UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PURDUE UNIVERSITY

DOCKET NO. 50-182

RENEWED FACILITY OPERATING LICENSE

License No. R-87

1. The U.S. Nuclear Regulatory Commission ("the Commission") has found that:

   A. The application for renewal of Facility Operating License No. R-87 filed by Purdue University ("the licensee"), dated July 7, 2008, as supplemented by letters dated June 4, 2010; November 15, 2011; January 4, January 30, January 31, June 1, June 15, June 29, July 13, and August 11, 2012; April 10, 2013; July 24, 2015; and January 29, February 26, March 31, May 9, July 7, July 19, September 19, and September 29, 2016 ("the application"), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the Commission's rules and regulations set forth in Title 10, Chapter I, of the *Code of Federal Regulations* (10 CFR);

   B. Construction of the Purdue University Research Reactor ("the facility") was completed in substantial conformity with the Construction Permit No. CPRR-64, issued to Purdue University on August 7, 1961, and the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

   C. The facility will operate in conformity with the application, as supplemented, the provisions of the Act, and the rules and regulations of the Commission;

   D. There is reasonable assurance that: (i) the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) such activities will be conducted in compliance with the Commission's regulations;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this license in accordance with the rules and regulations of the Commission;

   F. The licensee is a nonprofit educational institution and will use the facility for the conduct of educational activities, and has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. The issuance of this license is in accordance with 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession and use of byproduct and special nuclear materials as authorized by this facility operating license will be in accordance with the Commission's regulations in 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

2. Accordingly, Facility Operating License No. R-87 is hereby renewed in its entirety to read as follows:

A. This license applies to the Purdue University Research Reactor, a pool-type nuclear reactor (herein "the facility") owned by Purdue University (herein "the licensee"). The facility is located on the campus of Purdue University, in the city of West Lafayette, Tippecanoe County, Indiana and is described in the licensee's application for license renewal, dated July 7, 2008, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Purdue University as follows:

1. Pursuant to subsection 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location in West Lafayette, Indiana, in accordance with the procedures and limitations described in the application and set forth in this license;

2. Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use, but not separate: (1) up to 3.8 kilograms of contained uranium-235 of enrichment of less than 20 percent in the form of materials testing reactor (MTR)-type fuel; (2) up to 80.0 grams of plutonium contained in encapsulated plutonium-beryllium sources; and (3) up to 100 grams of contained uranium-235 of any enrichment in the form of fission chambers, flux foils and fueled experiments, all used in connection with the operation of the facility;

3. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in in 10 CFR Parts 20, 30, 50, 51, 55, 70, and 73 of the Commission's regulations; is subject to all applicable provisions of the Act and to the rules, regulations and, orders of the Commission now or hereafter in effect and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

The licensee is authorized to operate the facility at steady-state power levels not in excess of 12 kilowatts (thermal).

2. Technical Specifications

The Technical Specifications contained in Appendix A, as revised by Amendment No. 16, are, hereby, incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Physical Security Plan

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security plan, including all amendments and revisions made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). This approved physical security plan, which contains information withheld from public disclosure under 10 CFR 73.21, "Protection of Safeguards Information: Performance Requirements," is entitled "The Purdue University Reactor-1 Physical Security Plan," and is dated March 2016.

4. Reactor Protection and Control Systems

Purdue University shall perform verification and validation testing, factory acceptance testing, and site acceptance testing described in the application for license amendment dated February 27, 2017, as supplemented by letters dated December 18, 2017, and March 2, 2019, on the reactor protection and control systems. The test results and any action taken to correct deficiencies shall be reviewed by the Purdue University Committee on Reactor Operations and reviewed and approved by the Facility Director prior to resuming operation of the reactor.

D. This license is effective as of the date of issuance and shall expire at midnight, twenty years from its date of issuance.

For the Nuclear Regulatory Commission

/RA/

William M. Dean, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A, Technical Specifications

Date of Issuance: October 31, 2016

TECHNICAL SPECIFICATIONS

FOR THE

PURDUE UNIVERSITY REACTOR, PUR-1

DOCKET NUMBER 50-182

FACILITY LICENSE NO. R-87

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April 2019

Amendment No. 14
April 1, 2019

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1. DEFINITIONS

The following frequently used terms are to aid in the uniform interpretation of these specifications:

1.1 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.

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

1.3 Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification may include comparison of the channel with other independent channels or methods of measuring the same variable.

1.4 Channel Test - A channel test is the introduction of a simulated signal into a channel to verify that it is operable.

1.5 Confinement – Confinement is an enclosure of the overall facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled or defined pathways.

1.6 Core Configuration – The core configuration includes the number, type, or arrangement of fuel assemblies (elements), reflector elements, reflector element configuration, and regulating/control rods occupying the core grid.

1.7 Core Experiment - A core experiment is one placed in the core, in the graphite reflector, or within six inches (measured horizontally) of the reflector. This includes any experiment in the pool directly above or below the core.

1.8 Direct Supervision – In visual and audible contact.

1.9 Excess reactivity – Excess reactivity is that amount of reactivity that would exist if all control rods 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.

1.10 Experiment – Any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the pool, on or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.

1.11 Experimental Facility - Experimental facilities are:

a. those regions specifically designated as locations for experiments or

b. systems designed to permit or enhance the passage of a beam of radiation to another location.

1.12 Experiment With Movable Parts (Secured or Nonsecured) - An experiment with movable parts is an experiment that contains parts that are intended to be moved while the reactor is operating.

1.13 Explosive Material - Explosive material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in "Dangerous Properties of Industrial Materials" by N. I. Sax, Tenth Ed. (2000), or is given an Identification of Reactivity (Stability) index of 2, 3 or 4 by the National Fire Protection Association in its publication 704, "Identification System for Fire Hazards of Materials."

1.14 Fueled Experiment - A fueled experiment is any experiment planned for irradiation of uranium 233, uranium 235, plutonium 239, or plutonium 241.

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

1.16 Licensed – See licensee.

1.17 Licensee – An individual or organization holding a license.

1.18 Measured Value - The measured value is the value of a parameter as it appears at the output of a channel.

1.19 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.

1.20 New Experiment - A new experiment is one whose nuclear characteristics have not been experimentally determined.

1.21 Non-secured Experiment – See Unsecured Experiment.

1.22 Operable - A system or component is operable when it is capable of performing its intended function in a normal manner.

1.23 Operating - A system or component is operating when it is performing its intended function.

1.24 Pool Experiment - A pool experiment is one positioned within the pool more than six inches (measured horizontally) from the graphite reflector.

1.25 Power Level – There are three important and separately defined power levels.

a. Instantaneous Power Level shall be the power level of the reactor at any given moment, as indicated by the reactor instrumentation.

b. The Operating Level shall be the power level from which setpoints for scram and setback shall be calculated. The Operating power level shall be 10 kW or less.

c. The Maximum Power Level shall be the maximum instantaneous power level allowed by the PUR-1 License. The Maximum Power Level shall be 12 kW, which shall not be exceeded.

1.26 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.

1.27 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 or removed from its intended position.

1.28 Reactor Facility - The reactor facility is that portion of the ground floor of the Duncan Annex of the Electrical Engineering Building occupied by the School of Nuclear Engineering used for activities associated with the reactor.

