6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

PUR-1 is located in a high-bay room that serves the function of confinement in case of emergency. The room is kept at negative air pressure (minimum of -0.05 inches of water) at all times when the reactor is operating, and air intake and outlet are through HEPA filters. The ventilation system can be shut down during emergency situations to restrict release of contamination to the surrounding environment.

The LEU MTR fuel type serves as containment for fission products produced in the fuel during reactor operation. This fuel type is widely used, and has shown excellent fission product retention under operating conditions much harsher than what is experienced at PUR-1, both in burnup and water quality. In the unlikely event that radioactive fission products were to leak from the fuel plates, they would be contained in the reactor pool water, which is monitored for contamination regularly, and also filtered through the water process system.

6.2 Detailed Descriptions

6.2.1 Confinement

The Duncan Annex of the Electrical Engineering Building is of brick, concrete block and reinforced concreted construction which was originally designed as a large high voltage laboratory. It was subsequently subdivided into offices, classrooms and laboratories. The reactor is located in the southwest corner on the ground floor in a high bay area of the building. Figure 3.1 shows the floor plan of the Nuclear Engineering laboratories, including the reactor room.

The outside air supply and exhaust are both passed through HEPA filters. The reactor room is maintained at negative air pressure (minimum -0.05 inches of water). All doors to the reactor room have foam rubber seals. Steam heat is used to heat the room and a room air conditioner circulates and cools the reactor room air.

The only floor drain to the sewers is sealed except for a vent opening. This vent is raised about two feet above the floor and has a filtered inverted opening. Condensate from the air conditioner is released to this drain through an opening 12.0 feet above the floor.

During emergency conditions the exhaust systems is shut off and the sealed room will prevent the rapid spread of contamination. During an emergency the air conditioner and the valve on the drain from the condensate holdup tank are shut off with the same switch that shuts off the exhaust system. The condensate is held until it is tested by Radiological Control before it is released to the sewer. If contamination is found, it is disposed of as radioactive liquid waste.

6.2.2 Containment

The PUR-1 reactor does not require containment.

6.2.3 Emergency Core Cooling System

PUR-1 is a low power pool-type reactor, and in no credible accident scenarios would an Emergency Core Cooling System be necessary.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The PUR-1 instrumentation is summarized in Table 7-1. Is consists of 3 operating channels, and 1 safety channel. The three operating channels are the startup range channel, a log power channel, and a linear power channel. The safety channel has its own detector, as well as monitoring the log power channel for period.

Table 7-1: Summary of PUR-1 Instrumentation.

Instrumentation	Detail
Startup Channel	
Detector	Fission chamber
Range	Source level to 10^{-6} N_f
Indicators	Log count rate meter
Range	Decade Scalar
Log-N Period Channel	
Detector	CIC
Range	10^4 to 10^{10} n/cm^2 -sec
Indicators	Log-N meter
Range	10^{-4} to 300 N_f
Indicators	Period Meter
Range	-30 to +3 sec
Power Channel	
Detector	BF_3 Ion Chamber
Range	10^4 to 10^{10} n/cm^2-sec
Indicator	Linear Level Meter
Range	0 to 100 N_f

Safety Channels	
Detector (period)	CIC
Range	10^4 to 10^{10} n/cm^2 sec
Detector (level)	BF_3 Ion Chamber
Range	10^{-3} to 150 N_f
Rod Position Indicators	
Regulating	10-turn precision potentiometer
Range	0-2500 ohms
Shim Safety	10-turn precision potentiometer
Range	0-2500 ohms
Indicators	Voltmeters
Coarse Range	0-70 cm
Fine Range	0-70 cm Digital Voltmeter
Area Monitors (3)	
Detectors	Scintillation Detectors
Range	0.05 to 50 mr/hr
Indicators	Local & Remote Meters

7.2 Design of Instrumentation and Control Systems

The design of the instrumentation and control system for the Purdue University reactor is based on systems in use at such reactors as the Bulk Shielding Reactor, the Tower Shielding Reactor, and the Pennsylvania State University Reactor.

The instrumentation system consists of three operational channels and one safety channel. The operational channels include a counting rate channel, a log-N period channel, and a linear level channel. The log-N period channel sends a signal to a composite safety amplifier and the sigma bus to provide a fast period scram. Scram systems are interconnected into these channels to

effect reactor shutdown in the event of an emergency or abnormal condition. An annunciator and alarm system are included to indicate specific trouble.

7.2.1 Channel 1—Start-up Channel

The startup channel is used to monitor the neutron flux. The channel consists of a fission chamber, a preamplifier, a pulse amplifier, a scaler for accurate counting, a log count rate and period amplifier, a log count rate recorder, and shares a period recorder with Channel #2. The range of this equipment if from 10 to 10^4 counts/second with periods from -30 to +3 seconds. In addition to the outputs shown on the recorders, readout is also provided by a log count rate meter and a period meter shared with Channel #2 on the console and instrument panel. The complete reactor power range may be monitored by this instrument by appropriate repositioning of the detector by means of the fission chamber drive mechanism. The fission chamber may be raised into a cadmium shield by means of a drive mechanism similar to the control rod drive units. The controls and position indication for this drive are located on the console. Two set points, specified in the Technical Specification and based on the reactor period, provide for a reactor setback and trip in the event of a short reactor period.

The first activity in the reactor is indicated by means of the counting rate channel. At start-up, the fission chamber is placed near the reactor core (lower limit). Neutron-produced pulses from the fission chamber are amplified and counted on the scaler, an electronic high-speed counting device with a mechanical register. (Smaller pulses produced by any agency other than neutrons, such as gamma radiation, are rejected by a pulse height discriminator circuit.) These amplified pulses are also fed into the log count rate meter where they are integrated and the average counting rate displayed on a logarithmic indication meter calibrated for four decades (1 to 10,000 counts per second).

The average counting rate is also displayed on a remote indicating meter and recorded on a 10 millivolt Speedomax recorder provided with four-cycle logarithmic paper. When the control rod has been pulled out far enough to produce an effective reproduction constant greater than one (1.0), the reactor power level increases exponentially (at low power levels) and a straight line is drawn on the logarithmic recorder chart. The slope of this line is the pile period, or the time required (in seconds) for the power level to change by a factor of e (approximately 2.718).

The reactor period is displayed on a pile period meter located on the log count rate meter and is recorded on the period recorder until the lower limit of the log-N period channel range is reached.

When the counting rate channel is near the limit of its counting range, the log-N period and linear servo channels become the principal means of controlling the reactor, and the fission chamber is withdrawn to a region of lower neutron flux (to keep the recorder on scale).

7.2.2 Channel 2—Log N and Period Channel

The log N channel indicates the reactor power level over the range from 0.0001 to 300 percent power level. The detector for this channel is a compensated ionization chamber followed by a log N amplifier plus period instrumentation with outputs to the log N recorder and to the period instrumentation with outputs to the log N recorder and to the period recorder shared with Channel #1. Indication, in addition to the recorders, is provided by log N and period meters on both console and instrument rack, with the console period meter shared with Channel #1. This

channel is not 'on scale' at startup, but will be indicating before the range of the fission is exceeded.

A reactor trip will be initiated if this channel indicates power levels in excess of 120% of the licensed power. Two set points, specified in the Technical Specifications and based on the period, provide for a reactor setback or trip in the event of a short reactor period. In the event of the loss of high voltage to the compensated ion chamber, a trip will be initiated.

7.2.3 Channel 3—Linear Power

The linear level channel is capable of measuring neutron flux in a reactor operating range from shutdown to > 100 kilowatt. The sensing element is BF_3 ionization chamber coupled to a micro-microammeter. The range of the instrument is adjustable by means of a range switch located on the instrument (instrument panel) from $0–10.0\times10^{-12}$ to $0–10.0\times10^{-4}$ amperes. Detector characteristics, however, limit its maximum output to 10^{-4} amperes. This channel will thus read from startup to full power by adjustment of the range switch. The output is recorded on the linear power recorder on the instrument panel and indicated on meters on the console and on the instrument panel.

This channel has two set points that will initiate a reactor set back at either zero or 100% range. These set points insure that the instrument is kept on range at all times during reactor operation. This is also a 120% range set point that will initiate a reactor trip.

The linear servo channel consists of a BF3 ionization chamber (whose output current is proportional to neutron flux) and its power supply, a micro-microampere amplifier, a Speedomax recording controller with a position-adjusting type control unit, and the regulating rod drive unit. Because this channel is the most precise channel in measuring neutron flux level, it is used as the means to obtain uniform neutron flux density.

The BF_3 ionization chamber provides a DC current which is proportional to the neutron flux at the chamber. This current is amplified by the micro-microampere amplifier which provides a 0-10 millivolt input signal to the recording controller. The controller provides a signal to the servo control unit which, through a drive unit, drives the control rod the necessary amount in the correct direction to maintain the power level at the control point set on the controller. Whenever the power level exceeds the control point setting the control rod is inserted into the reactor the amount necessary to decrease the reactivity and restore the power level to the control point setting. Conversely, whenever the power level drops below the control point setting, the control rod is withdrawn the amount necessary to increase the reactivity and restore the power level to the control point setting.

In practice, the operator sets the red control pointer to the desired power level and selects the range on the micro-microampere amplifier which will provide a reading on the linear recorder equivalent to the power level reading desired on the log-N recorder. The rods are then withdrawn until the black indicating pointer coincides with the red control pointer at which time the power is leveled off manually and the control unit is placed on automatic operation.

7.2.4 Log-N Period Channel

The log-N period channel includes a compensated ionization chamber (whose output current is proportional to neutron flux only) and its power supply; a log-N amplifier and a Speedomax

recorder which records log-N; and a composite safety amplifier whose components are a period safety preamplifier, sigma amplifier and magnet amplifier.

The compensated ionization chamber supplies a d-c current (proportional to the neutron flux at the chamber) which is amplified by the log-N amplifier. One output of the log-N amplifier is the logarithm of the power level which is indicated on the power level meter and recorded on the log-N recorder. This output is also differentiated and indicated on the pile period meter. (The pile period is the number of seconds necessary for the power level to increase or decrease by a factor of e.) Positive periods from infinity to three seconds, and negative periods from infinity to thirty seconds are indicated.

7.2.5 Channel 4—Safety Channel

This channel utilized a BF_3 ion chamber and feeds directly into the safety amplifiers. The sensitive range of this instrument is from a few percent to at least 150 percent of power, linearly. Its output is indicated on the instrument chassis (instrument panel). The purpose of this channel is solely to provide a trip at the measure value as specified in the Technical Specifications.

The level safety channel, in conjunction with the safety circuit of the log-N period channel, functions to shut down the reactor immediately by dropping both safety rods whenever the power level increases above 120% N_f and/or an abnormally short period occurs. (Set for a period of 7 seconds.)

In the level safety channel a BF_3 ionization chamber supplies a d-c current proportional to the reactor power level to the composite safety amplifier. Since the current produced by neutron flux at high levels of operation is much greater than that produced by gamma radiation or other sources, this chamber is not compensated. Each magnet amplifier supplies current to an electromagnet which holds a safety rod. The output of each sigma amplifier is connected to the sigma bus which is connected to the input of each magnet amplifier. When there is no signal from the ionization chamber, the sigma amplifiers maintain the potential of the sigma bus at 37 volts. However, if the positive period should become abnormally short (7 seconds) or if the neutron flux (power) level should become dangerously high the sigma bus potential is increased which causes the magnet current in both magnet amplifiers to be quickly reduced to zero, thus dropping both safety rods into the reactor core. This is what is meant by a "fast scram."

The same result is also achieved if the sigma bus potential should increase for any other reasons, since the sigma bus is connected to the input of each magnet amplifier. (Note that a fast scram may be initiated automatically by a short period or a high power level.) Indicator lamps in the annunciator system located on the control console indicate whether a fast scram is due to high level or to period.

7.3 Reactor Control System

Four control channels are incorporated into the reactor control system to raise or lower the two shim-safety rods, the regulating rod and the fission chamber. A fifth control channel is provided to raise and lower the source. The control rods may always be lowered with a jam circuit being the only restriction; however, permissive circuits are included in the raise circuits of the control rods and fission chamber to prevent raising the rods under the following circumstances:

Indicator	Condition
Source Missing	The log count rate recorder must indicate the presence of a source of neutrons by indicating a count rate of at least two counts per second. A source missing indicator shows when this condition is not fulfilled.
Log Count Rate Recorder On	A relay in the log count rate recorder prevents raising of the control rods or the fission chamber when the log count rate recorder power is turned off, thus prohibiting the disabling of the source missing circuit. The log count rate recorder may be turned off after the log-N recorder comes on scale. This is accomplished by a relay in the log-N recorder circuit.
Source Drive in Operation	Interlock circuits on the source drive switches prevent raising the control rods or fission chamber when the source is being raised or lowered.

A jam circuit is incorporated in the drive circuits to operate a jam indicator light on the console in the event of a mechanical jam in the drive. This indication alerts the operator to the possibility of cable kinking in the source and fission chamber drive units, or mechanical friction in the rod drives.

7.3.1.1. Shim Safety Rod Drive System

The shim-safety rod to be driven is selected by pushing the desired drive indicator lamp. This connects the drive system for that rod and clears any other drive circuit that may be energized. Electrical interlocks prevent the raising of more than one control rod or the fission chamber simultaneously.

The reactor operator controls the rod with the raise-lower switch on the control console. The raise-lower switch operates mercury relays which control the rod drive motors.

Indicator	Condition
Upper Limit	The drive unit is at the upper limit of its travel.
Shim	The drive unit is two-thirds out. This point is where the shim-safety rods are set during critical experiment fuel loading.

Engage	The shim-safety rod is attached to the drive electromagnet.
Lower Limit	The drive unit is at the lower limit of its travel.
Drive	The drive unit is connected to the raise-lower switch.

7.3.1.2. Regulating Rod Drive System

The regulating rod drive system is activated by pushing the reg rod drive indicator lamp. This connects the regulating rod system to the raise-lower switch and clears any other drive circuit that may be energized. Electrical interlocks prevent raising more than one control rod or the fission chamber simultaneously.

The reactor operator controls the regulating rod with the raise-lower switch on the control console. The raise-lower switch operates mercury relays which control the rod drive motors.

The regulating rod may be controlled by the servo amplifier if the reactor operator pushes the servo-permit indicator. The servo system is turned off by pushing the servo indicator.

The following indicator lights are provided for the regulating rod:

1. Upper Limit
2. Servo
3. Servo Permit
4. Lower Limit
5. Drive

7.3.1.3. Fission Chamber Drive System

The fission chamber drive system is activated by pushing the fission chamber drive indicator lamp. This connects the fission chamber drive system to the raise-lower switch and clears any other drive circuit that may be energized. Electrical interlocks prohibit raising the fission chamber while the control rods are being driven.

The reactor operator controls the fission chamber with the raise-lower switch on the control console.

The following indicator lights are provided for the fission chamber:

1. Upper Limit
2. Lower Limit
3. Drive

7.3.1.4. Source Drive System

The source is raised or lowered by pushing either the source raise or source lower indicator lamp. The source drive switches are of the momentary type and must be pushed and held on until it is desired to stop the source drive or one of the limits is reached.

