TECHNICAL SPECIFICATIONS FOR THE PENN STATE BREAZEALE REACTOR (PSBR) FACILITY LICENSE NO. R-2

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1.0 INTRODUCTION

Included in this document are the Technical Specifications and the bases for the Technical Specifications. These bases, which provide the technical support for the individual technical specifications, are included for information purposes only. They are not part of the Technical Specifications and they do not constitute limitations or requirements to which the licensee must adhere.

1.1 Definitions

1.1.1 ALARA

The ALARA (As Low As Reasonably Achievable) program is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable.

1.1.2 Automatic Control

Automatic control mode operation is when normal reactor operations, including start up, power level change, power regulation, and protective power reductions are performed by the reactor control system without, or with minimal, operator intervention.

1.1.3 Channel

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

1.1.4 Channel Calibration

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

1.1.5 Channel Check

A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

1.1.6 Channel Test

A channel test is the introduction of a signal into the channel to verify that it is operable.

1.1.7 Cold Critical

Cold critical is the condition of the reactor when it is critical with the fuel and bulk water temperatures both below 100°F (37.8°C).

1.1.8 Close Packed Array

Closed packed array is an arrangement of fuel elements wherein no empty grid positions are completely surrounded by fuel elements.

1.1.9 Confinement

Confinement means an enclosure on the overall facility which controls the movement of air into it and out through a controlled path.

1.1.10 Core Lattice Position

The core lattice position is that region in the core over a grid plate hole used to position a fuel element. It may be occupied by a fuel element, a control rod, an experiment, an experimental facility, or a reflector element.

1.1.11 Excess Reactivity

Excess reactivity is that amount of reactivity that would exist if all control rods (safety, regulating, etc.) were moved to the maximum reactive condition from the point where the reactor is exactly critical (keff=1) in the reference core condition.

1.1.12 Experiment

Experiment shall mean (a) any apparatus, device, or material which is not a normal part of the core or experimental facilities, but which is inserted in these facilities or is in line with a beam of radiation originating from the reactor core; or (b) any operation designed to measure reactor parameters or characteristics.

1.1.13 Experimental Facility

Experimental facility shall mean beam port, including extension tube with shields, thermal column with shields, vertical tube, central thimble, in-core irradiation holder, pneumatic transfer system, and in-pool irradiation facility.

1.1.14 Instrumented Element

An instrumented element is a TRIGA fuel element in which sheathed chromel-alumel or equivalent thermocouples are embedded in the fuel.

1.1.15 Limiting Conditions for Operation

Limiting conditions for operation of the reactor are those constraints included in the Technical Specifications that are required for safe operation of the facility. These limiting conditions are applicable only when the reactor is operating unless otherwise specified.

1.1.16 Limiting Safety System Setting

A limiting safety system setting (LSSS) is a setting for an automatic protective device related to a variable having a significant safety function.

1.1.17 Manual Control

Manual control mode is operation of the reactor with the power level controlled by the operator adjusting the control rod positions.

1.1.18 Maximum Elemental Power Density

The maximum elemental power density (MEPD) is the power density of the element in the core producing more power than any other element in that loading. The power density of an element is the total power of the core divided by the number of fuel elements in the core multiplied by the normalized power of that element.

1.1.19 Measured Value

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

1.1.20 Movable Experiment

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

1.1.21 Normalized Power

The normalized power, NP, is the ratio of the power of a fuel element to the average power per fuel element.

1.1.22 Operable

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

1.1.23 Operating

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

1.1.24 Pulse Mode

Pulse mode operation shall mean operation of the reactor allowing the operator to insert preselected reactivity by the ejection of the transient rod.

1.1.25 Reactivity Limits

The reactivity limits are those limits imposed on reactor core reactivity. Quantities are referenced to a reference core condition.

1.1.26 Reactivity Worth of an Experiment

The reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

1.1.27 Reactor Control System

The reactor control system is composed of control and operational interlocks, reactivity adjustment controls, flow and temperature controls, and display systems which permit the operator to operate the reactor reliably in its allowed modes.

1.1.28 Reactor Interlock

A reactor interlock is a device which prevents some action, associated with reactor operation, until certain reactor operation conditions are satisfied.

1.1.29 Reactor Operating

The reactor is operating whenever it is not secured or shutdown.

1.1.30 Reactor Secured:

The reactor is secured when:

a. It contains insufficient fissile material or moderator present in the reactor, adjacent experiments, or control rods, to attain criticality under optimum available conditions of moderation, and reflection, or

b. A combination of the following:

   1) The minimum number of neutron absorbing control rods are fully inserted or other safety devices are in shutdown positions, as required by technical specifications, and

   2) The console key switch is in the off position and the key is removed from the lock, and

   3) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and

   4) No experiments in or near the reactor are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment or one dollar whichever is smaller.