1.29 Reactor Operating – The reactor is operating whenever it is not secured or shut down.

1.30 Reactor Operator – An individual who is licensed to manipulate the controls of the reactor.

1.31 Reactor Safety System - The reactor safety system is that combination of measuring channels and associated circuitry which forms the automatic protective system of the reactor, or provides information which requires manual protective action to be initiated.

1.32 Reactor Secured – A reactor is secured when

a. *Either* there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection

b. *Or* the following conditions exist:

1. Both shim-safeties and the regulating rod shall be fully inserted

2. Electrical power to the control rod circuits shall be switched off

3. The reactor key shall be out of the key switch and under control of a licensed operator or locked in an approved location

4. No work shall be in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods

5. No experiments shall be moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment

6. The control console is placed in a permissions status where the controls are not operable.

1.33 Reactor Shutdown - That subcritical condition of the reactor where the negative reactivity, with or without experiments in place, is equal to or greater than the shutdown margin.

1.34 Readily Available on Call - Readily available on call shall mean the licensed senior operator shall be within a reasonable driving time (1/2 hour) or less than 15 miles from the reactor building, and the operator on duty is currently informed, and can rapidly contact the senior reactor operator by phone.

1.35 Reference core condition - The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible ($<0.003\ \Delta k/k$).

1.36 Removable Experiment - A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system, which can reasonably be anticipated to be moved one or more times during the life of the reactor.

1.37 Rod, control - A control rod is a device fabricated from neutron-absorbing material that is used to establish neutron flux changes and to compensate for routine reactivity losses. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged.

1.38 Rod, regulating - The regulating rod is a low worth control rod used primarily to maintain an intended power level that need not have scram capability. Its position may be varied manually or by a servo-controller.

1.39 Rod, Shim-Safety - The control rods used in PUR-1 as described in the definition for Rod, control.

1.40 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 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 which are normal to the operating environment of the experiment, or by forces which can arise as a result of credible malfunctions.

1.41 Senior Reactor Operator – An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

1.42 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.

1.43 Shutdown Margin – The 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.

1.44 Surveillance and Test Intervals - These are intervals established for periodic surveillance and test actions. Established intervals shall be maintained on the average. Maximum intervals are allowed to provide operational flexibility, not to reduce frequency.

1.45 Tried Experiment - A tried experiment is:

a. An experiment previously performed in this facility, or

b. An experiment of approximately the same nuclear characteristics as an experiment previously tried. These nuclear characteristics include but are not limited to neutron activation cross-sections, absorption cross-sections, and moderating ability.

1.46 True Value -The true value of a parameter is its exact value at any instant.

1.47 Unscheduled Shutdown – An unscheduled shutdown is defined as any unplanned shutdown of the reactor 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.

1.48 Unsecured Experiment - Any experiment, experimental facility, or component of an experiment is considered to be unsecured when it is not secured as defined in this section.

2. SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit

Applicability - This specification applies to the temperature of the reactor fuel and cladding under any condition of operation.

Objective - The objective is to ensure fuel cladding integrity.

Specification – The fuel and cladding temperatures shall not exceed 530°C (986°F).

Basis – Safety limits for nuclear reactors are limits upon important process variables that are necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The principal physical barrier is the fuel cladding.

In the Purdue University Reactor, the first and principal barrier protecting against release of radioactivity is the cladding of the fuel plates. The 6061 aluminum alloy cladding of the LEU fuel plates has an incipient melting temperature of 582°C. However, measurements (NUREG-1313) on irradiated fuel plates have shown that fission products are first released near the blister temperature (~550°C) of the cladding. To ensure that the blister temperature is never reached, NUREG 1537 concludes that 530°C is an acceptable fuel and cladding temperature limit not to be exceeded under any condition of operation.

2.2 Limiting Safety System Setting

Applicability - This specification applies to the reactor power level safety system setting for operation.

Objective - The objective is to assure that the safety limit is not exceeded.

Specification – The measured value of the power level scram shall be no higher than 12.0 kW.

Basis - The LSSS has been chosen to assure that the automatic reactor protective system will be actuated in such a manner as to prevent the safety limit from being exceeded during the most severe expected abnormal condition.

The function of the LSSS is to prevent the temperature of the reactor fuel and cladding from reaching the safety limit under any condition of operation. During steady-state operation, a power level of 98.6 kW is required to initiate the onset of nucleate boiling. This is far higher than the maximum power of 18 kW, which allows for 50% instrument uncertainties in measuring power level.

For the transients that were analyzed, the temperature of the fuel and cladding reach maximum temperatures of 43.20°C, assuming reactor trip at 18 kW after failure of the first trip. This temperature is far below the safety limit of 530°C.

3. LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability - These specifications apply to the reactivity conditions of the reactor, and the reactivity worths of control rods and experiments

Objective - The objective is to assure that the reactor can be shut down at all times, that the safety limits will not be exceeded, and that operation is within the limits analyzed in the SAR.

Specification - The reactor shall not be operated unless the following conditions exist:

a. The shutdown margin, relative to the reference core condition with the most reactive shim rod fully withdrawn, and the regulating rod fully withdrawn shall be at least 0.010 $\Delta k/k$.

b. The reactor shall be subcritical by more than 0.03 $\Delta k/k$ during core loading changes.

c. No shim-safety rod shall be removed from the core if the shutdown margin is less than 0.01 $\Delta k/k$ with the remaining shim-safety rod fully withdrawn.

d. The reactor shall be shutdown if the maximum positive excess reactivity of the core and any installed experiment exceeds 0.006 $\Delta k/k$.

e. The absolute value of the reactivity worth of each experiment shall be limited as follows:

Experiment	Maximum Reactivity Worth
Movable	0.003 $\Delta k/k$
Unsecured	0.003 $\Delta k/k$
Secured	0.004 $\Delta k/k$

f. The sum of the absolute value of the total worth of all movable and unsecured experiments shall not exceed 0.003 $\Delta k/k$.

g. The sum of the absolute value of the total worth of all secured experiments shall not exceed 0.005 $\Delta k/k$.

Bases - The shutdown margin required by Specification 3.1.a assures that the reactor can be shut down from any operating condition and will remain shut down even if the control rod of the highest reactivity worth should be in the fully withdrawn position.

Specifications 3.1.b and 3.1.c provide assurance that the core will remain subcritical during loading changes and shim-safety rod maintenance or inspection.

Specification 3.1.d limits the allowable excess reactivity to the value assumed in the SAR. This limit assures that the consequences of reactivity transients will not be increased relative to transients previously reviewed, and assures reactor change rate of minimal magnitude such that the reactor may be shutdown without exceeding the safety limit.

Specification 3.1.e limits the reactivity worth of secured experiments to values of reactivity which, if introduced as a positive step change, are calculated not to cause fuel melting. This specification also limits the reactivity worth of unsecured and movable experiments to values of reactivity which, if introduced as a positive step change, would not cause the violation of a safety limit. The manipulation of experiments worth up to 0.003 $\Delta k/k$ will result in reactor change rates smaller than 12 percent per second. This change rate is sufficient to initiate a setback but not a scram. These change rates can be readily compensated for by the action of the safety system without exceeding any safety limits.