The following indicator lights are provided for the source drive:

1. Upper Limit
2. Source-Raise
3. Source-Lower
4. Lower Limit

7.3.1.5. Gang Lower System

When the reactor is to be shut down, it is permissible to lower all control rods simultaneously by pushing the gang lower indicator light and thus activating the gang lower switch and relay. The gang lower relay directly energizes the lower drive relays for the control rods. The gang lower relay is de-energized by momentarily placing the raise-lower switch into either its raise or lower position. In addition, the gang lower relay may be energized by the setback system described in Section 1.5.5.2.

7.3.1.6. Rod Position Indicating System

Four coarse rod position indicators are provided which continuously monitor the positions of the two shim safety rods, the regulating rod and the fission chamber with a resolution of 2 centimeters. A ratiometer is used as a fine position indicator which may be switched to any one of the control rods or fission chamber to provide a position indication with a resolution of 0.1 millimeter. The rod position indicator measures a voltage across a potentiometer connected to the drive unit. The reference voltage for all drive unit potentiometers is supplied by a 68 volt power supply.

7.4 Reactor Protection System

Two types of scrams are included in the control system to effect shutdown of the reactor in the event of emergency conditions. A fast scram can be accomplished when a short period signal or a high power-level signal removes magnet current. The fast scram operation is described in Section 7.2.5.

A slow scram is initiated by various safety circuits located in the reactor system. The slow scram circuit opens the input power line to the magnet amplifier power supplies located in the composite safety amplifiers.

A setback system is included in the reactor control system to insert the control rods into the reactor without producing a scram. This minimizes the necessity of repeating the startup procedure.

When a scram or setback condition occurs, or when other trouble arises, a buzzer alarm will sound, and a lamp or lamps will be lighted on the control console, indicating the source of the trouble. An annunciator acknowledge button is used to turn off the buzzer. If the trouble results in a scram, a scram reset switch must be actuated before magnet power can be reapplied. In the event of trouble other than a scram, the corresponding switch-lamp is pushed to extinguish the lamp and reset the annunciator system after the trouble has been corrected. If the trouble is not corrected, the indicator light will remain lighted. An annunciator test switch is provided for checking the lamps and the buzzer. An evacuation alarm horn is installed in the reactor instrument racks. This horn is activated by pushing the alarm button on the reactor console.

7.4.1.1. Slow Scram System

Opening of a fault switch actuates a slow scram channel which causes primary power to be removed from the two magnet amplifiers located in the composite safety amplifiers, thus resulting in the shim-safety rods dropping into the core. The following conditions result in a slow scram.

Slow Scram Type	Condition
Scram - Console	Manual pushbutton on console depressed.
High Level - Console Monitor	High radiation level at the console radiation monitor.
High Level - Process System Monitor	High radiation level at the process system radiation monitor.
High Level - Rod Drive Area Monitor	High radiation level at the rod drive area radiation monitor.
CIC Power Supply Trouble	Failure of the high voltage power supply for the compensated ion chamber.
Composite Safety Amplifier Trouble	Failure of one or more of the circuits in either composite safety amplifier.

The following scram conditions are initiated by relay type meters located on the auxiliary scram panel. The scram set points are determined and set by the reactor supervisor prior to operation.

Scrams	Conditions
Low-Level Period	Period circuit in the log count rate meter indicates a short period.
High-Level Period	Period circuit in the log-N period amplifier indicates a short period.
High-Level Log-N	The log-N channel indicates a high power level.
High-Level - Linear Level	The linear level channel indicates a high power level.

The following permissive circuits are located in the slow scram circuit and must be in operation before the shim-safety rods can be raised:

1. The log-N selector switch must be in the <u>operate</u> position.
2. The log-count-rate meter switch must be in the <u>use</u> position.
3. The key switch must be on.
4. The scram reset button must be pushed and the magnet power lamp energized.

7.4.1.2. <u>Setback System</u>

Four setback circuits are included in the reactor control system. Actuation of any one of these circuits will result in the shim-safety rods and the regulating rod being driven into the core to their lower limits unless the trouble is cleared and the circuit reset before the lower limits are reached. The setback set points are determined and set by the reactor supervisor prior to operation. The following setback conditions are initiated by relay type meters located on the auxiliary scram panel:

Setbacks	Conditions
Low Level Period	Period circuit in the log count rate meter indicates a short period.
High-Level Period	Period circuit in the log-N period amplifier indicates a short period.
High-Level - Linear Level	The linear level channel indicates a high power level.
High-Level - Safety Amplifier	The level safety amplifier indicates a high power level.

7.5 Control Console and Display Instruments

The reactor console is designed to provide maximum visibility of the instruments and accessibility to the controls and indicators. All indicators and controls necessary for startup and shutdown operations are located in one group in front of the operator.

Colors for the indicator lights on the console show the operator the status of the reactor at a glance. All trip and warning indicators are red or yellow. Operating procedure, as well as interlock, keep the operator from withdrawing the control rods when a warning indicator is showing.

7.6 Radiation Monitoring Systems

Three scintillation type area monitors are installed in the reactor area to monitor the radiation level at the top of the pool, in the pool water flowing through the process system, and at the reactor console. Each area monitor is connected into the slow scram circuit and is equipped with a local lamp and alarm that is activated if the radiation level exceeds the set point. Three remote area monitor meters are mounted on the reactor instrument racks to provide the operator with an indication of the radiation level at the area monitor sites.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The instrumentation and controls of PUR-1 use the standard building electrical power system at 120 Volts on a dedicated circuit. The water process system uses three-phase power at 240 Volts. Room lighting and HVAC are also powered by building power on their own circuits.

8.2 Emergency Electrical Power System

There are no emergency electrical power systems for any of the systems associated with PUR-1. Loss of electrical power during operation will de-energize the magnet amplifiers on the shim-safety rods, which will drop them into the core and shutdown the reactor similar to a scram, leaving the reactor in a safe condition under any credible operating or accident scenario.

Emergency lighting for the room will be activated upon loss of building power, enabling the operators to ensure the safe shutdown condition of the reactor (i.e. shim-safety rods inserted) and direct personnel in the reactor room as necessary.

9 AUXILLARY SYSTEMS

9.1 Heating, Ventilation and Air Conditioning Systems

The outside air supply and exhaust are both passed through HEPA filters. The reactor room is maintained at negative air pressure (minimum 0.05 inches of water). All doors to the reactor room have foam rubber seals. Steam heat is used to heat the room and a room air conditioner circulates and cools the reactor room air.

9.2 Fuel Storage and Handling

There are two in-pool storage racks located on the bottom of the pool opposite from the reactor, each of which could hold all of the fuel from the reactor if necessary. The configuration of the racks are two rows of 9 spaces, for a total of 18 assemblies each. These rows are separated by a one quarter inch BORAL plate to reduce the k_{eff} of the fuel stored there. There is also a dry storage facility, located within the facility, but outside of the reactor room.

An MCNP model of the in-pool storage racks was constructed. Two cases for the LEU fuel were examined. One case was run with standard LEU assemblies in all of the 18 positions (which is not possible with the anticipated LEU inventory, but was run as a limiting case), and no credit was taken for the ¼" BORAL plate between the two rows. This first case had a calculated eigenvalue of 0.7660±0.0046. The second case was modeled with the ¼" BORAL plate with a boron density of $23.8E21^{atoms}/_{cc}$, and 16 standard assemblies with 14 fuel plates in each. The eigenvalue for this calculation was determined to be 0.3319 ±0.00178. Both of these bracketing cases are below the TS 5.3 limit of 0.8, thus TS 5.3 will be met. The geometry of the model is shown in Figure 9-1 below.

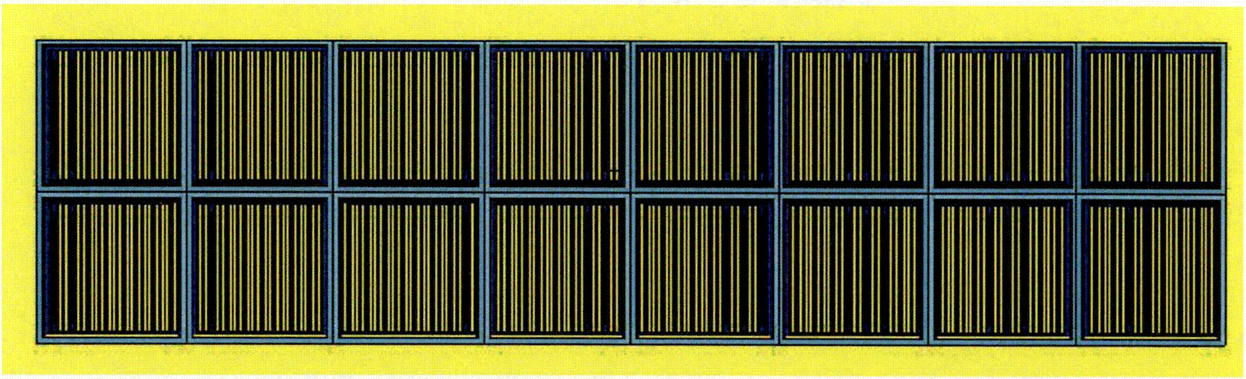

Figure 9-1: MCNP model of in-pool fuel storage rack.

The storage of the extra LEU plates is in the dry storage facility. Parametric eigenvalue calculations have been performed with MCNP5, varying the storage plate spacing in a hypothetical flooded condition to ensure that a critical configuration could not be achieved. These results are shown in Figure 9-2. The maximum k_{eff} obtained in these calculations was 0.41, which is less than the Technical Specification (TS 5.3.1) requirement of less than 0.8. The model used for the dry storage is shown in Figure 9-3.

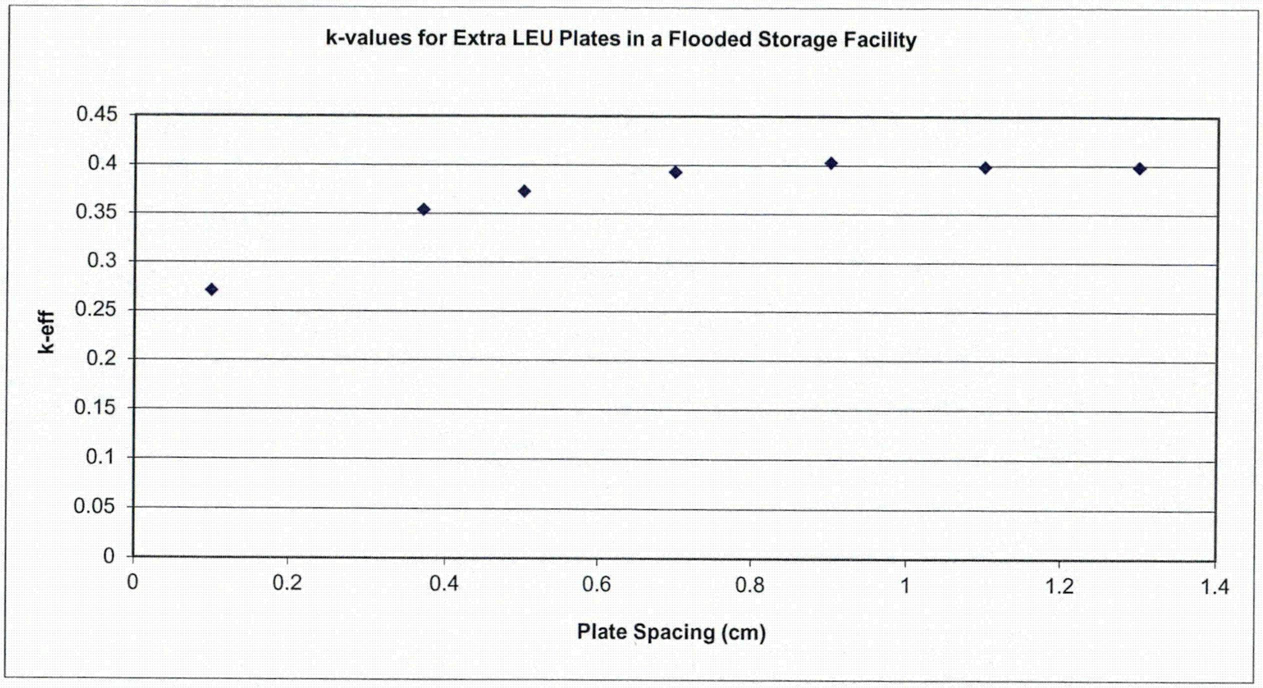

Figure 9-2: k-eff Values for Flooded Condition of Fuel Storage Facility.

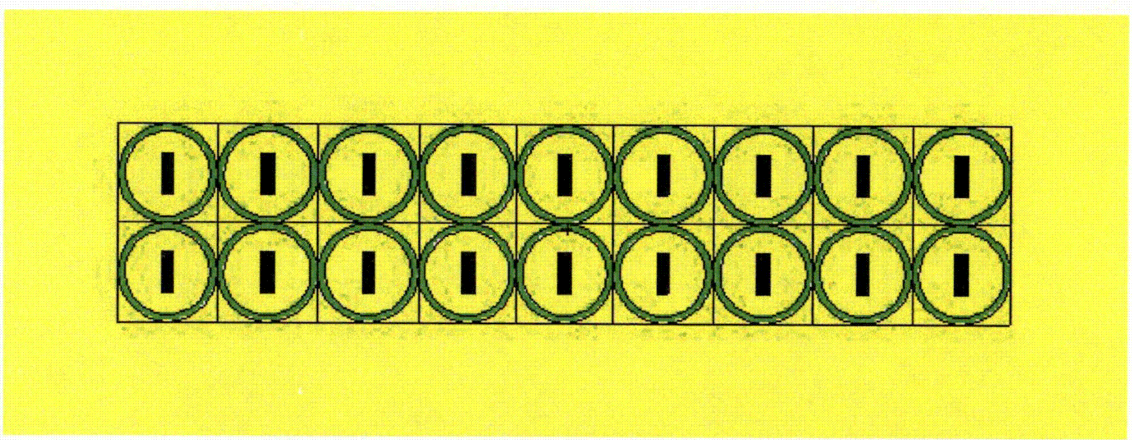

Figure 9-3: MCNP model of dry fuel storage facility.

With all of the LEU fuel plates in the inventory located in the dry storage facility, the calculated eigenvalue of the storage rack is 0.015±0.0003. Should the storage flood with water, the eigenvalue becomes 0.26±0.004. In both of these conditions, the TS limit of k_{eff}<0.8 is met. Parametric analyses were performed with spacing of the plates within the tubes, and the above listed cases are the worst case. All movement of fuel is performed under the direction of licensed senior reactor operators.

9.3 Fire Protection Systems

A fire extinguisher, maintained by the Purdue Fire Department, and appropriate for the types of fires that could be encountered in the reactor room, is located within the reactor room.

9.4 Communication Systems

A telephone is located at the console from which the operator can contact the Senior Operator on call, or any other assistance that may be necessary to support operations. Dialing 911 from the console phone will direct a call to Emergency Services.