1.1.31 Reactor Shutdown

The reactor is shutdown if it is subcritical by at least one dollar in the reference core condition and the reactivity worth of all experiments is included.

1.1.32 Reactor Safety System

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

1.1.33 Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is negligible (<0.21% $\Delta k/k$(~\$0.30)).

1.1.34 Research Reactor

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

1.1.35 Reportable Occurrence

A reportable occurrence is any of the following which occurs during reactor operation:

a. Operation with the safety system setting less conservative than specified in Section 2.2, Limiting Safety System Setting.

b. Operation in violation of a limiting condition for operation.

c. Failure of a required reactor safety system component which could render the system incapable of performing its intended safety function.

d. Any unanticipated or uncontrolled change in reactivity greater than one dollar.

e. An observed inadequacy in the implementation of either administrative or procedural controls which could result in operation of the reactor outside the limiting conditions for operation.

f. Release of fission products from a fuel element.

g. Abnormal and significant degradation in reactor fuel, cladding, coolant boundary or containment boundary that could result in exceeding 10 CFR Part 20 exposure criteria.

h. Any other violation of NRC regulations.

1.1.36 Rod-Transient

The transient rod is a control rod with SCRAM capabilities that is capable of providing rapid reactivity insertion for use in pulse or square wave mode operation.

1.1.37 Safety Limit

Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain physical barriers which guard against the uncontrolled release of radioactivity. The principal physical barrier is the fuel element cladding.

1.1.38 SCRAM Time

SCRAM time is the elapsed time between reaching a limiting safety system set point and a specified control rod movement.

1.1.39 Secured Experiment

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

1.1.40 Secured Experiment with Movable Parts

A secured experiment with movable parts is one that contains parts that are intended to be moved while the reactor is operating.

1.1.41 Shall, Should, and May

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

1.1.42 Shim, Regulating, and Safety Rods

A shim, regulating, or safety rod is a control rod having an electric motor drive and SCRAM capabilities. It has a fueled follower section.

1.1.43 Shutdown Margin

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

1.1.44 Square Wave Mode

Square wave (SW) mode operation shall mean operation of the reactor allowing the operator to insert preselected reactivity by the ejection of the transient rod, and which results in a maximum power within the license limit.

1.1.45 TRIGA Fuel Element

A TRIGA fuel element is a single TRIGA fuel rod of standard type, either 8.5 wt% U-ZrH in stainless steel cladding or 12 wt% U-ZrH in stainless steel cladding enriched to less than 20% uranium-235.

1.1.46 Watchdog Circuit

A watchdog circuit is a circuit consisting of a timer and a relay. The timer energizes the relay as long as it is reset prior to the expiration of the timing interval. If it is not reset within the timing interval, the relay will de-energize thereby causing a SCRAM.

2.0 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit-Fuel Element Temperature

Applicability

The safety limit specification applies to the maximum temperature in the reactor fuel.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element and/or cladding will result.

Specification

The temperature in a water-cooled TRIGA fuel element shall not exceed 1150°C under any operating condition.

Basis

The important parameter for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured at a point within the fuel element and the relationship between the measured and actual temperature is well characterized analytically. A loss in the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the maximum fuel temperature exceeds 1150°C. The pressure is caused by the presence of air, fission product gases, and hydrogen from the dissociation of the hydrogen and zirconium in the fuel-moderator. The magnitude of this pressure is determined by the fuel-moderator temperature, the ratio of hydrogen to zirconium in the alloy, and the rate change in the pressure.

The safety limit for the standard TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding due to the increase in the hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1150°C and the fuel cladding is below 500°C. See Safety Analysis Report, Ref. 13 and 30 in Section IX and Simnad, M.T., F.C. Foushee, and G.B. West, "Fuel Elements for Pulsed Reactors," Nucl. Technology, Vol. 28, p. 31-56 (January 1976).

2.2 Limiting Safety System Setting (LSSS)

Applicability

The LSSS specification applies to the SCRAM setting which prevents the safety limit from being reached.

Objective

The objective is to prevent the safety limit (1150°C) from being reached.

Specification

The limiting safety system setting shall be a maximum of 650°C as measured with an instrumented fuel element if it is located in a core position representative of the maximum elemental power density (MEPD) in that loading. If it is not practical to locate the instrumented fuel in such a position, the LSSS shall be reduced. The reduction of the LSSS shall be by a ratio based on the calculated linear relationship between the normalized power at the monitored position as compared to normalized power at the core position representative of the MEPD in that loading.