A limitation of 0.003 $\Delta k/k$ for the total reactivity worth of all movable and unsecured experiments provides assurance that a common failure affecting all such experiments cannot result in an accident of greater consequences than the maximum credible accident analyzed in the HSR.

Specification 3.1.g along with 3.1.a assures that the reactor is capable of being shut down in the event of a positive reactivity insertion caused by the flooding of an experiment.

3.2 Reactor Safety System

Applicability - This specification applies to the reactor safety system and other safety-related instrumentation.

Objective - The objective is to specify the lowest acceptable level of performance or the minimum number of acceptable components for the reactor safety system and other safety related instrumentation.

Specification – The two shim-safeties shall not be moved more than 6 cm from the fully inserted position unless the following conditions are met:

a. The reactor safety channels and safety-related instrumentation shall be operable in accordance with Tables I and II including the minimum number of channels and the indicated maximum or minimum set points.

b. Both shim-safety rods and the regulating rod shall be operable.

c. The time from the initiation of a scram condition in the scram circuit until the shim-safety rod reaches the rod lower limit switch shall not exceed one second.

d. The pool top radiation monitor shall be capable of indicating an alarm to off-site reactor staff when a high limit is reached and the reactor has been secured. The alarm may be out of service up to thirty days. Loss of functionality beyond thirty days shall require a visual pool level inspection in intervals of 24 hours, not to exceed 30 hours.

e. Building alternating current power must be supplied to the reactor Instrumentation and Control during normal operation. Loss of power shall require immediate shutdown by the operator to be completed within an interval of 15 minutes.

TABLE I. SAFETY CHANNELS REQUIRED FOR OPERATION

Channel	Minimum Number Required	Setpoint (c)(d)	Function
Log count rate and change rate	1[a]	2 cps or greater 8 %/s or less 15 %/s or less 6 %/s or less	2 cps rod withdrawal interlock Setback Scram Rod withdrawal interlock
Log N and change rate	1[b]	8 %/s or less 15 %/s or less 6 %/s or less 12kW, 120% Operating power level, or less	Setback Scram Rod withdrawal interlock Scram
Linear	1	0% Selected Range, or greater 110% Selected Range or less 120% Selected Range or less	Setback Setback Scram
Safety	1[b]	11 kW, 110% Operating power level, or less 12 kW, 120% Operating power level, or less	Setback Scram
Manual Scram (console) (hallway)	 1 1		 Scram Scram

(a) Not required after Log N-Change Rate channel comes on scale.
(b) Required to be operable but not on scale at startup.
(c) All percentage based setpoints shall be tripped when the measured value is greater than or equal to the specified value. Counts per second (cps) setpoints are at values less than or equal to the specified value. Exception: Trip point for 0% shall happen as the value goes from the positive to negative value.
(d) Setbacks shall be set such that they will be initiated prior to a Scram

TABLE II. SAFETY-RELATED CHANNELS (AREA RADIATION MONITORS)

Channel	**Minimum Number Required(e)**	**Setpoint**	**Function**
Pool top monitor	1	50 mR/hr, 2x full power background, or less than either	Scram
Water process	1	7 ½ mR/hr or less	Scram
Console Monitor	1	7 ½ mR/hr or less	Scram
Continuous air sampler	1	Stated on sampler	Air sampling

(e) For periods of one week or for the duration of a reactor run, a radiation monitor may be replaced by a gamma sensitive instrument which has its own alarm and is observable by the reactor operator.

Bases - The neutron flux level scrams provide redundant automatic protective action to prevent exceeding the safety limit on reactor power, and the change rate scram conservatively limits the rate of rise of the reactor power to values which are manually controllable without reaching excessive power levels or fuel temperatures.

The rod withdrawal interlock on the Log Count Rate and Change Rate Channel assures that the operator has a measuring channel operating and indicating neutron flux levels during the approach to criticality.

The manual scram button and the "reactor on" key switch provide two methods for the reactor operator to manually shut down the reactor if an unsafe or abnormal condition should occur and the automatic reactor protection does not function.

The use of the area radiation monitors (Table II) will assure that areas of the Purdue University Reactor (PUR-1) facility in which a potential high radiation area exists are monitored. These fixed monitors initiate a scram whenever the preset alarm point is exceeded to avoid high radiation conditions as well as alert facility personnel when the reactor has been secured and an elevated radiation level exists. Use of more conservative values are permitted on the setpoint to allow greater safety margin.

Specifications 3.2.b and 3.2.c assure that the safety system response will be consistent with the assumptions used in evaluating the reactor's capability to withstand the maximum credible accident.

In specification 3.2.c. the rod lower limit switches are positioned to measure, as close as possible, the fully inserted position.

Shielding from radiation is one of the primary reasons for the pool's level. An offsite alarm from the pool top radiation monitor alerts facility staff of a rising radiation level which must be mitigated or otherwise addressed and this is addressed in 3.2.d.

The Instrumentation and Control system is designed to be capable of performing a normal shutdown in the event of a loss of off-site power. The time period to complete this shutdown is up to 30 minutes as that is the rating of the UPS units with a maximum power loading. Loss of off-site power starts the power draw from the UPS units which therefore starts the 30 minute timeframe.

3.3 **Primary Coolant Conditions**

Applicability - This specification applies to the limiting conditions for reactor operation for the primary coolant.

Objective - The objective is to assure a compatible environment, adequate shielding, and a continuous coolant path for the reactor core.

Specification –

a. The primary coolant conductivity shall be maintained at a value less than 3 $\mu Siemens/cm$.

b. The primary coolant shall be maintained at least 13 feet above the core whenever the reactor is operating. The primary coolant shall be maintained at least 13 feet above the top of the core or at a level sufficient for the pool top radiation monitor to indicate less than 1 mRem/hour during non-operational periods.

c. The primary coolant (bulk pool volume) shall be maintained at or below $30\ °C$ while the reactor is operating.

d. The primary coolant radiation levels shall not exceed the levels for water in 10 CFR 20 Appendix B, Table 2.

Bases - Experience at the PUR-1 and other facilities has shown that the maintenance of primary coolant system water quality in the ranges specified in specification 3.3.a will minimize the amount and severity of corrosion of the aluminum components of the primary coolant system and the fuel element cladding.

The height of water in specification 3.3.b is enough to furnish adequate shielding as well as to guarantee a continuous coolant path.

Maintaining the primary coolant temperature in Specification 3.3.c will ensure the margin to the onset of nucleate boiling is maintained and analyses shown in the Safety Analysis Report remain valid.

Limiting the amount of radioactivity in the primary coolant minimizes the health risk to the public as well as to facility personnel.

3.4 **Confinement**

Applicability - This specification applies to the integrity of the reactor room.

Objective - The objective is to limit and control the release of airborne radioactive material from the reactor room.

Specification -

a. During reactor operation and when radioactive material is being handled with potential for airborne release, the following conditions shall be met:

1. The reactor room shall be maintained at a negative pressure of at least 0.05 inches of water with the operation of the room exhaust fan.

2. All exterior doors in the reactor room shall remain closed except as required for personnel, equipment, or materials access.

b. All inlet and exhaust air ducts and the sewer vent shall contain a HEPA filter or its equivalent.

c. Dampers in the ventilation system inlet and outlet ducts shall be capable of being closed.

d. Concentration of Ar-41 shall not exceed $2.08 \times 10^{-7}\ \mu Ci/cm^3$ at the top of the confinement exhaust stack.