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

Purdue University has a Broadscope license, and all byproduct material generated with the reactor falls under this license, as described in Chapter 11.

10 EXPERIMENT FACILITIES AND UTILIZATION

10.1 Summary Description

PUR-1 experimental facilities consist of experiment locations within the graphite reflector on one side of the reactor, and drop tubes located as close as next to the reflector boundary, and as far away as 30 inches. All of the drop tubes are dry air tubes, and the in-reflector facilities are aluminum tubes normally filled with graphite, which can be replaced with experiment capsules.

10.2 Experimental Facilities

The aluminum sample tubes in the irradiation facility are filled with graphite unless they are being used. Replacing the graphite in the six sample tubes with air, as would be expected if samples were inserted, reduces the calculated eigenvalue of the LEU core from 1.00731±0.0002 to 0.996±0.0002 without any reactivity bias, but with the estimated bias of 0.32%, these values would be 1.0041 and 0.993, respectively.

The effect of replacing graphite in the irradiation facility with an aluminum sample holder on the reactor power distribution used in the thermal-hydraulic analyses is well covered by the global hot channel factor of 1.5 for the reactor power measurement uncertainty used in all the reported analyses. The reactor power hot channel factor was used in addition to (1) the radial power factor (used to account for plate-to-plate power variation), (2) the factor used to account for power density variation along the width of the hot plate, and (3) the axial power profile. While replacing the graphite in the irradiation facility with an aluminum sample holder will cause a change in the reactor power distribution, this change will be small compared with the hot channel factors already applied in the thermal-hydraulic analysis. Therefore, no additional thermal-hydraulic analysis is needed.

Other experimental facilities include dry air drop tubes. Two tubes are located next to the core, one 5/8" in diameter, and the other 1.75" in diameter. A 3" PVC drop tube is located in the pool, and a 5" stainless steel drop tube is available that can be located in any grid location. An analysis of the flooding of the 5" drop tube is included in the safety analysis in Chapter 13.

10.3 Experiment Review

Review of new experiments is performed by the reactor staff, and then by the Committee on Reactor Operations, as required by the Technical Specifications.

11 RADIATION PROTECTION AND WASTE MANAGEMENT

11.1 Radiation Protection Program

Purdue University has a structured radiation safety program. Policies for the program are determined by the University Radiation Safety Committee, which has the mission to ensure the safety of the University and community in the utilization of all radioactive materials and radiation producing devices at the University by faculty, staff or students. This includes all teaching, research, and outreach programs. The program is administered by the Radiation Safety Officer (RSO) and his staff, as part of Radiological and Environmental Management (REM). The staff is equipped with radiation detection instrumentation to determine, control and document occupational radiation exposures at the reactor facility, and all laboratories using radioisotopes at the university under the By-product License 13-02812-04 (Broadscope).

Natural background radiation levels in the West Lafayette area result in an average exposure of about 100 mrem/yr. On the basis of normal reactor use, the maximum potential non-reactor room dose would be less than 1 mrem/yr, so there should be no significant contribution to the background radiation in unrestricted areas.

11.1.1 Radiation Sources

11.1.1.1. <u>Reactor</u>

Radiation from the reactor core if the primary source of radiation directly related to reactor operations. Radiation exposure rates from the reactor core are reduced to acceptable levels by the water in the pool and concrete shielding.

11.1.1.2. <u>External Sources</u>

Sources of radiation associated with reactor use include radioactive isotopes produced for research, activated components of experiments and activated samples.

11.1.2 Gaseous Effluents

The primary gaseous radionuclide considered to be produced in more than a negligible quantity by PUR-1 operations is Ar-41. This isotope is produced whenever air is in contact with a neutron radiation field. Naturally-occurring Ar-40, which comprises over 99% of all argon, undergoes a neutron capture reaction to produce Ar-41, which decays by beta and daughter product (K-41) gamma emission, with a half-life of 1.83 hours. Argon is found in air at slightly less than 1% concentration under STP conditions.

Smaller concentrations of gaseous radioisotopes will also be produced from other activation products in air, experimental procedures, and a slight possibility of very small quantities of fission product gases released into the reactor room environment from dissolved fission product gases in the pool water. However, the quantities of these other sources are very small compared to Ar-41 production.

The water in the reactor pool also contains dissolved and it is assumed the dissolved air has an argon concentration equal to that found in atmospheric air. Some of this argon will activate and be released from the surface of the reactor pool into the reactor room air.

11.1.2.1. Argon Production in Experimental Facilities

Production of Ar-41 can occur in the dry drop tubes mounted near the core, and a movable drop tube that can be located near the core but is usually stored across the pool from the core. The production of Ar-41 can be estimated by:

$$A(t) = N\sigma\phi\left(1 - e^{-\lambda t_i}\right)e^{-\lambda t_d}$$

where

N	= total number of target atoms available for activation
σ	= microscopic absorption cross-section
φ	= neutron flux in neutrons/cm^2/s
λ	= decay constant
t_i	= irradiation time, and
t_d	= decay time.

For the most conservative evaluation, it is assumed that the quantity of Ar-41 reaches saturation, i.e., the irradiation time is 10 half-lives, or about 18.3 hours (Ar-41 $T_{1/2}$=1.83 hrs). PUR-1 is generally not operated more than four consecutive hours. It is also assumed that the Ar-41 is released immediately after irradiation, such that t_d=0. Thus the production equation simplifies to:

$$A = N \varphi \sigma$$

The Chart of the Nuclides[7] gives a value for σ(Ar-40) of 0.65 barns (6.5 x 10^{-25} cm^2) for the thermal neutron absorption capture cross-section. The number of Ar-40 atoms available for activation is a function of the volume of air in the experimental facility and the concentration of argon in air under STP conditions. Etherington[8] reports a concentration of 2.5 x 10^{17} atoms of argon per cubic centimeter of air under STP conditions. Using the isotopic abundance of 0.996 for Ar-40, a concentration of 2.49 x 10^{17} atoms of Ar-40 per cubic centimeter of air at STP is estimated.

There are four different drop tubes that can be used at PUR-1, two that are fixed and set against the reflector, one moveable drop tube, and one fixed tube located about 18 inches from the core. The moveable drop tube can be placed just outside the reflector, similar to the two that are fixed there. These tubes, and their associated fluxes (estimated from measured fluxes at 1000W and extrapolated to 18 kW), are described in the table below.

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[7] Baum, E. M., Knox, H. D., Miller, T. R., Knolls Atomic Power Laboratory. & Lockheed Martin. *Chart of the nuclides : nuclides and isotopes.* 16th edn, (KAPL : Lockheed Martin, 2002).

[8] Etherington, H. *Nuclear engineering handbook.* 1st edn, (McGraw-Hill, 1958).

Table 11-1: Estimated Neutron Flux

Tube	ID (cm)	Proximity to core	Estimated Neutron Flux at 18 kW (12 kW + 50%)
Small (SS)	1.55	Against reflector	2.79×10^8
Medium Aluminum	3.11	Against reflector	2.79×10^8
Large PVC	7.52	46 cm	8.79×10^5
Large moveable SS	12.08	Can be against reflector	2.79×10^8 max

Table 11-2: Ar-41 Saturation Activity

Tube	Effective Exposed Volume (cm^3)	Atoms of Ar-41 in Volume	Neutron Flux	Saturation Ar-41 Activity in Microcuries	Saturation Ar-41 Activity Production in Microcuries
Small (SS)	114.43	2.85E+19	2.79×10^8	0.14	1.47E-05
Medium Aluminum	463.68	1.15E+20	2.79×10^8	0.57	5.95E-05
Large PVC	2709.67	6.75E+20	8.79×10^5	0.01	1.10E-06
Large moveable SS	6982.02	1.74E+21	2.79×10^8	8.52	8.97E-04

11.1.2.2. Argon Production from Pool Water

Estimation of the Ar-41 production from dissolved air in the water of the reactor pool begins with a calculation of the exposure time of water passing through the core. Table 4-21 noted that the average coolant velocity in the through the core is 1.85 cm/second, assuming a 12 kilowatt operating power and natural convection through the core. The length of the active fuel channel is 60.96 cm, which gives a coolant transit time of 33.0 seconds, assuming the maximum velocity through the core. This is taken to be the exposure time of the water to the average flux throughout the core. Based on measurements of the peak thermal neutron flux in the core region at a 1 kilowatt power level, comparing to the MCNP5 calculation, and assuming linearity of thermal flux with reactor power, the peak thermal neutron flux in the core at 12 kW is assumed to be 1.66E11 n/(cm2*s).

The volume flow rate of water through the core is the product of the coolant velocity and the total flow area. Assuming a core with 13 standard fuel elements and 3 control rod fuel elements, the total flow area is the product of the flow area of an individual coolant channel and the total number of channels in the core. A standard fuel element has 15 flow channels, and a control element has 11. The total core flow area can be calculated to be 532.56 cm^2, and the total volumetric flow rate is found to be 985.24 cm^3/s.

The average out-of-core cycle time is given by:

$$T = V_p / \dot{V}$$

where

V_p = total pool volume, and

$\dot{V}$ = volumetric flow rate through the core

Assuming a pool volume of 6400 gallons (2.42×10^7 cm^3), and out-of-core cycle time of 2.46×10^4 seconds is obtained. This can be assumed to be the decay time for Ar-41 produced in the pool water.

The concentration of argon gas in the pool water can be predicted by Henry's Law. The dissolved concentration of a gas in contact with a liquid is proportional to the partial pressure of the gas and the temperature of the liquid. Dorsey[9] reports values for air at STP conditions in water that allow an estimation of 8.65×10^{15} atoms of Ar-40 per milliliter of water, assuming a water temperature of 25 degrees C. The saturation activity of Ar-41 in the pool water may then be predicted from:

$$A(t) = N\sigma\phi\left(1 - e^{-\lambda t}\right) / \left(e^{-\lambda[t+T]}\right)$$

where

- N = concentration of Ar-40 atoms in the water
- σ = microscopic absorption cross-section
- φ = average neutron flux in neutrons/cm^2/s
- λ = decay constant for Ar-41
- t = exposure time of water in the core, and
- T = average out-of-core cycle time.

Substituting all of the appropriate values into this equation, we find an estimate of 3.5 disintegrations/second/cm^3. Dividing this value by the decay constant yields an estimated density of 3.32×10^4 atoms of Ar-41/cm^3.

As water passes through the core it is heated slightly (~5 degrees), which reduces the solubility of air in the water. For this calculation, it is assumed that 25% (very conservative) of the dissolved argon is released from the water because of core heating. Some of this released argon will be re-dissolved as it mixes with cooler water in other regions of the pool. Measurements done at other reactors allow an estimate of 50% re-dissolving fraction. Thus, the argon available for release to the building air is given by:

$$S_1 = F_1 F_2 N_{41} \dot{V}$$

where

[9] Dorsey, N. E. *Properties of ordinary water-substance in all its phases: water-vapor, water, and all the ices*. (Reinhold Publishing Corporation, 1957).

F_1 = Ar-41 concentration in the water at equilibrium,
F_2 = release fraction from heating (assumed to be 25%),
N_{41} = redissolving fraction (assumed to be 50%), and
$\dot{V}$ = volumetric flow rate through the core.

Substituting the appropriate values in this equation leads to an available release term of 4.09 x 10^6 atoms of Ar-41/second. This represents one component of Ar-41 release from the water.

Another release term arises from the tendency of dissolved gas at the surface of a liquid to escape to the air across the water-air boundary. Estimating the magnitude of this release term requires calculation of an effective exchange coefficient for argon (exchange coefficient being the amount of gas in a unit volume exchanged at the surface per unit time per unit area).

Other reactor facilities have analyzed this problem and provide possible exchange coefficients that appear to cover a wide range. For example, analyzing the gas exchange at the liquid-gas boundary in terms of the diffusion coefficient of argon gas dissolved in water and the mean-square distance traversed by a molecule, an estimate of 2.35×10^{-3} cm/second is obtained. However, measurements made of the Ar activity in the pool water of a TRIGA Mark III and subsequent analysis of these data indicate an exchange coefficient of about 2.9×10^{-4} cm/second. Further, Dorsey reports approximately equal surface exchange coefficients for gases such as air, O_2, and N_2. Assuming that the exchange properties of argon are similar to those of these gases, an exchange coefficient of about 5.7×10^{-3} cm/second is possible. Note that these estimates vary by almost a factor of 10.

In the interest of conservatism, the largest exchange coefficient (5.7×10^{-3} cm/second) is assumed in this calculation. Using this, the release rate from gas exchange at the surface of the pool is given by:

$$S_2 = 0.93B \cdot N_{41} A_S \qquad (0)$$

N_{41} = Ar-41 concentration in the pool water
B = exchange coefficient, and
A_S = surface area of the pool (4.67×10^4 cm^2)

Using this equation a release rate of 8.23×10^6 atoms/second is obtained. Now, the total source term for Ar-41 released from the pool water is obtained by adding this to the previous estimate for dissolved argon:

$$\begin{aligned} S_{41} &= S_1 + S_2 \\ &= 4.09 \times 10^6 + 8.23 \times 10^6 \\ &= 1.23 \times 10^7 \text{ atoms/s} \end{aligned} \qquad (0)$$

This is the source term for Ar-41 released from the pool water to be used later in estimating doses and isotopic concentrations. The source term assumes 12 kilowatt operation for a time sufficient to attain saturation activity.

11.1.3 Estimated releases in the Restricted Area

11.1.3.1. <u>Types of Releases</u>

Release of Ar-41 from experimental facilities can occur as either a puff from displacement of the air in a drop tube, or the surface of the pool, a continuous stream. Section 6.1.2 discussed the estimated source terms for Ar from experimental facilities and Ar from the surface of the pool, assuming 12 kilowatt operation. The following sections will analyze individual release scenarios and their consequences. These analyses concern releases made within the confines of the reactor room, which is defined as a restricted area.

11.1.3.2. <u>Puff Release from the Drop Tubes</u>

The Ar-41 activities in the drop tubes presented in Tables 11-2 are very conservative, and it is not realistic for those to be achieved. Any theoretical release from those tubes might arise from displacement of the air in the tube by the insertion or removal of a sample with a diameter close to the inner diameter of the tube itself. Looking at the largest tube, a saturation activity of 8.52 microcuries is available for release. One could even consider the dispersal of the air containing Ar-41 from all of the tubes at once. The sum of the activities in all of the tubes in Table 11-2 is only 9.24 microcuries. If we assume instantaneous and ideal mixing of that activity to the reactor room air (4.24×10^{8} cm^3), we achieve a concentration of 2.18×10^{-8} microcuries/cc of room air. Per 10 CFR 20, the derived air concentration limit for Ar-41 is 3×10^{-6} microcuries/cc. Thus, the Ar-41 concentration from dispersal of the air within the drop tubes to the reactor room is two orders of magnitude lower than the allowable limit. Furthermore, the exhaust fan will reduce that concentration quickly, changing the room air out in under two hours. Therefore we can conclude that Ar-41 release from the drop tubes will not endanger those working within the reactor room.

