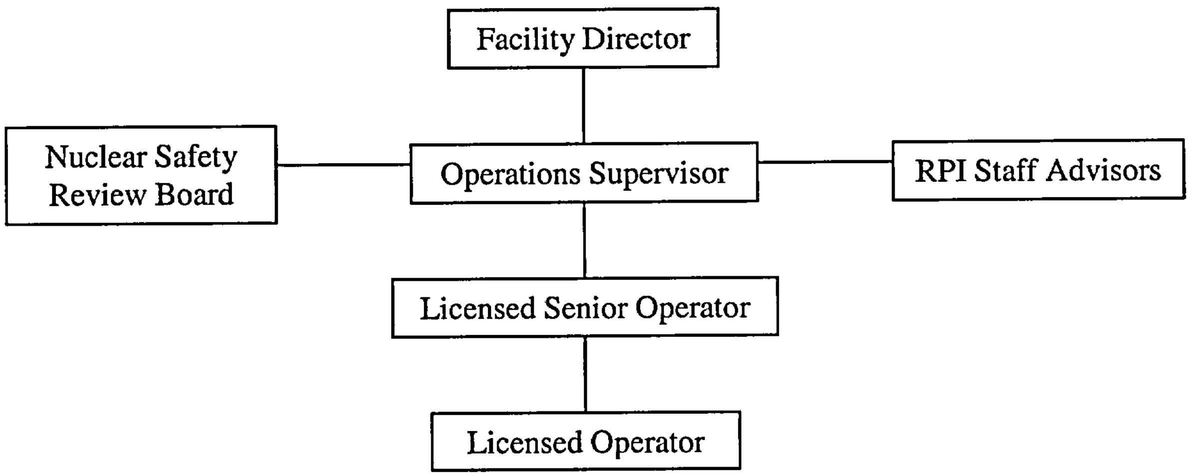

Figure A.1: RCF Management Organization