Bases - The PUR- 1 does not rely on a containment building to reduce the levels of airborne radioactive material released to the environment in the event of the maximum hypothetical accident. However, in the event of such an accident, a significant fraction of the airborne material will be confined within the reactor room, and the specifications stated above will further reduce the release to the environment.

The limit on the concentration of Argon at the top of the confinement exhaust stack is the maximum theoretical concentration of the isotope and therefore a fan malfunction ventilating the room would be the only way to violate this technical specification. It is validated by the dose readings obtained through the effluent surveillances in section 4.7.

3.5 **Limitations on Experiments**

Applicability- This specification applies to experiments installed in the reactor and its experimental facilities.

Objective - The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure, and to assure the safe operation of the reactor.

Specification - The reactor shall not be operated unless the following conditions are met:

a. All experiments shall be constructed of material which will be corrosion resistant for the duration of their residence in the pool.

b. All experiments and experimental procedures shall receive approval by the Committee on Reactor Operations.

c. Known explosive materials shall not be placed in the reactor pool.

d. No experiment shall be placed in the reactor or pool that interferes with the safe operation of the reactor.

e. Any failure of an experiment shall not have a consequence that could exceed dose limits as set forth in 10 CFR Part 20, as analyzed and approved by the Reactor Supervisor and the Committee on Reactor Operations.

f. A fueled experiment shall not produce more than 0.5 Curies of radio-iodine.

Bases - Specification 3.5.a through 3.5.f are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from experiment failure and serve as a guide for the review and approval of new experiments by the facility personnel and the Committee on Reactor Operations.

Limiting the amount of radio-iodine levels in a fueled experiment will ensure that the Maximum Hypothetical Accident analyzed in the Safety Analysis Report remains the bounding incident which could occur at the PUR-1.

3.6 **Fuel Parameters**

Applicability- This specification applies to fuel plates installed in the reactor and in use during the previous surveillance period.

Objective - The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of a fuel cladding failure, and to assure the safe operation of the reactor.

Specification - The reactor shall only be operated when the following specifications have been met:

a. The inspection of fuel assemblies shall be performed to identify any abnormal or previously undocumented defect present on a fuel plate. These defects may include but are not limited to blistering of the cladding on the fuel plate from elevated temperatures beyond the design of the cladding, deep scratches or gouges on the plate due to debris in the coolant flowing along the face, scratches on the edges of the plate due to insertion and removal from the assembly, discoloration from the deposition of particulates within the coolant or corrosion of the plate itself.

b. The reactor shall not operate with fuel plates that have been determined to be unsound for use as outlined above in 3.6.a. These plates shall be removed from service and the manufacturer consulted to determine possible causes.

Bases – The fuel parameter Limiting Condition on Operation is intended to limit the possibility radioactivity releases resulting from fuel failure by identifying issues prior to potential release.

4. SURVEILLANCE REQUIREMENTS

4.1 Reactivity Limits

Applicability - This specification applies to the surveillance requirements for reactivity limits.

Objective - The objective is to ensure that the reactivity limits of Specification 3.1 are not exceeded.

Specification -

a. The shim-safety rod reactivity worths shall be measured and the shutdown margin calculated biennially with no interval to exceed 2½ years and whenever a core configuration is loaded for which shim-safety rod worths have not been measured. This may be deferred with CORO approval during any extended reactor shutdown. Additionally, if a new rod is used, its worth must be measured on the first start-up following installation. In the case of a deferred measurement, the measurement must be performed prior to resuming routine reactor operations.

b. The shim-safety rods shall be visually inspected biennially with no interval to exceed 2½ years, which may be deferred with CORO approval during any reactor shutdown. If the rod is found to be deteriorated, it shall be replaced with a rod of approximately equivalent or greater worth, meeting the limiting conditions of operation specified in 3.1. In the case of a deferred measurement, the measurement must be performed prior to resuming routine reactor operations.

c. The reactivity worth of experiments placed in the PUR-1 shall be measured during the first startup subsequent to the experiment's insertion and shall be verified if core configuration changes cause increases in experiment reactivity worth which may cause the experiment worth to exceed the values specified in Specification 3.1

Bases - Specification 4.1.a will assure that shim-safety rod reactivity worths are not degraded or changed by core manipulations which cause these rods to operate in regions where their effectiveness is reduced.

The boron stainless steel shim-safety rods have been in use at the PUR-1 since 1962, and over this period of time, no cracks or other evidence of deterioration have been observed. Based on this performance and the experience of other facilities using similar shim-safety rods, the specified inspection times are considered adequate to assure that the control rods will not fail.

4.2 Reactor Safety System

Applicability - This specification applies to the surveillance of the reactor safety system.

Objective - The objective is to assure that the reactor safety system is operable as required by Specification 3.2

Specification –

a. A channel calibration of the reactor safety channels as described in Table I shall be performed as follows:

   1. An electronic calibration shall be performed annually, with no interval to exceed 15 months. The electronic calibration may be deferred with CORO approval during periods of reactor shutdown, but shall be performed prior to startup.

   2. A power calibration by foil activation shall be performed annually, with no interval to exceed 15 months. The power calibration may be deferred with CORO approval during periods of reactor shutdown, but shall be performed as soon as practicable after reactor startup.

b. A channel check on the radiation monitoring equipment shall be completed daily during periods when the reactor is in operation. Calibration of the Safety-Related Channels specified in Table II and hand held radiation survey instruments shall be performed annually, with no interval to exceed 15 months. Calibration may be deferred with CORO approval during periods of reactor shutdown, but shall be performed prior to startup.

c. Shim-safety rod drop times shall be measured annually, with no measurement's interval to exceed 15 months. These drop times shall also be measured prior to operation following maintenance which could affect the drop time or cause movement of the shim-safety rod control assembly. Drop times may be deferred with CORO approval during periods of reactor shutdown, but shall be performed prior to startup.

d. A channel check of each of the Scram capabilities specified in Table I shall be performed prior to each day's startup.

e. A channel check of the pool top radiation monitoring equipment's off-site alarm capability shall be done biannually, not to exceed 7 ½ months.

f. A simulated loss of off-site power shall be performed annually with no interval to exceed 15 months to verify the UPS units are capable of providing Instrumentation and Control power for at least 30 minutes.

g. Appropriate surveillance testing on any technical specification required system shall be conducted after replacement, repair, or modification before the system is considered operable and returned to service unless reactor operation is required for the performance of the surveillance, whereby it shall be done as soon as practicable after reactor startup.

Bases - A test of the safety system channels prior to each startup will assure their operability, and annual calibration will detect any long-term drift that is not detected by normal intercomparison of channels. The channel check of the neutron flux level channel will assure that changes in core-to-detector geometry or operating conditions will not cause undetected changes in the response of the measuring channels.

Area monitors will give a clear indication when they are not operating correctly. In addition, the operator routinely records the readings of these monitors and will be aware of any reading which indicates loss of function.