11.1.3.3. <u>Continuous Release from the Pool Water</u>

The release rate of Ar-41 from the pool water was estimated at of 1.18×10^{7} atoms/s. Multiplying this release rate by the decay constant for Ar-41 (1.05×10^{-4}/s), and converting this activity to microcuries yields a release estimate of 0.035 microcuries/second from the pool. Expressing the concentration buildup of the isotope in the air of the Reactor Building, accounting for losses from radiological decay and building purging, leads to an equation similar in form to the production of a radioactive material by neutron irradiation, assuming a constant term for isotope production:

$$C(t) = P\left(1 - e^{-\lambda t}\right)/\lambda V$$

where

C(t) = time-dependent concentration of Ar-41 in the building air at time t after the reactor reaching full power

P = Release rate of Ar-41 from the pool

λe	=	decay constant based on the effective half-life, Te, of Ar-41 in the reactor room (see below)
V	=	reactor room volume.

The effective half-life of the Ar-41, T_e, is based on the half-life from radioactive decay, and the half-life from room purging. The reactor room has a volume of 424 cubic meters, and the exhaust fan will move 0.220 cubic meters/s. This means that the room will purge in approximately 32.1 minutes. Assuming an equilibrium condition, we can assume that the half-life from room purging is 32.1 minutes. Therefore, the effective half-life is defined as follows:

$$T_e = (T_d \times T_p) / (T_d + T_p)$$

and the effective half-life of 24.84 minutes is obtained. By substituting the appropriate values into the equation for C(t), and assuming a time t that is sufficiently long to achieve equilibrium (not realistic), the resulting estimate for the concentration of Ar-41 in the room air released from the pool is 1.77×10^{-7} microcuries/cc. This is also well below the DAC for Ar-41 as listed in 10 CFR 20. Even with the addition of the Ar-41 released from the experiment drop tubes, the limit is not reachable such as to impair worker safety or violate 10 CFR 20.

11.1.4 Radiation Protection Program

Purdue University "Executive Memorandum No. B-14" establishes the University administrative structure for radiation protection, and a "Radiation Safety Manual" is published and maintained by REM, which contains the rules and radiation safety procedures for all laboratories using radioisotopes and/or ionizing radiation, including the reactor. Routine surveys are performed of the reactor room and include analysis of the reactor pool and reactor room air by personnel from REM.

The University has a variety of detecting and measuring instruments for monitoring potentially hazardous ionizing radiation. The instrument calibration procedures and techniques ensures that any credible type of radiation and any significant intensities will be detected promptly and measured correctly.

All reactor-related personnel are required to attend a radiation safety training session before they begin work at the reactor. Written procedures have been prepared that address routine health physics monitoring at the University's research reactor facility, and all reactor personnel are trained in these as well.

11.1.5 ALARA Commitment

The University is committed to the principle of ALARA (As Low as Reasonably Achievable), and REM makes every effort to keep doses to a minimum. All unanticipated or unusual exposures are investigated.

11.1.6 Radiation Monitoring and Surveying

11.1.6.1. <u>Fixed-Position Monitors</u>

The PUR-I has 3 fixed-position radiation area monitors (RAM) with adjustable alarm set points and 1 continuous air monitor (CAM) in the reactor room. The CAM air filters are changed and analyzed semi-monthly.

11.1.6.2. Experimental Monitoring

Wipe tests of exposed surfaces of the reactor room are made monthly. Water samples are taken and counted monthly. All samples and material removed from the reactor are checked for levels of activity and wipe tests made for loose contamination.

11.1.6.3. Personnel Monitoring

TLD badges and TLD finger rings are assigned to all approved reactor personnel. In addition, self reading pocket dosimeters and dose rate instruments are used to administratively keep occupational exposures below regulator limits in 10 CFR 20. Students and visitors are provided self-reading pocket dosimeters.

11.1.7 Radiation Exposure Control and Dosimetry

The reactor staff always operates under the principles of ALARA, and personnel exposures are maintained at levels that are well below those required by 10 CFR 20. Personnel exposures for the recent operating history are detailed in Table 11-3 below.

Table 11-3: PUR-1 Personnel Exposures for 2003-2007

	Number of Individuals				
Exposure Range (rem)	**2003**	**2004**	**2005**	**2006**	**2007**
0	0	0	2	3	1
< 0.1	1	1	1	0	2
0.10-0.25	0	0	0	0	0
0.25-0.75	0	0	0	0	0
0.75-1.0	0	0	0	0	0
> 1.0	0	0	0	0	0

11.1.8 Contamination Control

Wipe tests of exposed surfaces of the reactor room are made monthly. Water samples are taken and counted monthly. All samples and material removed from the reactor are checked for levels of activity and wipe tests made for loose contamination.

11.1.9 Environmental Monitoring

11.1.9.1. Airborne Effluent

Argon-41 is produced by thermal neutron activation of Argon-40 in the air. No detectable traces of Ar-41 from air dissolved in the water or in the isotope irradiation tubes has been observed. No detectable tritium has been observed in the pool water. Even at the design power levels of 10 kW the neutron flux is too low to produce detectable quantities.

The main possible source of nitrogen-16 is from the fast neutron interaction with oxygen in the pool water. The nitrogen must then diffuse to the surface of the pool before it is released to the atmosphere. In normal operation, no strong currents are established in the reactor pool and with

the short half-life (7.14 seconds), the nitrogen decays before reaching the surface. No nitrogen-16 has been observed in the reactor room.

A continuous air monitor (CAM), which utilizes a GM tube as a detector, is in operation in the reactor room to indicate long term levels of radiation and to monitor any radioactive particulates released to the air in the room. No airborne radioactivity has been measured at PUR-1. No airborne effluents have been released from the PUR-1 facility.

11.1.9.2. Perimeter monitoring

TLD badges are located at the boundaries of the facility, and are checked for exposure every other month. The doses to these badges are at background levels. Therefore, facility operation has not resulted in public exposures greater than those specified in 10 CFR 20.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

Only very limited contaminated materials are generated by PUR-1. Any contaminated material is disposed of under the Purdue University Broadscope license, as described previously.

11.2.2 Radioactive Waste Controls

Every effort is made to limit the generation of radioactive waste. Very limited amounts of radioactive materials have been produced at PUR-1 in the 46 years of operation.

11.2.3 Release of Radioactive Waste

Disposal of radioactive material is under the Broadscope license, as described previously. No wastes have been released to the environment in an uncontrolled manner.

11.3 Conclusions

The licensed operation of PUR-1 will not result in radiation exposure to facility personnel or members of the public in excess of the limits set forth in 10 CFR 20. Licensed operation of PUR-1 does not result in radioactive material release to the environment. The facility has area radiation monitors, a continuous air monitor, and surveys are performed to ensure that radioactive material is controlled. Personnel and visitors use radiation dosimetry as appropriate.

12 CONDUCT OF OPERATIONS

12.1 Organization

12.1.1 Structure

The reactor facility is an integral part of the Schools of engineering at Purdue University as shown in Figure 6.1 The reactor supervisor has direct responsibility for the operation of the PUR-I. He is responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, Technical specifications, and other applicable regulations.

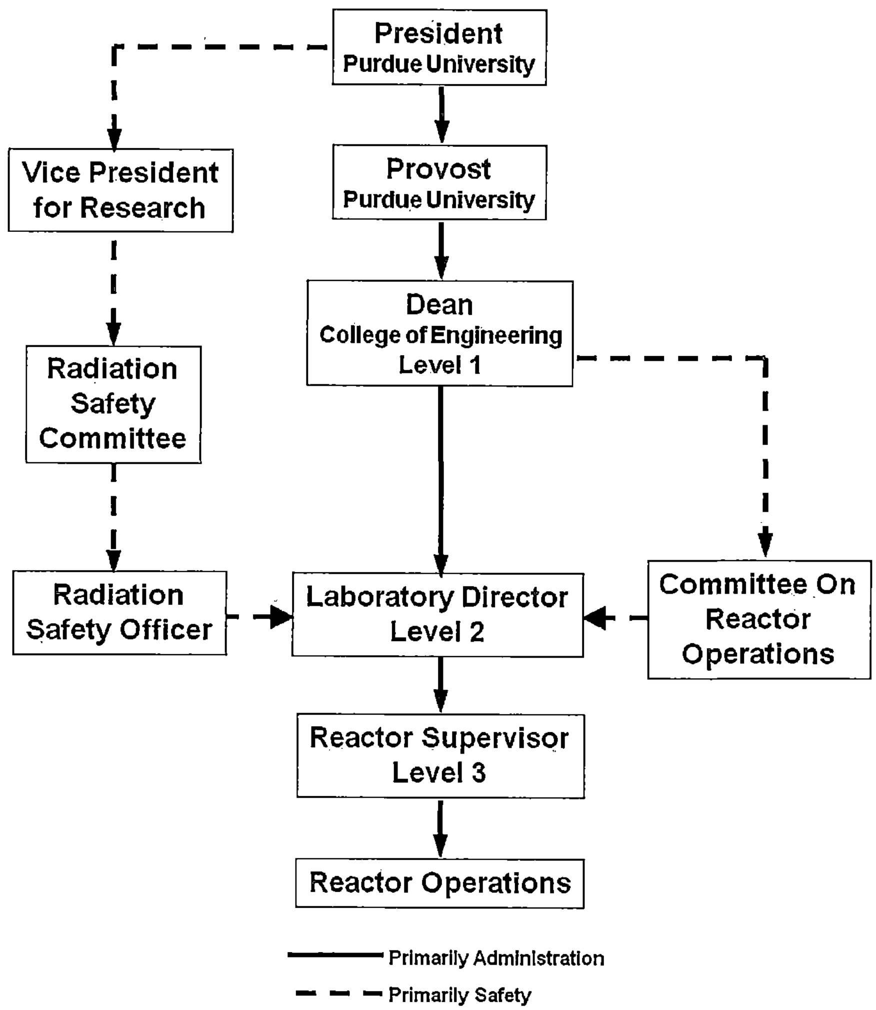

Figure 12-1: Safety and Administration Responsibilities for the PUR-1 Facility

12.1.2 Responsibility

The Dean of Engineering (Level 1) is the individual responsible for the facility's licenses or charter. The Laboratory Director (Level 2) or the designated alternate is responsible for overall reactor facility operation. The Reactor Supervisor (Level 3) shall be responsible for the day-to-day safe operation of the PUR-1. The Reactor Supervisor is be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, including the technical specifications and other applicable regulations.

In all matters pertaining to the operation of the reactor and the administrative aspects of these technical specifications, the Laboratory Director (Level 2) [or the Reactor Supervisor (Level 3) in the absence of the Laboratory Director] shall report to and be directly responsible to the Head of the School of Nuclear Engineering, and the Dean of Engineering (Level 1). In all matters pertaining to radiation safety they will work with the Radiation Safety Officer.

12.1.3 Staffing, Selection and Training of Personnel

At the time of appointment to the position, the Level 1 Licensee (Dean of the College of Engineering) shall receive briefings sufficient to provide an understanding of the general operational and emergency aspects of the reactor facility.

At the time of appointment to the position, the Laboratory Director (Level 2) shall have a minimum of five years of nuclear experience. The individual shall have a recognized baccalaureate or higher degree in an engineering or scientific field. Education or experience that is job-related may be substituted for a degree on a case-by-case basis. The degree may fulfill four years of the six years of nuclear experience required on a one-for-one time basis. The individual shall receive appropriate facility-specific training based upon a comparison of the individual's background and capabilities with the responsibilities and duties of the position. Because of the educational and experience requirements of the position, continued formal training may not be required. The Laboratory Director shall possess a valid Senior Operator License, and meet the certifications requirements of the licensing agency.

Reactor Supervisor (Level 3) - At the time of appointment to the active position, the reactor supervisor shall have a minimum of five years of nuclear experience. He shall have a baccalaureate degree or equivalent experience in an engineering or other scientific field. The degree may fulfill four years of experience on a one-for-one basis. The reactor supervisor shall possess a valid Senior Operator License. During periods when the Reactor Supervisor is absent, these responsibilities may be delegated to a Senior Reactor Operator (Level 4)

Licensed Senior Operator (Level 4) - At the time of appointment to the active position, a senior operator shall have a minimum of a high school diploma or equivalent and should have four years of nuclear experience. A maximum of two years of experience may be fulfilled by related academic or technical training on a one-for-one time basis. He shall hold a valid NRC Senior Reactor Operator's license.

Licensed Operator - At the time of appointment to the active position, an operator shall have a high school diploma or equivalent. He shall hold a valid NRC Reactor Operator's license.

Operator Trainee - An operator trainee shall have all the qualifications to become a licensed operator except for possessing an operator's license.

12.1.4 Radiation Safety

The reactor The reactor facility is an integral part of the Schools of engineering at Purdue University as shown in Figure 6.1 The reactor supervisor has direct responsibility for the operation of the PUR-I. He is responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, Technical specifications, and other applicable regulations.

In all matters pertaining to the administrative aspects of the operation of the reactor, the reactor supervisor reports directly to the Head of the School of Nuclear Engineering or his designated alternate. Financial budgets for the operation of the reactor are handled through the School of Nuclear Engineering.

In all matters pertaining to radiation safety the reactor supervisor is responsible to the Radiological Control committee, usually through the Radiological Control Officer. The Radiological Control Committee was established by the President of the University under an executive memorandum A-50. The duties of the committee, revised under executive memorandum B-14, include responsibility, from the standpoint of safety, for all University programs involving radioactivity or producing radiation. The committee determines the policies and reviews all applications for use of radioactive materials and radiation on the Purdue campus. The Radiological Control Program is administered by the Radiation Control Officer and his staff. The Radiological Control Committee also have several subcommittees reporting directly to it, including the Committee On Reactor Operations (CORO).

Qualifications for the reactor supervisor, reactor staff, and Radiological Control Officers are established in Section 6.1 of the Technical Specifications. The duties of the reactor staff are also specified in Section 6.1 of the Technical Specifications.

12.2 Review and Audit Activities

Review of the facility activities shall be performed by the CORO. Audits of the facility activities shall be performed under the cognizance of the CORO but in no case by the personnel responsible for the item audited. Individual audits may be performed by an individual who needs not be an identified CORO member. These audits shall examine the operating records, procedures, retraining of the facility staff, results of actions taken to correct deficiencies, facility emergency plan, and the facility security plan at intervals designated in the Section 6.2 of the Technical Specifications.

12.3 Procedures

Written procedures for all reactor operations are prepared by the reactor staff and reviewed by the Committee On Reactor Operations (CORO). Those proposed changes to procedures, equipment or systems that change the original intent or use and/or are non-conservative or those that involve an unreviewed safety questions as defined in Section 50.59, 10 CFR are reviewed and approved by CORO before being implemented.

12.4 Required Actions

In the event of a Safety Limit Violation, the following actions shall be taken:

a. The reactor will be shut down immediately and reactor operation will not be resumed without authorization by the Commission.

b. The Safety Limit Violation shall be reported to the Director of the appropriate NRC Office of Inspection and Enforcement (or designee), the Laboratory Director and to the CORO not later than the next work day.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the CORO. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the CORO and the Reactor Supervisor within 14 days of the violation, in support of a request to the Commission for authorization to resume operations.

12.5 Reports

12.5.1 Annual Operating Reports

A report covering the operations for the previous year is submitted to the director of the appropriate NRC office by March 31st of each year. It includes:

1. Changes in Plan design and operation
2. Power generation
3. Unscheduled shutdowns
4. Maintenance
5. Changes, tests and experiments
6. Radioactive effluent releases.

12.5.2 Non-Routine Reports

In the event of a reportable occurrence, notification shall be made within 24 hours by telephone and/or telegraph to the Director of Regional Regulatory Operations Office, followed by a written report within ten days. The written report on these abnormal occurrences, and to the extent possible the preliminary telephone and/or telegraph notification shall:

1. Describe, analyze and evaluate safety implications,
2. Outline the measures taken to assure that the cause of the condition is determined,
3. Indicate that corrective action taken to prevent repetition of the occurrence and/or similar occurrences involving similar components or systems,

4. Evaluate the safety implications of the incident in light of cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.

12.5.3 Unusual Events

A written report shall be forwarded within 30 days to the director of the appropriate Regulatory Operations Office in the event of:

1. Discovery of any substantial errors in the transient or accident analysis or in the methods used for such an analysis, as described in the Hazards Summary Report on the bases for the Technical Specifications.

2. Discovery of any substantial variance from performance specifications contained in the Technical Specifications.

3. Discovery of any condition involving a possible single failure which, for a system designed against assumed single failures, could result in a loss of the capability of the system to perform its safety function.

12.6 Records

Records to be maintained for the life of the facility are stored in filing cabinets in the Reactor Supervisor's office, or in storage cabinets in facility spaces. Records to be kept for five years are also kept in the Reactor Supervisor's office. Recent documentation also has electronic versions kept on private network servers. Security associated documentation is not stored on any servers, and is kept in a secure repository.

12.7 Emergency Planning

The PUR-I reactor Emergency Plan (EP) includes the guidelines, policy, and organization required to mitigate the consequences of an emergency. Specific implementation procedures are provided for each type of emergency in the standard operating procedures for the PUR-I reactor. A revised EP has been approved by the NRC, and is maintained as a separate document from this Safety Analysis Report.

12.8 Security Planning

There is a Physical Security Plan (PSP) for the PUR-I reactor facility which describes the physical protection system and the security organization which provides protection against radiological sabotage and detection of theft of special nuclear material from the facility and related laboratories. The physical security plan was submitted as Amendment 4 and Amendment 5 of the reactor license, and is on file with the NRC. It is withheld from public disclosure pursuant to 10 CFR 2.790 (d).

An audit of the PSP is performed biennially, which is overseen by the Committee on Reactor Operations, and the PSP is revised as necessary when changes are recommended in the audit process, or as other external conditions change. A revised PSP has been approved by the NRC, and is maintained as a separate document from this Safety Analysis Report.

12.9 Quality Assurance

The Laboratory Director and the Reactor Supervisor are responsible for maintaining quality assurance for continued safe operation of the PUR-1 reactor. This includes operator performance monitoring through the requalification and training program, continuous evaluation of existing operating procedures, development of new operating procedures, testing and calibration of reactor equipment, repair of existing equipment, and installation of new equipment. The Committee on Reactor Operations (CORO) provides review, audit and oversight functions as defined in the Technical Specifications for these activities. Checklists and other appropriate documentation, including operations logbooks, are kept to maintain a record of these activities.

12.10 Operator Training and Requalification

The PUR – I reactor facility has an NRC-approved operator requalification program that all licensed reactor operators and senor reactor operators must complete as a condition for renewal of their licenses. Persons who are preparing to take the NRC operator's licensing examination participate in essentially the same training program as well as receive extensive 'hands on' reactor operations training at the console. All licensed operators at the PUR-I participate in the program and must satisfactorily complete this program during each license renewal period. Each licensed operator or senior operator includes in his/her license renewal application a statement that he/she has satisfactorily completed the requirements of the requalification program. The requalification program is divided into three major areas which are designed to provide assurance that all operators maintain competence in all aspects of the licensed activities. The three areas are as follows:

1. Lectures followed by examination of various parts of the reactor operations Technical Specifications, emergency plans, and security plans. Special lectures or makeup studies are used to retrain those operators who demonstrate deficiencies in any part of the examinations.

2. An annual written examination is used to verify the operator's overall knowledge level of reactor operations.

3. An annual evaluation of the operator's performance on the reactor console to actual and/or simulated plant conditions.

13 ACCIDENT ANALYSES

In this chapter, details of the analysis of various accident scenarios are presented. The results of some of these analyses validate the safety system settings established in the Technical Specifications for the PUR-I. The potential effects of the accidents on the health and safety of the staff and public are analyzed.

13.1 Accident-Initiating Events and Scenarios

13.1.1 Maximum Hypothetical Accident

In the scenario of this accident it is assumed that the entire face of a fuel plate is stripped after the maximum power plate has been removed from the reactor after infinite burn time and is suspended in air directly above the reactor pool. The consequences of the release are analyzed for both the reactor staff and general public. As will be showed in subsequent analysis, the potential impact of this postulated accident bounds all other accidents and the complete removal of cladding of the maximum powered fuel plate is designated as the maximum hypothetical accident of the PUR-1.

13.1.2 Insertion of Excess Reactivity

This accident scenario characterizes the reactor response to an insertion of the maximum allowable excess reactivity for the PUR-1 reactor, 0.6% $\Delta k/k$. This transient is examined in four different scenarios, a fast (step) and slow insertion, and each of these 'with' and 'without' scram. In the 'with scram' case, it is assumed that the first scram signal (likely from short period) fails, and the reactor trips on power. The power scram is set for 120% power (12 kW), and it is also assumed that there is a 50% uncertainty in power measurement (actual scram happens at 18 kW), and that the scram signal is delayed by 0.1s before power is cut to the electromagnets that hold the shim-safety rods at their respective heights. It is also assumed that only the least reactive shim-safety rod, SS-2, is able to be inserted into the core (stuck rod scenario).

For the no scram case both rods are stuck at their fully raised position out of the core, which would be similar to the reactor being critical on one shim-safety rod, and that rod being ejected from the core (not a credible accident).

The results for these accidents are summarized in Table 13-1.

Table 13-1: Peak power and clad temperature for trip and no-trip insertions of 0.6% Δk/k.

SCRAM	**Reactivity Inserted**	**P_0 (kW)**	**P_{max} (kW)**	**Time of Peak Power (s)**	**$T_{clad,max}$ (°C)**		
					at t=0	at Peak Power	Maximum
YES	0.6% Δk/k step	10	40.3	0.187	49.94	49.73	49.94
YES	0.6% Δk/k over 10 s	10	18.4	7.83	49.94	42.30	49.98
NO	0.6% Δk/k step	10	2388	672	49.94	133.08	133.08
NO	0.6% Δk/k over 10 s	10	2388	680	49.94	133.08	133.08

13.1.3 Loss of Coolant

The reactor pool is designed to prevent the possibility of an unintentional drainage. It is constructed of steel and set in a second steel tank with the interstitial region filled with sand. The tank rests on a concrete pad about 15 feet below the floor of the Reactor Room, which is in the basement of the building. The pool has no drains. Therefore, a sudden loss of coolant is considered to be extremely remote. If the pool drained instantaneously, while the reactor was operating, the loss of water (moderator) would shut the reactor down.

The most severe problem identified in this accident scenario if the removal of decay heat during and after loss of coolant. There is no danger of significant fuel overheating as long as the core stays immersed and heat can be removed by the water. If the core were to become uncovered, heat transfer would occur by natural convection of ambient air. For this case, the amount of heat removed is proportional to the cladding temperature. According to commonly accepted models, the decay heat generation of a reactor is approximately 6.3% of the original operating power, and it drops off steadily as time goes on. Using this fraction, and applying it to the 12 kW operating power with an additional 50% power measurement uncertainty added in for a total of 18 kW, the power immediately after shutdown will be approximately 1098 W, with an average plate power of approximately 5.8 W. This heat rate will not be enough to lead to fuel damage.

In any reasonably conceivable accident scenario, the leakage of water from the reactor pool is expected to be rather slow. In such a case the radiation area monitor mounted directly above the core would detect any additional radiation coming from the core due to the decreasing pool water level. The pool water level is checked during daily routine operations. It is concluded, that a slow leak of pool water would be discovered early and specific actions could be taken to mitigate it consequences. It is concluded that no adverse consequences are to be expected to the health and safety of the public or the staff from this type of accident.

13.1.4 Loss of Coolant Flow

The Purdue University Reactor is cooled by natural convection, with peak flow rates at the onset of nucleate boiling (ONB, 98.11kW) of 5.41 cm/s, and nominal flow rates at 10 kW of 1.65 cm/s. The only consideration for a loss of coolant flow (LOCF) scenario would be a blockage in a channel. The power density of PUR-1 at 12 kW is such that the individual plate power is very low, and conduction of the heat to adjacent channels would occur, and plate temperatures would remain well below ONB temperatures of 112°C, which is still more than 400°C below the safety limit for the fuel. Therefore this accident is not analyzed.

13.1.5 Mishandling or Malfunction of Fuel

Operation of PUR-1 produces an inventory of radioisotopes in each fuel plate resulting from the fission of 235U. Section 13 of NUREG 1537 suggests that an analysis should be performed for a maximum hypothetical accident (MHA) involving a release of fission products that would have consequences greater than any credible accident. For a low-power (< 2 MW) MTR fuel reactor, the recommended analysis is that resulting from cladding stripped from one face of one fuel plate. In this section, the consequences of an event in which fission fragment radioisotopes are released from a fuel plate to the pool water and subsequently to the atmosphere of the building, ultimately to be released to the outside air, will be analyzed. Since the potential impact of this postulated accident is greater than in any other accident analyzed, the failure removal of cladding from an entire face of a fuel plate is designated as the maximum hypothetical accident of the PUR-I.

Such an event is extremely unlikely during routine operation of the PUR-1. The core operates in a natural convection cooling mode, so failure of a fuel plate as a result of hydraulic pressures in the core is not likely. Previous analyses show that the average coolant velocity in an average channel is 1.8 cm/sec, with 1.93 cm/sec being the coolant velocity in the hot channel. Such coolant velocities are low enough that significant pressures will not be generated anywhere in the core, nor will excessive wear or erosion of the fuel plate surface occur.

Fuel plate cladding failure can occur as a result of corrosion action over years. However, previous studies of MTR-type fuel plates have shown that a cladding hole on the order of several square centimeters in size must be present before significant amounts of radioactivity can be detected in the pool water. It is very unlikely that a hole of such size will form suddenly during normal operations. Current operating procedures call for periodic testing of the pool water to determine radioisotopic content on a regular basis. In addition, fuel element inspections are performed annually. Each year some of the fuel elements are inspected such that all of the elements are inspected at least once in a five year period. During the operating history of the HEU-fueled PUR-1 (over 45 years), no evidence of significant fuel plate surface corrosion (to the point of showing major defects such as formation of thick oxidation layers, fracture defects, or stress lines) has been found.

Fuel element maneuvers are always conducted in the reactor pool. They are removed from the core and moved into the storage racks, one at a time, using a hand-held fuel handling tool. Annually a fuel element is removed from the pool for inspection. A fuel element weighs about 4.5 kg (9.9 lb) in air and only about 2.8 kg (6.2 lb) in water. Therefore, even if a fuel element should fall from the handling tool during its transfer it is not heavy enough to cause any considerable damage. The most severe consequence likely to occur would be some denting of the end fittings since the fuel element, being an elongated object, would tend to fall in water in a rather upright position.

The PUR-I Standard Operating Procedures define administrative steps which are intended to prevent a fuel handling mishap. They are:

1. All fuel handling is done in accordance with written procedures.

2. Loading operations are done by qualified personnel under direct supervision of a Senior Operator.

3. The fuel handling tool is kept locked with the key secured to prevent unauthorized movement of fuel. It is concluded that no adverse consequences are to be expected to the health and safety of the public or the staff from this type of accident.

Even in light of all of this, to provide a bound on the consequences of any hypothetical release of fission products, an analysis was performed in subsequent sections.

13.1.6 Experiment Malfunction

In this section an analysis is performed to assess the hazard associated with the failure of an experiment in which fissile material has been irradiated in the reactor. In the scenario of this accident it is assumed that a capsule containing irradiated fissile material breaks and a portion of the fission product inventory becomes airborne. The consequences of the release are analyzed for both the reactor staff and general public. This accident is analyzed in Section 13.2.1.

The flooding of the 12.7 cm drop tube when placed in position G6 as described in the SER NUREG-1283 resulted in a reactivity insertion of 0.246% $\Delta k/k$, which is below the insertion resulting from the failure of a moveable experiment examined in the conversion proposal. This accident scenario is within the envelope of the examined accident cases, and therefore was not analyzed.

13.1.7 Loss of Normal Electrical Power

The loss of normal electric power at PUR-1 will shutdown the reactor by simulating a scram with a loss of power to the electromagnets that hoist the shim-safety rods. This action will shutdown the reactor from any conceivable operating condition. Since the reactor is cooled by natural convection, there are no shutdown and decay heat issues. There is adequate heat capacity in the reactor pool to address shutdown and decay heat loads. Therefore, this accident scenario is not addressed.

13.1.8 External Events

13.1.8.1. <u>Fire or Explosion</u>

The reactor building is a steel frame structure with concrete block and brick construction. Additionally, the reactor pool is located below ground level, and the reactor itself below the floor level in the reactor room. The materials surrounding the reactor core are concrete, steel and earth. There is limited combustible material in the reactor room, but it is virtually impossible to exclude all burnable materials. Therefore, some small fire possibility exists. There is portable fire-fighting equipment available, and PUR-1 staff members are trained in use of that equipment. Purdue Fire Department is located on campus, and is available for assistance 24 hours a day.

In general, personnel judgment is used in deciding the response to a fire. If the fire is small and can be easily controlled, little action is necessary to safely shutdown the reactor and control the fire. If the fire is of larger magnitude, the reactor will be secured and outside assistance obtained.

Because of the large volume of water surrounding the core, damage to the fuel is not likely even in a severe fire scenario. Even if the fire involved the control systems, power to those circuits could be cut at the wall of the reactor room, and the control rods would drop and safely shut down the reactor.

Explosions external to the building could affect the reactor and its systems; however explosions external to the reactor pool will have little effect on the reactor due to its location and the construction of the pool. The amount of water covering the core will aid in the reduction of risk to pyrotechnic devices thrown into the pool, either serving to quench the devices, or absorb most of the energy from the detonation.

With these facts, and the location of the reactor in the pool, damage from fire or explosion is not likely.

13.1.8.2. Acts of Sabotage

The reactor room is always kept locked except when personnel or materials access is needed. There is a security plan in place, and security systems designed to protect all of the controlled areas in the reactor facility. Purdue Police, a fully functioning police department with arrest authority, is available for assistance, and is staffed 24 hours a day. The PUR-1 Security Plan is reviewed and approved by the U.S. NRC, and provides for in-house procedures to ensure facility protection.