Reactor operations are required to perform power calibrations. The Technical Specification requirement "as soon as practicable" prohibits any other reactor operations other than those to perform the power calibration. This conditional ensures the only operation following replacement, repair, or modification is to perform the power calibration and before continuing normal work.

The area monitoring system employed at the PUR-1 has exhibited very good stability over its lifetime, and annual calibration is considered adequate to correct long-term drift.

The measured drop times of the shim-safety rods have been consistent since the PUR-1 was built. An annual check of this parameter is considered adequate to detect operation with materially changed drop times. Binding or rubbing caused by rod misalignment could result from maintenance; therefore, drop times will be checked after such maintenance.

A daily check of the scram functionality ensures functionality of the system.

Annual checks of the UPS unit functionality will verify the UPS units are capable of providing power for at least 30 minutes. Changes in battery functionality are expected to be nominal over a period of several years. Therefore the frequency of an annual check is sufficient to verify operability of the controlled shutdown condition during loss of off-site power.

Testing replaced, repaired, or otherwise modified systems shall be done to ensure their adequate performance with the integral reactor safety and control system. Appropriate surveillance testing is taken to mean actions which provide reasonable assurance it will provide any required protective function and not inhibit other systems from performing their respective functionality.

4.3 **Primary Coolant System**

Applicability - This specification applies to the average surveillance schedules of the primary coolant system.

Objective - The objective is to assure high quality pool water, adequate shielding, and to detect the release of fission products from fuel elements.

Specification –

a. The conductivity of the primary coolant shall be recorded monthly, not to exceed six weeks. This cannot be deferred during reactor shutdown.
b. The primary coolant shall be sampled monthly, not to exceed six weeks, and analyzed for gross alpha and beta activity. This cannot be deferred during reactor shutdown.
c. During reactor shutdown, the primary coolant level or radiation level shall be monitored monthly with an interval not to exceed six weeks. Primary coolant height shall be measured prior to reactor operation.
d. The Primary Coolant temperature shall be recorded in the log book at no interval to exceed four hours if any shim-safety or regulating rod is at a height greater than 6 cm.

Bases - Monthly surveillance of pool water quality provides assurance conductivity changes will be detected before significant corrosive damage could occur.

When the reactor pool water is at a height of 13 feet above the core, adequate shielding during operations is assured. Experience has shown that approximately 35-40 gallons of water will evaporate weekly and weekly water make-up is sufficient to maintain the reactor pool water height. Analysis has shown radiation levels to remain sufficiently low with excessive water loss during non-operational periods.

Analysis of the reactor water for gross alpha and beta activity assures against undetected leaking fuel assemblies.

4.4 **Confinement**

Applicability - This specification applies to the surveillance requirements for maintaining the integrity of the reactor room.

Objective - The objective is to assure that the integrity of the reactor room is maintained, by specifying average surveillance intervals.

Specification -

a. The negative pressure of the reactor room shall be recorded weekly.

b. Operation of the inlet and outlet dampers shall be checked semiannually, with no interval to exceed 7 1/2 months.

c. Operation of the air conditioner shall be checked semiannually, with no interval to exceed 7 1/2 months.

Bases - Specification a, b, and c check the integrity of the reactor room. Based upon past experience these intervals have been shown to be adequate for ensuring the operation of the systems affecting the integrity of the reactor room.

4.5 **Experiments**

Applicability - This specification applies to the surveillance of limitations on experiments.

Objective – To assure compliance with the provision of the utilization license, the Technical Specifications, and 10 CFR Parts 20 and 50.

Specification – No experiments shall be performed unless:

a. It is a tried experiment.

b. The experiment has been properly reviewed and approved according to Section 6 of the technical specifications.

   1. Proposed experiments shall be approved by the Committee on Reactor Operations

   2. Submitted proposed experiments shall provide a comprehensive list of steps to be performed, quantities to be measured, hazards to be considered, limiting initial conditions of the reactor, and required available personnel.

Bases - The basis for this specification is to ensure the safety of the reactor and associated components, personnel, and the public by verification of proper review and approval of experiments as specified in Section 6 of these technical specifications.

4.6 **Fuel Parameters**

Applicability - This specification applies to the surveillance requirements for fuel integrity.

Objective - The objective is to assure that the fuel clad remains unblemished and there has been no release of radioactivity to the reactor coolant or facility.

Specification - Representative fuel plates shall be inspected annually, with no interval to exceed 15 months through visual inspection of the assembly. Representative is set forth to mean at least one plate from the assembly expected to have the highest burn as well as a plate from one of the 12 remaining, non-control assemblies.

Bases - Specification 4.6 will ensure reactor fuel integrity is not compromised. The inspection period is set forth to verify the integrity of the fuel cladding thereby ensuring there are no unexpected releases of fission products exposing facility workers or members of the public. Inspection of an assembly from the highest power region (as outlined in the PUR-1 SAR) ensures those plates under the largest thermal stress are considered. Inspection of another assembly ensures that a single plate passing inspection does not provide a single false negative data point representing the entire core. Non-control assemblies are chosen to inhibit undue burden on the facility.

4.7 **Effluents**

Applicability - This specification applies to the surveillance requirements for radioactive effluents which may leave the facility through the confinement system.

Objective - The objective is to assure requirements set forth in 10 CFR 20.110(d) and 10 CFR 20.1301 are not exceeded and public safety is maintained.

Specification –

a. Dosimetry shall be placed at the following locations

   1. The location inside the reactor room which represents the hypothetical minimum distance a member of the public could reach to the reactor pool.

   2. At the exhaust location of the reactor facility which is representative of effluent release from the reactor facility.

b. Dosimetry shall be changed out according to the guidance of the Purdue Radiological Management on the same time period as facility personnel or semiannually, not to exceed 7 ½ months, whichever is lesser.

Bases - Specification 4.7 will ensure that the dose given to member of the public is measured to be below those set forth in 10 CFR 20.110(d) and 10 CFR 20.1301.

5. DESIGN FEATURES

5.1 Site Description

Applicability – This specification applies to the general design and areas under which the PUR-1 Technical Specifications shall have jurisdiction.

Objective – This section is to clarify those areas which are involved with the PUR-1 Facility.

Specifications –

a. The reactor shall be located on the ground floor of the Duncan Annex of the Electrical Engineering Building, Purdue University, West Lafayette, Indiana.

b. The School of Nuclear Engineering shall control approximately 5000 square feet of the Duncan Annex ground floor, which includes the reactor room. Access to the Nuclear Engineering controlled area shall be restricted except when classes are held there.

c. The licensed areas shall include the reactor room, and a fuel storage room. Both of these areas shall be restricted to authorized personnel, or those escorted by authorized personnel.

d. The reactor room shall remain locked at all times except for the entry or exit of authorized personnel or those escorted by authorized personnel, equipment, or materials.

e. The PUR-1 reactor room shall be a closed room designed to restrict leakage.

f. The minimum free volume of the reactor room shall be approximately 15,000 cubic feet.

g. The ventilation system shall be designed to exhaust air or other gases from the reactor room through an exhaust vent at a minimum of 50 feet above the ground.

h. Openings into the reactor room shall consist of no more than the following:

   1. Three personnel doors
   2. One door to a storage room with no outside access.
   3. Air intake
   4. Air exhaust
   5. Sewer vent

Bases

The bases for the above specifications are the naming of the buildings, city and state at the time of the enactment of this amendment to the PUR-1 Technical Specifications. The access to the restricted areas is controlled to inhibit the removal of materials, information, or other import aspects of the facility to maintain confidence in safe operation under which the Safety Analysis was completed.