Riots or acts of civil disobedience directed against the reactor facility would be recognizable in sufficient time to ensure safe shutdown of the reactor. Personnel at the reactor could quickly alert local law enforcement, and quickly come under their protection. Then access to the area could be restricted until it is safe to resume normal operations.

13.2 Accident Analysis and Determination of Consequences

13.2.1 Maximum Hypothetical Accident (Mishandling or Malfunction of Fuel)

Section 13 of NUREG 1537 suggests the maximum hypothetical accident for a low power (<2 MW) MTR reactor is the cladding being stripped from the face of one fuel plate. This accident is not seen as credible, but would serve as a bounding case for partial breakage of a fuel plate, or failure of a plate in water with mixing of the fission products in the pool with eventual release to the reactor room air.

For this analysis, the highest power place is chosen from the analysis in Table 4-8. Plate 1348 has a projected plate power of ~97 W at 12 kW, and if a 50% power uncertainty is added to envelope all possible power uncertainties, the plate power for 1348 becomes 145.5 W.

If the plate cladding is stripped, it is the gaseous fission products that will be the main concern, and those will be the only radionuclides analyzed here. Further simplifying assumptions are as follows:

1. All of the fission products are assumed to be at saturation, which assumes an infinite operation at the specified power.
2. the fuel plate which fails does so at the end of this operation,
3. the failed fuel plate is located at the peak flux point in the core,
4. the released radionuclides are perfectly mixed with the reactor facility air volume at the time of release,
5. the primary radionuclide of consequence is elemental iodine.

13.2.1.1. Source Term Estimation for Radioiodine Release

This section will estimate the total amount of radioiodine released from the failed fuel plate. No credit is taken for the reduction in activity resulting from radioactive decay during the time of the release, i.e. an instantaneous release of the radioiodine that can escape the fuel is assumed. Complete and perfect mixing of the available radioiodine inventory with the reactor pool water is also assumed.

The activity via the production rate of the *i*th radioiodine isotope in a plate is determined by the following:

$$A_i = \lambda_i N_i = KPF_i$$

where

- K = conversion constant = 3.1×10^{10} fissions/second/watt
- P = plate power in watts
- F_i = fractional fission yield for the i^{th} radioiodine
- λ_i = decay constant for the i^{th} radioiodine, and
- N_i = saturation number of atoms of the i^{th} radioiodine produced

The constants F_i, λ_i, and the results of calculations for $\lambda_i N_i$ and Ni for plate 1348, are shown in Table 13-2 assuming 18 kW core operating power (to incorporate the aforementioned uncertainties). The total number of radioiodine atoms available in the plate is 1.764×10^{17}. Assuming that the iodine forms the I_2 molecule, the total number of I_2 molecules is thus 8.82×10^{16}.The calculated values for each of the five iodine isotopes of concern are shown in Table 13-2.

Table 13-2: Values and Results for Radioiodine Production in PUR-1 Plate 1348

Isotope	F_i	$T_{1/2}$ (s)	λ_i (s^{-1})	$A_i=\lambda_i N_i$ (dis/s)	N_i (atoms)
I-131	0.029	6.934E+05	9.997E-07	1.308E+11	1.308E+17
I-132	0.043	8.262E+03	8.390E-05	1.940E+11	2.312E+15
I-133	0.065	7.499E+04	9.243E-06	2.932E+11	3.172E+16

I-134	0.08	3.150E+03	2.200E-04	3.608E+11	1.640E+15
I-135	0.064	2.369E+04	2.926E-05	2.887E+11	9.865E+15
				TOTAL	**1.764E+17**
				I_2	**8.82E+16**

It is assumed that not all of the iodine produced will be released from the plate. As suggested NUREG/CR-2079, only the fission fragment gases within recoil range of the surface of the fuel (1.37×10^{-3} cm for aluminum matrix fuels) will escape in this scenario. The thickness of the fuel meat in a PUR-1 plate is 0.0508 cm. Therefore the fraction of the fission product gas release is given by:

$$f = \frac{1.37 \times 10^{-3}\,\text{cm}}{0.0508\,\text{cm}} = 0.0207$$

Therefore, Table 13-2 can be revised as shown in Table 13-3.

Table 13-3: Radioiodines released from Plate 1348 into pool water.

Isotope	**F_i**	**$T_{1/2}$ (s)**	**λ_i (s-1)**	**$A_i = \lambda_i N_i$ (dis/s)**	**N_i (atoms)**	**N_i (atoms) Released**
I-131	0.029	6.934E+05	9.997E-07	1.308E+11	1.308E+17	2.709E+15
I-132	0.043	8.262E+03	8.390E-05	1.940E+11	2.312E+15	4.785E+13
I-133	0.065	7.499E+04	9.243E-06	2.932E+11	3.172E+16	6.566E+14
I-134	0.08	3.150E+03	2.200E-04	3.608E+11	1.640E+15	3.394E+13
I-135	0.064	2.369E+04	2.926E-05	2.887E+11	9.865E+15	2.042E+14
				TOTAL	1.764E+17	3.651E+15
				I_2	8.82E+16	1.826E+15

Therefore, a total of 1.826×10^{15} radioiodine molecules are expected to be released to the reactor pool water. The reactor pool contains about 6400 gallons of water. With the assumption of perfect and complete mixing of the radioiodine in the water, the mole fraction in the pool water is calculated from:

$X_W \quad = (N_I/N_A) / (VK\rho/M)$ = mole fraction of radioiodines in the water

where

N_I = number of radioiodine molecules released
N_A = Avogadro's Number (6.023×10^{23}/mole)
V = Reactor pool volume (6400 gallons)

K	=	conversion factor (3.8×10^3 cm^3/gallon)
ρ	=	density of water (1 gram/cm^3)
M	=	molecular weight of water (18 grams/mole)

Substituting these values in the equation gives a mole fraction of 2.24×10^{-15} for the radioiodine. The partial pressure of radioiodine in air, P_I, can be estimated by:

$$P_I = P_o X_W$$

with the value of X_W from above, and Po being the vapor pressure of pure iodine, which can be estimated by:

$$\log_{10}(P_o) = AC/T + B$$

where

P_o	=	vapor pressure of pure iodine in mm of Hg
A	=	molar heat of vaporization (13057 cal/mol for I_2)
C	=	constant = -0.2185
T	=	bulk pool temperature in degrees Kelvin
B	=	constant = 9.24 (for I_2)

Assuming a bulk pool temperature of 30°C (303 K), a value of 0.32 mm Hg is obtained for P_o. Further substitution yields an estimate for P_I of 1.50×10^{-15} mm Hg for the partial pressure of the radioiodine in air. From this, the molar fraction of radioiodine in air, assuming equilibrium at standard atmospheric pressure, is given as:

$$X_{air} = 1.50 \times 10^{-15} \text{ mm Hg} / 760 \text{ mm Hg} = 1.97 \times 10^{-18}.$$

Assuming the free volume, V_r, of the reactor room is approximately 424 m^3 (4.24×10^5 L), the total moles of radioiodine present can be estimated as:

$$M_I = X_{air} V_r / (24.5 \text{ liters/mole})$$

which yields an estimate of 3.41×10^{-14} moles.

It is now possible to estimate the number of moles of the ith radioiodine, denoted as M_i, from the estimates shown in Table 13-3 for the relative populations of the relevant iodine isotopes. These values are given by:

$$M_i = M_I \cdot \frac{N_i}{N_p}$$ where N_p is the total iodine atoms released from plate 1348.

The activities of the ith radioidine isotope in the building air is then calculated from:

$$A_i = 2 \cdot M_i \cdot N_A \cdot \lambda_i$$

The results are shown in Table 13-4. Now it is possible to estimate the quantity of a given radioiodine present at time *t* after the rupture of plate 1348, by using:

$S_i \quad = S_i(0)e^{-\lambda_i t}$

where

λ_i decay constant of the ith iodine

t = time following iodine release

Si(0) = initial quantity present

For this analysis, we will assume that the quantity S_i is expressed in units of dose (rads) to the thyroid gland of a person breathing the radioiodine-bearing air. Although the concentration of iodine in the building air is continuously reduced by various processes such as radioactive decay, purging of the building air by the exhaust fan, and plating out of the iodine on surfaces, we will assume only a reduction in concentration resulting from radioactive decay. Essentially, this assumes that the releases from the pool surface are balanced by losses other than radioactive decay, and equilibrium is established between the pool water radioiodine and that in the building air. This also implies that the concentrations of iodine in the air outside the building, taking no credit for dissipation in the air outside the building, will be the same as those in the building. It is believed that these assumptions are conservative.

The concentration of a given radioiodine in units of thyroid dose per unit volume of air is given as:

Ci = Si/Vr

where Vr is the free air volume of the reactor room. The results are shown in Table 13-4.

Table 13-4: Number of Moles and Activities of Radioiodine in Reactor Room Air After Release from the Failed Plate

Isotope	**M_i (mol)**	**λ_i (s^{-1})**	**$A_{i(0)}=\lambda_i N_i$ (dis/s)**	**$A_{i(0)}$ (Ci/cm^3)**
I-131	2.53E-14	9.997E-07	8.23E-07	1.94E-15
I-132	4.47E-16	8.390E-05	1.22E-06	2.88E-15
I-133	6.13E-15	9.243E-06	1.84E-06	4.35E-15
I-134	3.17E-16	2.200E-04	2.27E-06	5.35E-15
I-135	1.91E-15	2.926E-05	1.82E-06	4.28E-15

13.2.1.2. Thyroid Dose Consequences

The integrated thyroid dose to a person breathing the reactor room air containing radioiodine contaminate for a time period from 0 to time T is given by:

$$D_i = \int_0^T B \cdot C_i dt = \left[\frac{B \cdot S_i(0)}{V_b}\right]\int_0^T e^{-\lambda t}dt = \frac{B \cdot S_i(0)}{V_b} \cdot \left(1 - e^{-\lambda t}\right)$$

Integrating the expression over the given limits yields

$$D_i = \frac{B \cdot S_i(0)}{V_b \lambda_i}\left(1 - e^{-\lambda T}\right)$$

where B is the standard breathing rate, assumed to be a constant of 3.47×10^{-4} m^3/s (347 cm^3/s). The integrated thyroid dose estimates for varying exposure times are then given as shown in Table 13-5.

If it is assumed that the fission products are instantaneously released and uniformly distributed in the Reactor Room air, and the free volume of the Reactor Room is approximately 4.24×10^8 cm^3 (4.24×10^5 liters). The quantity of a given quantity of radioiodine present at time "t" after the fuel plate rupture is estimated by

$$A_{i,t} = A_{i,(0)} e^{-\lambda_i t}$$

The dose to the thyroid due to inhaled radioiodines can be estimated by multiplying the activities of the isotopes by the Dose Conversion Factors (DCFs) for the respective nuclides (as given in Reg Guide 1.25) or

$$S_i = DCF_i \square A_i.$$

And if follows that the concentration of a given radioiodine in units of thyroid dose per unit volume of air is thus

$$C_i = S_i / V_b$$

Estimates of the respective activities and concentrations of the radioiodines are shown in Table 13-3, and estimates of the thyroid dose rates are shown in Table 13-5.

Table 13-5: Thyroid Dose Rates

Isotope	**$A_i(0)$ (Ci/cm^3)**	**DCF_i (rads/Ci)**	**$C_i(0)$ ($rads/cm^3$)**
I-131	1.94E-15	1.48E+06	2.87E-09
I-132	2.88E-15	5.35E+04	1.54E-10
I-133	4.35E-15	4.00E+05	1.74E-09
I-134	5.35E-15	2.50E+04	1.34E-10
I-135	4.28E-15	1.24E+05	5.31E-10

Table 13-6: Integrated thyroid dose estimates for several exposure periods following release of Plate 1348 radioiodines into the reactor pool.

Isotope	$S_{i(0)}$ (rads/cm^3)	λ_i (s^{-1})	Dose in mrem for Several Exposure Periods					
			90 sec	**2 hours**	**1 day**	**2 days**	**7 days**	**30 days**
I-131	2.87E-09	9.997E-07	0.09	7.15	82.52	158.21	452.45	922.51
I-132	1.54E-10	8.390E-05	0.00	0.29	0.64	0.64	0.64	0.64
I-133	1.74E-09	9.243E-06	0.05	4.21	35.94	52.11	65.09	65.33
I-134	1.34E-10	2.200E-04	0.00	0.17	0.21	0.21	0.21	0.21
I-135	5.31E-10	2.926E-05	0.02	1.20	5.80	6.26	6.30	6.30
			0.17	**13.01**	**125.10**	**217.42**	**524.69**	**994.99**

As can be seen from these results, about one month of continuous exposure to released radioiodine in the reactor room air would be required to attain a thyroid dose equivalent of ~1 rem. Even this exposure would be extremely unlikely, since it is difficult to conceive of a credible combination of accident conditions and personnel occupancy which will result in such doses being achieved. The imposition of such limited cloud dispersion effects required to approach these estimates is not realistic. It is more likely that dispersive effects will result in much lower doses. For example, even the building blower exhaust at 424 CFM volume flow rate will cause a concentration reduction. This dispersion will also be enhanced by natural dispersive effects such as wind speed.

13.2.1.3. MHA considering All Gaseous Fission Products

A similar analysis can be done for all gaseous radionuclides released from a single ruptured fuel plate. The same analysis for iodine as was done in the preceding sections applies, as well as all noble gases (fission products) available in the plate released to the pool water and then to the building air. As before, we will take no credit for decay of the radioisotopes during release and dispersion. This assumption, as before, leads to conservative results in that the estimates obtained are higher than those that would actually occur in this postulated accident.

13.2.1.4. Whole-Body Gamma Dose Estimation

As with the radioiodine release model considered in previous sections, we will assume that the released radionuclides are dispersed into a hemispherical cloud, perfectly mixed with the air in this hemispherical volume. The dose consequences to a person submersed in this cloud are considered.

For a submersion dose, Cember [7] recommends calculating the dose from an infinite hemisphere of gas. To accomplish this, first we calculate the dose from an infinite cloud and then divide by two to account for an infinite hemispherical geometry. By assuming an infinite cloud, we can assume that the density of absorbed energy is equal to the density of emitted energy. This allows us to do a straight-forward calculation from activity density to dose rate. For an isotope i:

$$\dot{D}_i = \frac{A_i}{\rho V} e^{-(\lambda_i+\Lambda)t} E_i F_1 F_2 F_3 F_4 F_5 F_6$$

where

Ai	= source term activity in Ci
ρ	= density of air = 1.293 kg/m3
V	= reactor room volume
λi	= nuclide decay constant in s-1
Λ	= building leakage volume in s-1
t	= time after release in seconds
Ei	= gamma energy in MeV per disintegration
F1	= factor to convert Ci to dis/s
F2	= factor to convert MeV to J = 1.6x10-13
F3	= factor to convert J/kg to rad = 100
F4	= factor to convert rad to rem (Q-factor) = 1
F5	= factor to account for stopping power of tissue ≈1.1
F6	= factor to account for hemispherical geometry

The term $e^{-(\lambda_i+\Lambda)t}$ accounts for reduction in the source term over time due to both radioactive decay and leakage from the building. If we integrate this equation from the release time t=0 to the final time t=T, we get a dose for that period of exposure.

$$\dot{D}_i = \frac{A_i}{\rho V} \frac{e^{-(\lambda_i+\Lambda)t}}{-(\lambda_i+\Lambda)} E_i F_1 F_2 F_3 F_4 F_5 F_6$$

Information relevant to these calculations is shown in Table 13-8. Also, in Table 13-8, the value for the activity $A_i = \lambda_i N_i$ is calculated as follows:

$A_i \quad = \lambda_i N_i = KPF_i$

where

K	= conversion constant = 3.1×10^{10} fissions/second/watt
P	= plate power in watts
F_i	= fractional fission yield for the i^{th} radioisotope
λ_i	= decay constant for the i^{th} radioisotope, and
N_i	= saturation number of atoms of the i^{th} radioisotope produced

Values for the isotopes in question are shown in Table 13-7 and Table 13-8. Dose results for the isotopes in question and for a number of different lengths of time can be seen in Table 13-9 and Table 13-10. Table 13-9 has results for the purge fan turned off, and Table 13-10 has results for the purge fan turned on.