The volume of the reactor room and its leakage properties are so set forth to further ensure safety to facility workers and the general public is maintained during all operational circumstances.

5.2 Reactor Coolant System

Applicability – This specification outlines the make-up and properties of the PUR-1 Reactor Coolant System.

Objective – By outlining the systems which are required to be in place during operations, the validity of the Safety Analysis Report calculations is ensured.

Specifications -

a. Primary Cooling System – The PUR-1 primary cooling system shall be a pool containing approximately 6,400 gallons of water.

b. Process Water System – The process water system shall be assembled in one unit and contain a pump, filter, demineralizer, valves, flow meters, and a heat exchanger (see 5.2.d). The demineralizer shall contain a removable cartridge that is monitored continuously for radioactivity buildup.

c. Primary Coolant Makeup Water System – Makeup water for the pool shall be taken batchwise from the Purdue University water line and passed through the demineralizer enroute to the pool. A vacuum breaker shall exclude any possibility of siphoning pool water into the supply line. The pool makeup water system, in addition to the demineralizer, also shall include a normally closed manual shutoff and throttle valve and a check valve.

d. Primary Coolant Chiller System – The chiller shall be designed with three loops. Pool water shall pass through the primary loop, a Freon refrigerant in the secondary loop, and water from the building water supply shall be used to remove heat, which shall then be discharged to the building sewer system. The heat-removal capacity of the heat exchanger shall be 10.5 kW or greater.

Bases – The basis of having a reactor pool with the listed volume is to ensure there is an adequate cooling path for the PUR-1 core as well as providing a shield to direct shine from the reactor’s standard operation. The make-up water to the pool has a set of processing and monitoring equipment to ensure the long-term operation of the facility and fuel integrity by suppressing corrosion and other effects due to submersion in water. This system shall limit, by the use of filters and ion-exchange resin, the aluminum corrosion rate, corrosion product buildup, and neutron activation of impurities in the coolant.

The chiller system must maintain the reactor pool temperature to be lower than the specified value while operating to ensure the margin to the onset of nucleate boiling does not go beyond the values determined in the PUR-1 Safety Analysis Report. It shall be capable of maintain the reactor pool temperature at or below $30°C$ during steady state operation at 10 kW.

5.3 Reactor Core and Fuel

Applicability – This specification outlines the limits on the design and loading of the PUR-1 Core.

Objective – The standard loading, fuel type, and inspection period is given to ensure the shutdown margin, accident analysis, and operational characteristics are maintained and remain valid.

Specifications -

a. The fuel assemblies shall be MTR type consisting of U_3Si_2-Al, 6061 Aluminum clad plates enriched up to 20% in the U-235 isotope.

b. A standard fuel assembly shall consist of up to 14 fuel plates containing a maximum of 180 grams of U-235.

c. A control fuel assembly shall consist of up to 8 fuel plates containing a maximum of 103 grams of U-235.

d. Partially loaded fuel assemblies in which some of the fuel plates are replaced by aluminum plates containing no uranium may be used.

e. The core configuration shall consist of 13 standard fuel assemblies as described in b, and 3 control fuel assemblies as described in c.

f. The core shall include two shim-safety rods and one regulating rod placed within a control assembly. The two shim-safeties shall be made of solid borated 304 stainless steel. The Regulating Rod shall be stainless steel in composition. Each control blade shall be protected by an aluminum guide plate on each side within the control fuel assemblies.

Bases – The basis of enriching the MTR fuel up to 20% is to allow a core loading compact enough to fit within existing structures as well as to continue historic operations with approximately the same reactor characteristics while keeping the strategic significance of the material as low as possible.

Limiting the amount of grams of U-235 in each plate and assembly keeps the expected shutdown margin and accident analyses valid.

Those types of fuel plate defects listed in the specification have been exhibited in other facilities but the inspector should be cognizant of any change in plate appearance. Changes in cladding appearance may be indicative of larger issues within the core and be precursors to failure of cladding integrity.

5.4 **Fuel Storage**

Applicability – The specification for fuel storage shall apply to the placement of fuel when it is not in the core configuration.

Objective – Ensuring that fuel outside of the highly analyzed reactor core does not go critical is desirable to maintain safety to facility workers and members of the public.

Specifications -

a. All reactor fuel and fueled devices shall be stored in a geometric array where k_{eff} is less than 0.8 for all conditions of moderation and reflection.

b. Irradiated fuel assemblies and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel integrity is maintained per the Safety Analysis Report.

Bases - The requirement to store fuel in such a way that the k_{eff} is less than 0.8 will be adequate to provide reasonable certainly that an accidental criticality event is not possible.

Placing fuel in an array which allows for adequate cooling will ensure that those elements which have experienced high burnup and have elevated levels of decay heat do not undergo loss of cladding integrity by blistering or other means due to high temperature.

6. ADMINISTRATIVE CONTROLS

6.1 Organization

The PUR-1 Facility is managed and run by members of the university's College of Engineering, specifically the School of Nuclear Engineering. The Dean of the College of Engineering shall be the final authority on all PUR-1 matters. The Laboratory Director is responsible to the Dean for the administration and proper and safe operation of the facility. Figure 6.1 shows the administration chart for the PUR-1. The Committee on Reactor Operations advises the director of the PUR-1 on all matters or policy pertaining to safety. The Radiological Safety Officer provides advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection.