Table 13-7: Calculation results for gaseous fission products released from the failed fuel plate.

Isotope	Fission Yield	λ_i (s^{-1})	Activity Released to pool	Isotope Activity in Reactor Room Air (C_i)	Gamma Energy (MeV/dis)
I-131	0.029	9.997E-07	7.318E-02	8.23E-07	0.4
I-132	0.043	8.390E-05	1.085E-01	1.22E-06	2.12
I-133	0.065	9.243E-06	1.640E-01	1.84E-06	0.55
I-134	0.08	2.200E-04	2.019E-01	2.27E-06	1.25
I-135	0.064	2.926E-05	1.615E-01	1.82E-06	1.50
Kr-85^m	0.013	4.298E-05	3.280E-02	3.28E-02	0.19
Kr-87	0.025	1.514E-04	6.309E-02	6.31E-02	0.63
Kr-88	0.036	4.068E-03	9.084E-02	9.08E-02	2.18
Xe-131^m	0.029	6.776E-07	7.318E-02	7.32E-02	0.002
Xe-133^m	0.065	3.650E-06	1.640E-01	1.64E-01	0.006
Xe-133	0.065	1.529E-06	1.640E-01	1.64E-01	0.08
Xe-135^m	0.064	7.556E-04	1.615E-01	1.62E-01	0.15
Xe-135m	0.064	2.107E-05	1.615E-01	1.62E-01	0.24

Table 13-8: Associated photon information for the gaseous fission products.

isotope	Avg. Gamma Energy (MeV)	Linear Attenuation Coefficient in Air (m-1)
I-131	0.4	3.90×10^{-3}
I-132	0.8	3.70×10^{-3}
I-133	0.55	3.90×10^{-3}
I-134	1.3	3.40×10^{-3}
I-135	1.5	3.30×10^{-3}
Kr-85m	0.2	3.50×10^{-3}
Kr-87	2	3.00×10^{-3}
Kr-88	2	3.00×10^{-3}
Xe-131m	0.16	3.30×10^{-3}
Xe-133m	0.23	3.60×10^{-3}
Xe-133	0.08	3.20×10^{-3}
Xe-135m	0.52	3.90×10^{-3}
Xe-135m	0.25	3.60×10^{-3}

These values that were obtained assuming an infinite cloud set a very conservative upper bound on the dose. If the air volume of the reactor building is considered to be in the shape of a hemisphere, this hemisphere would have a radius of about ten meters. The gammas coming from the nuclides of interest have an average path length in air on the order of 250 meters. This indicates that the infinite cloud assumption is grossly overestimating the actual dose, so we need to use a factor to correct for the non-infinite extent of the cloud.

To obtain this factor, we can think of a spherical cloud consisting of a number of thin concentric shells. Consider a shell at an arbitrary distance 'r' from the origin, which is our point of interest. The contribution of gammas from a point on this shell passing through the origin is:

$$\varphi = \frac{S}{4\pi r^2} e^{-\mu r}$$

where

φ	= flux and the origin
S	= volume-distributed source (Ci/m3)
μ	= gamma absorption coefficient in air (m-1)
r	= distance from origin (m)

This equation accounts for the spread of the radiation away from the point as well as the attenuation from interactions with air. If we sum over all the points on this shell we get a contribution of:

$$\varphi = Se^{-\mu r}$$

Now, by integrating these thin shells over a radius of zero to infinity, we find the quantity of gammas passing through the origin from the infinite cloud.

$$\varphi = \int_0^{\infty} Se^{-\mu r} dr = \frac{S}{\mu}$$

Likewise, integrating from zero to the radius of our hemisphere 'R', we can find the relative quantity of gammas from the finite cloud.

$$\varphi = \int_0^{R} Se^{-\mu r} dr = \frac{S}{\mu}\left[1 - e^{-\mu r}\right]$$

Taking the ratio of these two quantities gives our correction factor.

F = $1 - e^{-\mu R}$

If we take a Taylor expansion of this result and discard the higher-order terms (since μR is small), we get a correction factor of μR. Refer to Table 13-8 for values of the absorption coefficient for the different isotopes. Using this correction factor on the data from Table 13-9 and Table 13-10 gives the dose information seen in Table 13-11 and Table 13-12. These values are not as overly conservative as those found from the infinite-cloud assumption, but they are still conservative due to the approach used to estimate the source term underlying this analysis.

The results show that doses can be kept low to persons inside the building if exposure times are reduced. Thus, room evacuation is an appropriate response to this postulated event. A 90 second evacuation time is reasonable. Both evacuation and shutdown of ventilation systems are part of the emergency response procedures for PUR-1 operation, and form a part of the overall PUR-1 emergency plan. However, even for prolonged exposures to this release, integral whole-body doses can be expected to be lower in actual experience because of the conservative assumptions made in this analysis.

Table 13-9: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Infinite Cloud and a Leakage Fraction of 0.005 Hr^{-1} (Exhaust Fan Off)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-132	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-134	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-135	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-85m	1.01	1.85	2.54	3.98	5.26	5.80	5.86	5.86
Kr-87	6.35	11.44	15.51	23.47	29.67	31.73	31.89	31.89
Kr-88	19.01	23.71	24.88	25.25	25.26	25.26	25.26	25.26
Xe-131m	0.02	0.04	0.06	0.10	0.13	0.15	0.15	0.15
Xe-133m	0.16	0.30	0.41	0.65	0.87	0.97	0.99	0.99
Xe-133	2.14	3.94	5.44	8.64	11.62	13.01	13.20	13.20
Xe-135m	3.55	5.92	7.51	9.75	10.62	10.70	10.70	10.70
Xe-135m	6.31	11.57	15.94	25.15	33.54	37.27	37.74	37.74
	38.56	58.77	72.29	97.00	116.97	124.90	125.79	125.79

Table 13-10: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Infinite Cloud and a Leakage Fraction of 1.87 Hr^{-1} (Exhaust Fan On)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-132	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-134	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-135	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-85m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-87	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01
Kr-88	0.06	0.06	0.06	0.06	0.06	0.06	0.06	0.06
Xe-131m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-133m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-135m	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01
Xe-135m	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01
	0.10	0.10	0.10	0.10	0.10	0.10	0.10	0.10

Table 13-11: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Finite Cloud and a Leakage Fraction of 0.005 Hr^{-1} (Exhaust Fan Off)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-132	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001
I-133	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-134	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001
I-135	0.0000	0.0000	0.0000	0.0001	0.0001	0.0001	0.0001	0.0001
Kr-85m	0.0354	0.0647	0.0889	0.1393	0.1840	0.2029	0.2050	0.2050
Kr-87	0.1905	0.3431	0.4652	0.7042	0.8900	0.9519	0.9566	0.9566
Kr-88	0.5703	0.7114	0.7463	0.7576	0.7577	0.7577	0.7577	0.7577
Xe-131m	0.0008	0.0014	0.0020	0.0032	0.0043	0.0048	0.0049	0.0049
Xe-133m	0.0058	0.0106	0.0147	0.0233	0.0313	0.0350	0.0355	0.0355
Xe-133	0.0686	0.1260	0.1741	0.2764	0.3720	0.4164	0.4224	0.4224
Xe-135m	0.1385	0.2311	0.2929	0.3802	0.4141	0.4173	0.4174	0.4174
Xe-135m	0.2272	0.4164	0.5740	0.9055	1.2075	1.3418	1.3587	1.3587
	1.2371	1.9048	2.3582	3.1899	3.8610	4.1282	4.1584	4.1584

Table 13-12: Integral Whole-Body Gamma Doses Inside the Reactor Room Assuming an Finite Cloud and a Leakage Fraction of 1.87 Hr^{-1} (Exhaust Fan On)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-132	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-133	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-134	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-135	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Kr-85m	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
Kr-87	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004
Kr-88	0.0019	0.0019	0.0019	0.0019	0.0019	0.0019	0.0019	0.0019
Xe-131m	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Xe-133m	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Xe-133	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
Xe-135m	0.0003	0.0003	0.0003	0.0003	0.0003	0.0003	0.0003	0.0003
Xe-135m	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004	0.0004
	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032	0.0032

Doses to persons outside the building will come from submersion in a cloud of released radionuclides and from radiation emitted from the reactor building. The submersion dose results from the diluted radionuclide stream from the exhaust fan or from natural flow of air through the building that exits at the roofline (if the exhaust fan has been shut off). An analysis for the activity concentration released from the building can be performed using the equation below, which includes release from the exhaust fan.

$$A_D = A \cdot q \cdot \Psi(x)$$

where

AD = effective exposure concentration in curies/m3

q = building exhaust rate in m3/second

$\Psi(x)$ = dilution factor at distance x, in sec/m3

A = activity concentration in the exhaust stream

The dilution factor, $\Psi(x)$, is calculated for the leeward side of the building (x=0), and assumes the release is made from the roofline of the building. It is further assumed that the wind velocity is steady at the time of the release, and is equal to 1 m/s. Thus, the dilution factor can be written as:

$$\Psi(x) = 1 / [(0.5) (s) (u)]$$

where

u = wind velocity in m/s

s = building cross sectional area normal to the wind direction (in m^2)

Assuming the prevailing winds are blowing at the time of the release, the cross sectional area of the building is 288 m^2. Making the appropriate substitutions, $\Psi(0)$ is found to be 6.94×10^{-3} s/m^3 for a release from the roofline of the building. Using this value for $\Psi(0)$ with the appropriate values for building exhaust rate gives values for activity concentrations outside the restricted area. These concentrations can then be used to calculate estimates for accumulated doses from the nuclides of interest as was done for immersion dose inside the building. For isotope 'i',

$$D_i = \frac{A_i \cdot q \cdot \Psi}{\rho V} \cdot \frac{1-e^{-(\lambda_i+\Lambda)t}}{} E_i \cdot F_1 \cdot F_2 \cdot F_3 \cdot F_4 \cdot F_5 \cdot F_6$$

where all variables are as defined previously in this section.

Table 13-13 shows the results of this calculation for submersion outside the building with the exhaust fan on. This can be considered to be a bounding case, and the dose rate is still below 10 CFR 20 limits for dose to the public. Turning the fan off would reduce the dose by more then three orders of magnitude. Even if the exhaust fan were left running, accumulated dose to persons outside would be minimal, particularly after the first day.

To calculate the direct dose from the reactor building to someone standing at ground level outside the building and restricted area, assume a half-hemisphere with a volume equivalent to that of the reactor room. The direct dose can then be calculated as a submersion dose from a finite hemisphere divided by two, since the dose only comes from half of a hemisphere.

$$D_i = \frac{A_i}{\rho V} \cdot \frac{1-e^{-(\lambda_i+\Lambda)t}}{\lambda_i+\Lambda} E_i \cdot F_1 \cdot F_2 \cdot F_3 \cdot F_4 \cdot F_5 \cdot F_7 \cdot \mu \cdot R$$

where

- F7 = factor to account for half-hemispherical geometry = ¼
- μ = gamma absorption coefficient in air (m^{-1})
- R = radius of half hemisphere (m)

and all other variables are as defined above.

The results of this calculation with the exhaust fan off are shown in Table 13-14. This estimate is very conservative in that it assumes that a person is standing up against a building wall for an extended period of time, and it does not take into account absorption from building walls or concrete in the building. Therefore the hypothetical dose at this boundary will be estimated to determine if it is within 10 CFR 20 limits.

Table 13-13: Integral Whole-Body Gamma Doses From Submersion Outside of the Restricted Area Assuming a Leakage Fraction of 1.87 Hr^{-1} (Exhaust Fan On)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-132	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-133	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-134	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
I-135	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
Kr-85m	0.0015	0.0028	0.0039	0.0061	0.0080	0.0089	0.0089	0.0089
Kr-87	0.0097	0.0175	0.0237	0.0358	0.0453	0.0484	0.0487	0.0487
Kr-88	0.0290	0.0362	0.0380	0.0386	0.0386	0.0386	0.0386	0.0386
Xe-131m	0.0000	0.0001	0.0001	0.0001	0.0002	0.0002	0.0002	0.0002
Xe-133m	0.0002	0.0005	0.0006	0.0010	0.0013	0.0015	0.0015	0.0015
Xe-133	0.0033	0.0060	0.0083	0.0132	0.0177	0.0199	0.0202	0.0202
Xe-135m	0.0054	0.0090	0.0115	0.0149	0.0162	0.0163	0.0163	0.0163
Xe-135m	0.0096	0.0177	0.0243	0.0384	0.0512	0.0569	0.0576	0.0576
	0.0589	0.0897	0.1104	0.1481	0.1786	0.1907	0.1920	0.1920

Table 13-14: Integral Whole-Body Gamma Doses From Direct (From the Building) Dose Assuming a Leakage Fraction of 0.005 Hr-1 (Purge Fan Off)

	Dose in mrem for various exposure times							
Isotope	5 Minutes	10 Minutes	15 Minutes	30 Minutes	60 Minutes	2 Hours	1 Day	7 Days
I-131	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-132	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-133	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-134	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
I-135	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Kr-85m	0.01	0.02	0.03	0.04	0.05	0.06	0.06	0.06
Kr-87	0.05	0.10	0.13	0.20	0.25	0.27	0.27	0.27
Kr-88	0.16	0.20	0.21	0.22	0.22	0.22	0.22	0.22
Xe-131m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Xe-133m	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.01
Xe-133	0.02	0.04	0.05	0.08	0.11	0.12	0.12	0.12
Xe-135m	0.04	0.07	0.08	0.11	0.12	0.12	0.12	0.12
Xe-135m	0.06	0.12	0.16	0.26	0.34	0.38	0.39	0.39
	0.35	0.54	0.67	0.91	1.10	1.18	1.19	1.19

13.2.2 Insertion of Maximum Allowed Excess Reactivity

The analyses of this transient utilizes the reactor physics and reactivity coefficients determined by MCNP5 as described in Chapter 4, and thermal-hydraulic parameters determined by NATCON as described in Chapter 4, and the PARET/ANL[1] code. For this accident, the initial power before the transient was 12 kW.