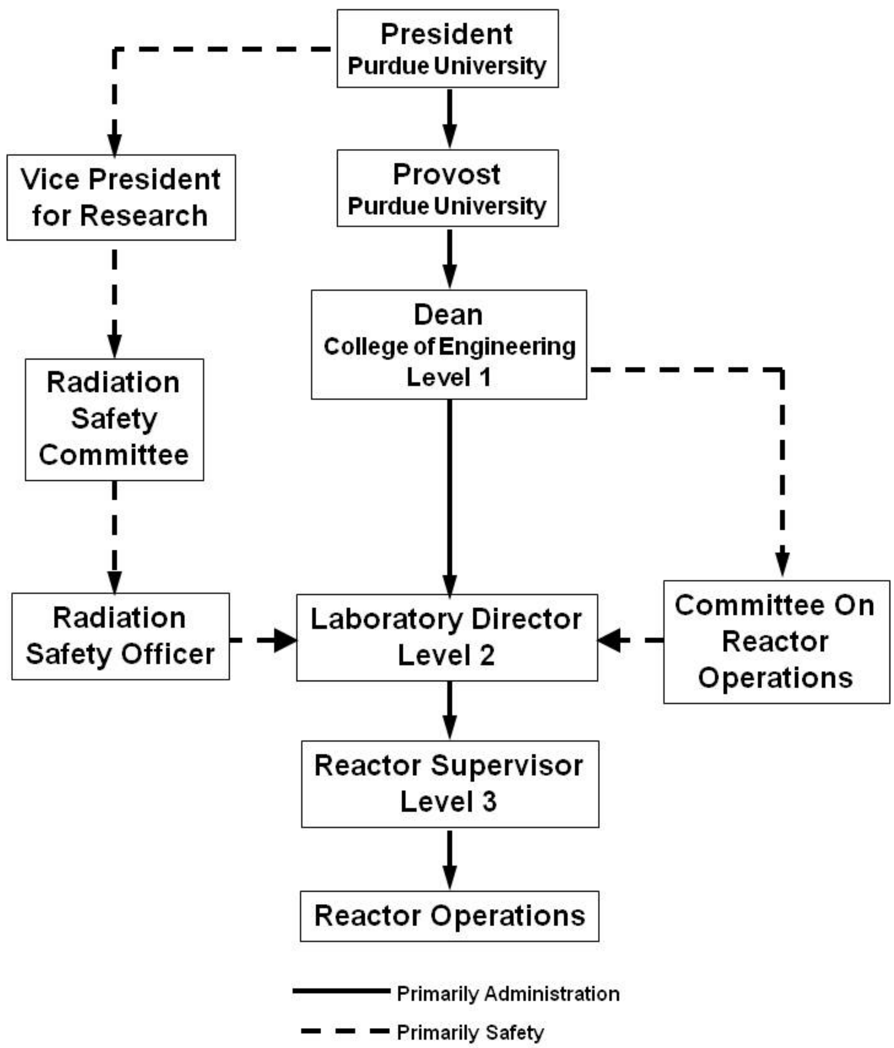

Figure 6.1: Organization Chart for Reactor Administration

a. Structure

1. A line management organizational structure provides for personnel who shall administrate and operate the reactor facility.

2. The Dean and the Facility Director shall have line management responsibility for adhering to the PUR-1 license and Technical Specifications and for safeguarding the public and facility personnel from undue radiation exposure.

3. Management Levels:

   a) Level 1: Dean of the College of Engineering: Responsible for the PUR-1.

   b) Level 2: PUR-1 Facility Director: Responsible for reactor facility operation and shall report to Level 1.

   c) Level 3: Reactor Supervisor: Responsible for the day-to-day operation of the PUR-1 including shift operation and shall report to Level 2.

   d) Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3.

   e) The reporting structure of Figure 6.1 is such that those personnel below shall report up and those personnel listed above may communicate down.

4. Committee on Reactor Operations (CORO):

   The CORO shall be responsible to the licensee for providing an independent review and audit of the safety aspects of the PUR-1.

b. Responsibility

Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established in Section 6.1.a. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

The reactor facility shall be under the direct control of the Reactor Supervisor, a Senior Reactor Operator, or Reactor Operator (RO). The RO shall be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures and requirements of the Radiation Safety Officer and the CORO.

c. Staffing

1. The minimum staffing when the reactor is not secured shall be as follows:

   a) At least two individuals shall be present at the facility complex and shall consist of at least a licensed reactor operator and a second person capable of calling 911. Unexpected absence for as long as 2 hours to accommodate a personal emergency are acceptable provided immediate action is taken to obtain a replacement. During periods when the reactor is not secured, it shall be under the direct control the of the reactor operator;

   b) During periods of reactor maintenance the two individuals who shall be present at the facility complex shall consist of a licensed senior reactor operator and a second individual capable of calling 911.

   c) A licensed reactor operator or senior reactor operator shall be in the reactor room;

   d) A Senior Reactor Operator shall be readily available for emergencies or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility); and

   e) A list of reactor facility personnel by name and telephone number shall be readily available for use in the reactor room. The list shall include:

      i. Senior Reactor Operator on Call,
      ii. Radiation Safety Officer
      iii. Other operations personnel, as deemed by the Facility Director

2. Events requiring the presence at the facility of the senior reactor operator:

   a) Initial startup and approach to power,

   b) A Senior Reactor Operator shall direct any loading or unloading of fuel or control rods within the reactor core region,

   c) A senior reactor operator shall direct the recovery from an unplanned shutdown, unscheduled shutdown, or unplanned power reduction of more than 5%.

d. Selection and Training of Personnel

The selection and training of operations personnel shall be in accordance with the following:

1. Responsibility: The Reactor Supervisor is responsible for the selection, training, and requalification of the facility reactor operators and senior reactor operators.

2. Selection: The selection of operations personnel shall be consistent with the standards related to selection in ANSI/ANS-15.4-2007

3. Training Program: The Training Program shall be consistent with the standards related to training in ANSI/ANS-15.4-2007.

4. Requalification Program: The Requalification Program shall be consistent with the standards related to requalification in ANSI/ANS-15.4-2007.

6.2 **Review and Audit**

a. Committee on Reactor Operations (CORO)

   The CORO shall be comprised of at least 3 voting members knowledgeable in fields which relate to Nuclear Safety. One of these members, the Radiation Safety Officer, will serve as the Chair. If the Chair is unable to attend one or a number of committee meetings, then the Chair may designate a committee member as Chair *pro tem*. The members are appointed by the Dean of the College of Engineering to serve three year terms. It is expected that the members will be reappointed each term as long as they are willing to serve so that their experience and familiarity with the past history of the PUR-1 will not be lost to the committee.

b. CORO Charter and Rules

   The operations of the CORO shall be in accordance with a written charter, including provisions for:

   1. Meeting Frequency: The CORO shall meet annually at intervals not to exceed 15 months. (Note: The facility license requires a meeting at least once per year and as frequently as circumstances warrant consistent with effective monitoring of facility activities);

   2. Quorum: A quorum shall be comprised of not less than one-half of the voting membership where the operating staff does not constitute a majority;

   3. Voting Rules: On matters requiring a vote, if only a quorum is present a unanimous vote of the quorum shall be required; otherwise a majority vote shall be required;

   4. Subcommittees: The Chair may appoint subcommittees comprised of members of the CORO to perform certain tasks. Subcommittees or members of the CORO may be authorized to act for the committee; and

   5. Meeting Minutes: The Chair shall designate one individual to act as recording secretary. It shall be the responsibility of the secretary to prepare the minutes which shall be distributed to the CORO, including the Dean of the College of

Engineering, within three months. The CORO shall review and approve the minutes of the previous meetings. A complete file of the meeting minutes shall be maintained by the Chair of the CORO and by the Facility Director.

c. CORO Review Function

The review responsibilities of the CORO or a designated subcommittee shall include, but are not limited to the following:

1. Review and evaluation of determinations of whether new tests or experiments and proposed changes to equipment, systems, or procedures can be made under 10 CFR 50.59 or would require a change in Technical Specifications or license conditions;

2. Review of new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems which have significant safety impact to reactor operations;

3. Review of new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

4. Review of proposed changes to the Technical Specifications and U.S. NRC issued license;

5. Review of the PUR-1 radiation protection program;

6. Review of violations of Technical Specifications, U.S. NRC issued license, and violations of internal procedures or instructions having safety significance;

7. Review of operating abnormalities having safety significance;

8. Review of reportable occurrences listed in Section 6.6.a and 6.6.b of these Technical Specifications; and

9. Review of audit reports.

d. CORO Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. Audits shall include but are not limited to the following:

1. Facility operations, including radiation protection, for conformance to the Technical Specifications, applicable license conditions, and standard

operating procedures: at least every 12 months (interval between audits not to exceed 15 months);

2. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment systems, structures, or methods of operations that affect reactor safety: at least once every 12 months (interval between audits not to exceed 15 months);

3. The retraining and requalification program for the operating staff: at least once every other calendar year (interval between audits not to exceed 30 months);

4. The reactor facility emergency plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months); and

5. The reactor facility security plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months).

   Deficiencies uncovered that affect reactor safety shall immediately be reported to the Dean of the College of Engineering (Level 1 Management). A written report of the findings of the audit shall be submitted to the Dean of the College of Engineering (Level 1 Management) and the review and audit group members within 3 months after the audit has been completed.

e. Audit of ALARA Program

The Chair of the CORO or designated alternate (excluding anyone whose normal job function is within the operating staff) shall conduct an audit of the reactor facility ALARA program annually. The auditor shall transmit the results of the audit to the CORO at the next scheduled meeting for its review and approval.

6.3 **Radiation Safety**

The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the PUR-1. The requirements of the radiation safety program are established in 10 CFR 20. The Program should use the guidelines of the ANSI/ANS-15.11-1993; R2004, “Radiation Protection at Research Reactor Facilities.”

6.4 **Procedures**

Written operating procedures shall be prepared, reviewed, and approved before initiating any of the activities listed in this section. The procedures shall be reviewed and approved by the Facility Director, the CORO, and shall be documented in a timely manner. Procedures shall be adequate to ensure the safe operation of the reactor but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be used for the following items:

a. Startup, operation, and shutdown of the reactor;

b. Fuel loading, unloading, and movement within the reactor;

c. Control rod removal or replacement;

d. Routine maintenance of the control rod, drives and reactor safety and interlock systems or other routine maintenance of major components of systems that could have an effect on reactor safety;

e. Surveillance checks, calibrations, and inspections of reactor instrumentation and controls, control rod drives, area radiation monitors, facility air monitors, the central exhaust system and other systems as required by the Technical Specifications;

f. Administrative controls for operations, maintenance, and conduct of irradiations and experiments, that could affect reactor safety or core reactivity;

g. Implementation of required plans such as emergency or security plans;

h. Radiation protection program to maintain exposures and releases as low as reasonably achievable (ALARA);

i. Use, receipt, and transfer of by-product material, if appropriate; and

j. Surveillance procedures for shipping radioactive materials.

6.5 Experiment Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures.

a. All new experiments or class of experiments shall be reviewed by the CORO as required by TS 6.2.c and implementation approved in writing by the Facility Director or designated alternate.

b. Substantive changes to previously approved experiments shall be made only after review by the CORO and implementation approved in writing by the Facility Director or designated alternate.

6.6 **Required Actions**

a. Action to be Taken in the Event of a Safety Limit Violation

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. NRC;

2. An immediate notification of the occurrence shall be made to the CORO Chair and the Facility Director, and reports shall be made to the U.S. NRC in accordance with Section 6.7.b of these specifications; and

3. A report shall be prepared which shall include:

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

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

   c) Corrective action to be taken to prevent recurrence.

   This report shall be submitted to the CORO for review and then submitted to the U.S. NRC when authorization is sought to resume operation of the reactor.

b. Action to be Taken in the Event of a Reportable Occurrence Other Than A Safety Limit Violation

1. PUR-1 staff shall return the reactor to normal operating via the approved PUR-1 procedure or shut down conditions. 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 a designated alternate;

2. The Facility Director or designated alternate shall be notified and corrective action taken with respect to the operations involved;

3. The Facility Director or designated alternate shall notify the CORO Chair who shall arrange for a review by the CORO;

4. A report shall be made to the CORO which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence; and

5. A report shall be made to the U.S. NRC in accordance with Section 6.7.b of these specifications.

6.7 **Reports**

a. Annual Operating Report

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the NRC before March 31 of each year providing the following information:

1. A narrative summary of (1) reactor operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;
2. Tabulation of the energy output of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;
3. The number of unscheduled shutdowns and inadvertent scrams, including, where applicable corrective action to preclude recurrence;
4. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;
5. A brief description, including a summary of the safety evaluations of changes in the facility or in procedures and of tests and experiments carried out pursuant to Section 50.59 of 10 CFR Part 50;
6. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient:

   a) Liquid Waste (summarized on a monthly basis)

      i. Radioactivity discharged during the reporting period.

         I. Total radioactivity released (in Curies),

         II. The effluent concentration used and the isotopic composition if greater than 1 x 10^{-7} μCi/cc for fission and activation products,

III. Total radioactivity (in Curies), released by nuclide during the reporting period based on representative isotopic analysis, and

IV. Average concentration at point of release (in μCi/cc) during the reporting period.

ii. Total volume (in gallons) of effluent water (including dilution) during periods of release.

b) Airborne Waste (summarized on a monthly basis)

i. Radioactivity discharged during the reporting period (in Curies) for:

I. ^{41}Ar, and

II. Particulates with half-lives greater than eight days.

c) Solid Waste

i. The total amount of solid waste transferred (in cubic feet),

ii. The total activity involved (in Curies), and

iii. The dates of shipment and disposition (if shipped off site).

7. A summary of radiation exposures received by facility personnel and visitors, including dates and time where such exposures are greater than 25% of that allowed or recommended; and

8. A description and summary of any environmental surveys performed outside the facility.

b. Special Reports

In addition to the requirements of applicable regulations, reports shall be made to the NRC Document Control Desk and special telephone reports of events should be made to the Operations Center as follows:

1. There shall be a report not later than the following working day by telephone and confirmed in writing by fax or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report that describes the circumstances of the event and sent within 14 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555, of any of the following:

a) Violation of safety limit (see TS 6.6.a);

b) Any release of radioactivity from the site above allowed limits; and

c) Any of the following:

i. Operation with actual safety system settings for required systems less conservative than the limiting safety system settings specified in the technical specifications.

ii. Operation in violation of limiting conditions for operation established in the technical specifications.

iii. A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required.

Note: Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra components or systems is not considered reportable provided that the minimum numbers of components or systems specified or required perform their intended reactor safety function.

iv. An unanticipated or uncontrolled change in reactivity greater than $0.006\ \Delta k/k$.

v. Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks).

vi. 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.

2. A written report within 30 days to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC, 20555, of:

a) Permanent changes in the facility organization involving Level 1 and Level 2; and

b) Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.8 **Records**

Records of facility operations in the form of logs, data sheets, or other suitable forms shall be retained for the period indicated as follows:

a. Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less Than Five Years

    1. Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year),
    2. Principal maintenance operations,
    3. Reportable occurrences,
    4. Surveillance activities required by the Technical Specifications,
    5. Reactor facility radiation and contamination surveys where required by applicable regulations,
    6. Experiments performed with the reactor,
    7. Fuel inventories, receipts, and shipments,
    8. Approved changes in operating procedures, and
    9. Records of meeting and audit reports of the CORO.

b. Records to be Retained for at Least One Certification Cycle

    Records of retraining and requalification of licensed operations personnel shall be maintained at all times the individual is employed or until the license is renewed.

c. Records to be Retained for the Lifetime of the Reactor Facility

    1. Gaseous and liquid radioactive effluents released to the environs,
    2. Radiation exposure for all personnel monitored,
    3. Drawings of the reactor facility, and
    4. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.