The original PARET code has been adapted by the Reduced Enrichment for Research and Test Reactors (RERTR) Program to provide transient and thermal-hydraulics analysis for research and test reactors with both plate and pin-type fuel assemblies. The PARET/ANL version of the code has been subjected to extensive comparisons with the SPERT I and SPERT II (light and heavy water) experiments. These comparisons were quite favorable for a wide range of transients up to and including melting of the clad. Revisions of the code include new and more appropriate heat transfer, departure from nucleate boiling (DNB) and flow instability correlations, improved edits, reactor trips, control insertion model, a decay heat power model, and a loss of flow model.

The rapid insertion of the maximum worth of reactivity excess for PUR-1 (0.6% Δk/k) as specified by the Technical Specifications was evaluated. An assumption was made that the period trip (7s) failed, and scram was initiated on the power trip at 12 kW. Since the assumed measurement uncertainty on core power is 50%, a core power trip setting of 18 kW was utilized in the accident calculation. A delay of 0.1 seconds from the sending of scram signal to beginning of control rod motion is assumed. Fuel/coolant channels that are representative of the hottest fuel plates (as identified in Chapter 4) were modeled in the PARET analyses.

Results of this transient are summarized in Table 13-15. The reactor power increases from 12 kW to the trip setting of 18 kW in less than 2.5 seconds. There is a negligible increase in the clad temperature as a result of this hypothetical accident. The safety limit is never in danger of being reached.

Table 13-15: Peak power and clad temperature for insertions of 0.6% Δk/k with scram.

SCRAM	Reactivity Inserted	P0 (kW)	Pmax (kW)	Time of Peak Power (s)	Tclad,max (°C)		
					at t=0	at Peak Power	Max
YES	0.6% Δk/k step	12	46.4	0.173	57.40	57.40	57.40
YES	0.6% Δk/k over 10 s	12	18.4	6.25	57.40	47.75	57.45

These results demonstrate the ability of the LSSS to protect the safety limit of fuel temperatures not to exceed 530°C. The maximum temperatures achieved in the fuel are well below temperature of incipient boiling as well. Therefore PUR-1 can maintain the fuel integrity during this accident scenario.

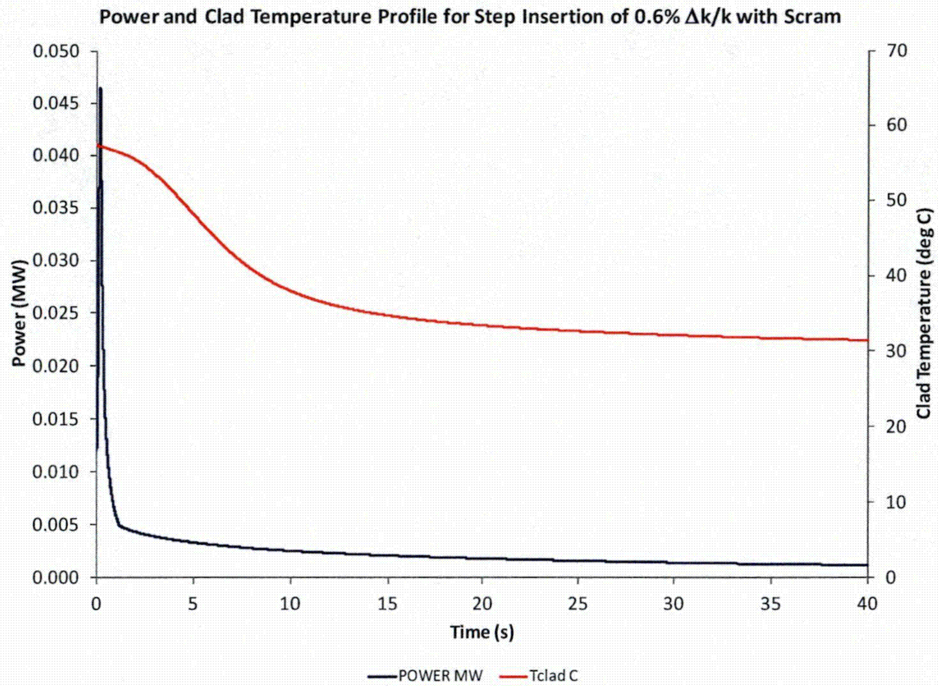

Figure 13-1: Power and Clad Temperatures for 0.6%Δk/k step insertion with scram.

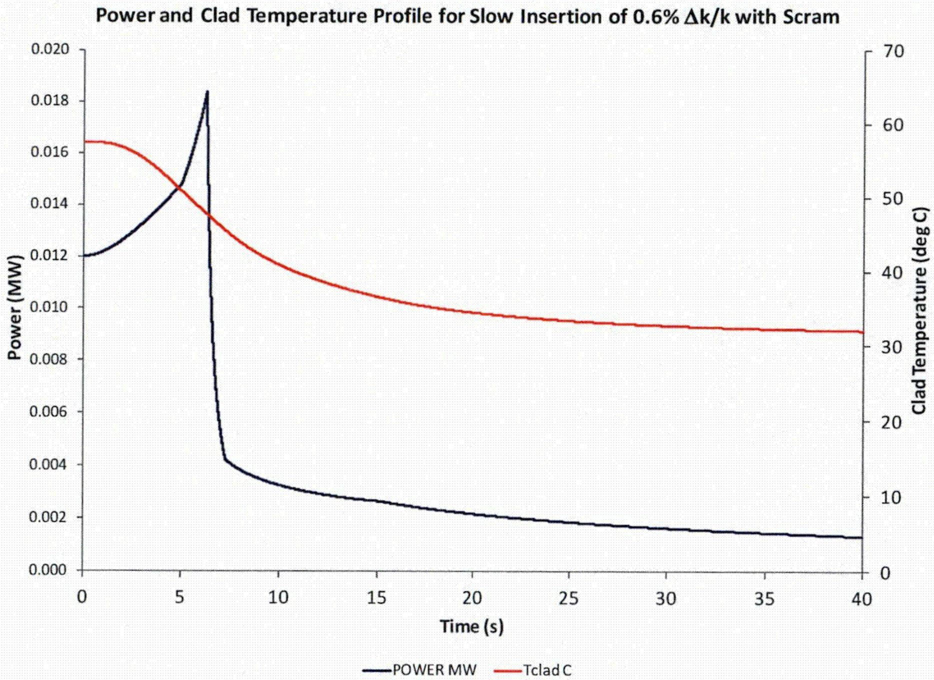

Figure 13-2:Power and Clad Temperatures for 0.6%Δk/k slow insertion with scram.

13.2.2.1. Insertion Without Scram

While it is an extremely unlikely event that all reactor protective systems will fail, the analysis of the step and slow insertion of the maximum excess reactivity without scram was analyzed. In both cases it is assumed that both shim-safety rods are completely removed from the core, and are not available for insertion to shut the reactor down. For this accident, the initial power before the transient was 10 kW.

The reactivity coefficients determined in Chapter 4 act in this case to keep the reactor power from increasing without bounds. The power in both cases (step and slow insertion) levels off at about 2.38 MW, with the maximum clad temperature reaching 133.08°C. This temperature is still well below the safety limit of 530°C, so there is no danger of failure of the clad. It should also be noted that emergency boration is available to the operators, and this would act to shut down the reactor in the unlikely case of failure of the reactor protection system. The peak power and clad temperatures are summarized in Table 13-16, and the power and clad temperature history for the accident are shown in Figure 13-3 and Figure 13-4

Table 13-16: Peak power and clad temperature for trip and no-trip insertions of 0.6% Δk/k.

SCRAM	Reactivity Inserted	P0 (kW)	Pmax (kW)	Time of Peak Power (s)	Tclad,max (°C)		
					at t=0	at Peak Power	Max
NO	0.6% Δk/k step	12	2591	674	54.43	133.56	133.56
NO	0.6% Δk/k over 10 s	12	2591	690	54.43	133.56	133.56

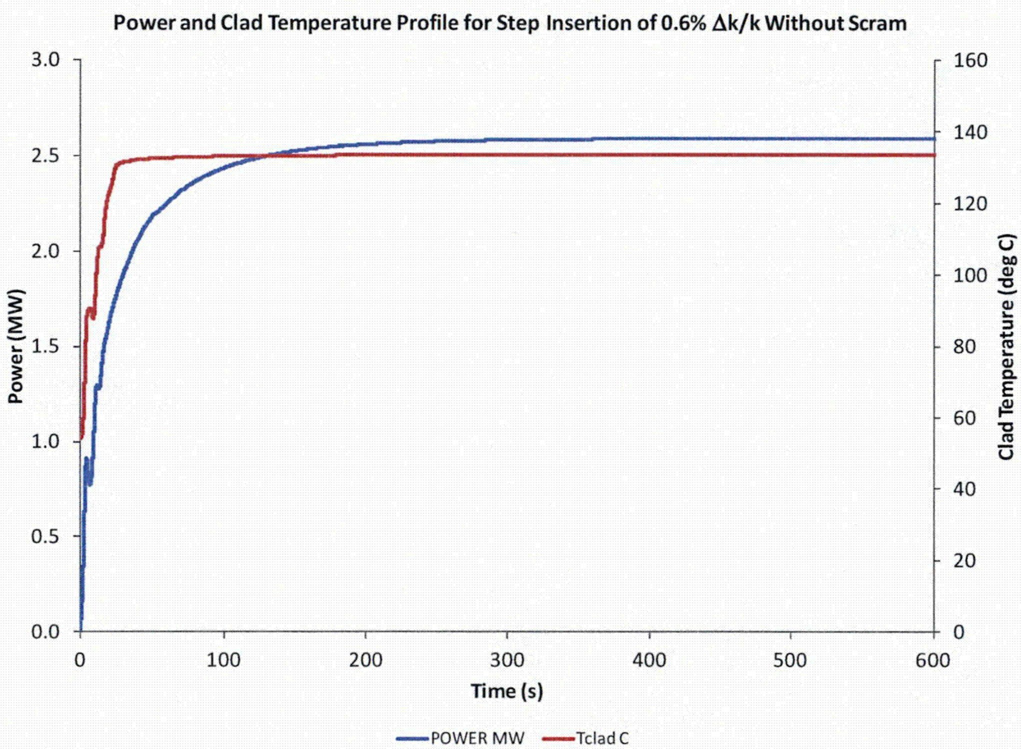

Figure 13-3: Power and Clad Temperatures for 0.6%Δk/k step insertion without scram.

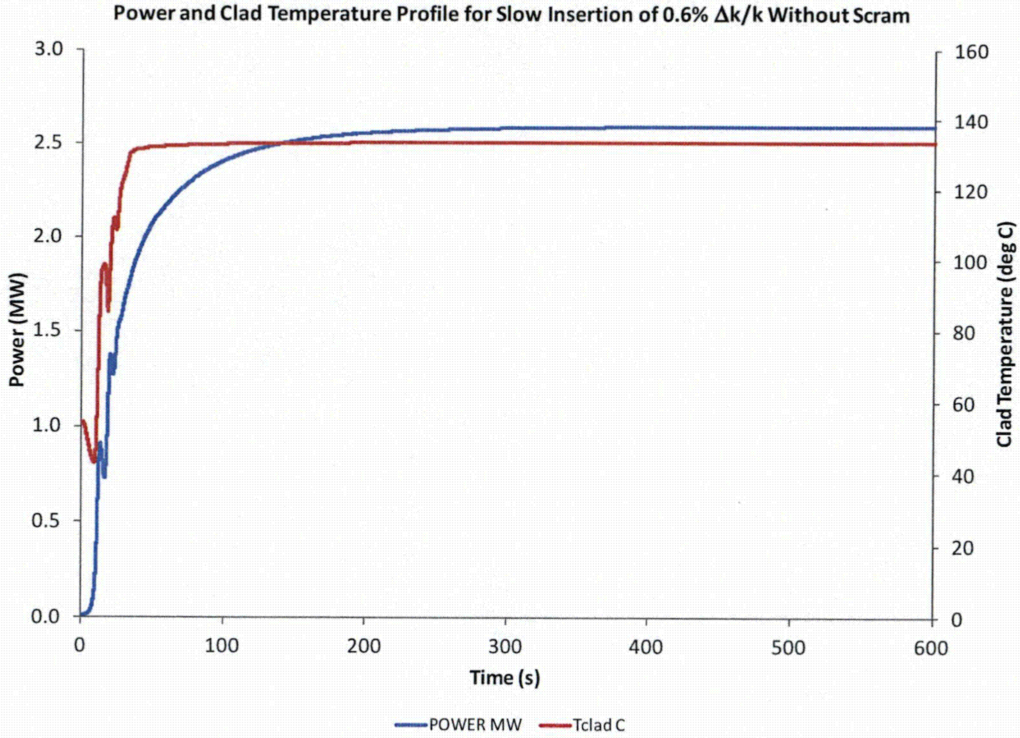

Figure 13-4: Power and Clad Temperatures for 0.6%Δk/k slow insertion without scram.

13.3 Summary and Conclusions

Evaluation of the accident scenarios detailed in this chapter lead to the conclusions that the PUR-1 reactor can operate safely and effectively at 10 kW continuously, with the reactor limiting safety system settings of 110% (11 kW) and 120% (12 kW) adequately protecting the reactor and its systems. Even in the very unlikely event of a complete failure of the reactor protection system and insertion of the maximum allowable reactivity, fuel integrity is maintained and there is little or no risk to the public.

13.4 References

W.L Woodruff and R.S. Smith, "A User's Guide for the ANL Version of the PARET Code, PAREL/ANL (2001 Rev)," Argonne National Laboratory (ANL), ANL/RERTR/TM-16, March 2001.

14 TECHNICAL SPECIFICATIONS

14.1 Summary Description of the Document

The NRC requires each applicant for and holder of a license to operate a non-power reactor to develop technical specifications that state the limits, operating conditions and other requirements for facility operation to protect the environment and health and safety of the facility staff and the public in accordance with 10 CFR 50.36. The technical specifications are typically derived from the facility descriptions and safety considerations contained in the SAR and represent a comprehensive envelope of safe operation.

The technical specifications for PUR-1 were first generated in 1978 to satisfy these requirements. They were made before the publication of the ANSI Standard 15.1, but have been modified over time to meet the evolving format recommendations of that standard.

The technical specifications of the PUR-1 are maintained as a separate document from this safety analysis report.

14.2 Administrative Control of Technical Specifications

The Committee on Reactor Operations (CORO) shall review any proposed changes to the Technical Specifications or licenses.

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Operate a Non-Power Reactor

Since the facility is an existing one, the capital costs are low. The operational costs are minimal while the benefits in the education process of nuclear engineering students, radiation health scientists and technicians are great, both to the individual people and to the national interests. No reasonable alternatives exist to the wide versatility of research/training reactors such as the PUR-1 in contributing to the education and scientific knowledge.

Purdue University, as a large research institution, has the capability to continue safe operation of PUR-1 for the foreseeable future.

15.2 Financial Ability to Decommission the Facility

Purdue University will provide a letter of intent to show it will meet the financial obligations of decommissioning the facility.