Basis

The limiting safety system setting is a temperature which, if reached, shall cause a reactor SCRAM to be initiated preventing the safety limit from being exceeded. Experiments and analyses described in the Safety Analysis Report, Section IX - Safety Evaluation, show that the measured fuel temperature at steady state power has a simple linear relationship to the normalized power of a fuel element in the core. Maximum fuel temperature occurs when an instrumented element is in a core position of MEPD. The actual location of the instrumented element and the associated LSSS shall be chosen by calculation and/or experiment prior to going to maximum reactor operational power level. The measured fuel temperature during steady state operation is close to the maximum fuel temperature in that element. Thus, 500°C of safety margin exists before the 1150°C safety limit is reached. This safety margin provides adequate compensation for variations in the temperature profile of depleted and differently loaded fuel elements (i.e. 8.5 wt% vs. 12 w % fuel elements). See Safety Analysis Report, Section IX.

If it is not practical to place an instrumented element in the position representative of MEPD the LSSS shall be reduced to maintain the 500°C safety margin between the 1150°C safety limit and the highest fuel temperature in the core if it was being measured. The reduction ratio shall be determined by calculation using the accepted techniques used in Safety Analysis Report, Section IX.

In the pulse mode of operation, the same LSSS shall apply. However, the temperature channel will have no effect on limiting the peak power or fuel temperature, generated, because of its relatively long time constant (seconds), compared with the width of the pulse (milliseconds).

3.0 LIMITING CONDITIONS FOR OPERATION

The limiting conditions for operation as set forth in this section are applicable only when the reactor is operating. They need not be met when the reactor is shutdown unless specified otherwise.

3.1 Reactor Core Parameters

3.1.1 Constant Power and Square Wave Operation

Applicability

These specifications apply to the maximum power generated during manual control mode, automatic control mode, and square wave mode operations.

Objective

The objective is to limit the source term and energy production to that used in the Safety Analysis Report.

Specifications

a. The reactor operational power shall not be intentionally raised above 1 MW except for pulse operation (see specification 3.1.4).

b. The steady state fuel temperature shall be a maximum of 650°C as measured with an instrumented fuel element if it is located in a core position representative of MEPD in that loading. If it is not practical to locate the instrumented fuel in such a position, the steady state fuel temperature shall be calculated by a ratio based on the calculated linear relationship between the normalized power at the monitored position as compared to normalized power at the core position representative of the MEPD in that loading. In this case, the measured steady state fuel temperature shall be limited such that the calculated steady state fuel temperature at the core position representative of the MEPD in that loading shall not exceed 650°C.

Bases

a. Thermal and hydraulic calculations and operational experience indicate that a compact TRIGA reactor core can be safely operated up to power levels of at least 1.15 megawatts with natural convective cooling. Power operation at 1.15 megawatts will not produce fuel temperatures which exceed the LSSS from section 2.2 using any allowed core configuration.

   Small local variations can occur about the maximum allowed power for a given core loading during normal operation and still provide a large margin of safety in that the maximum fuel temperature remains well below the safety limit. See Safety Analysis Report, section IX.

b. Limiting the maximum steady state measured fuel temperature of any position to 650°C places an upper bound on the fission product release fraction to that used in the analysis of a Maximum Hypothetical Accident (MHA). See Safety Analysis Report, section IX.

3.1.2 Reactivity Limitation

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worth of control rods, experiments, and experimental facilities. It applies to all modes of operation.

Objective

The objective is to assure that the reactor is operated within the limits analyzed in the Safety Analysis Report and to assure that the safety limit will not be exceeded.

Specification

The maximum excess reactivity above cold, clean, critical plus samarium poison of the core configuration with experiments and experimental facilities in place shall be 4.9% $\Delta k/k$ (~\$7.00).

Basis

Limiting the excess reactivity of the core to 4.9% $\Delta k/k$ (~\$7.00) prevents the fuel temperature in the core from exceeding 1150°C under any assumed accident condition as described in the Safety Analysis Report, Section IX .

3.1.3 Shutdown Margin

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worth of control rods, experiments, and experimental facilities. It applies to all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times and to assure that the safety limit will not be exceeded.

Specification

The reactor shall not be operated unless the shutdown margin provided by control rods is greater than 0.175% $\Delta k/k$ (~\$0.25) with:

a. All movable experiments, experiments with movable parts, and experimental facilities in their most reactive state.

b. The highest reactivity worth control rod fully withdrawn.

Basis

A shutdown margin of 0.175% $\Delta k/k$ (~\$0.25) assures that the reactor can be made subcritical from any operating condition even if the highest worth control rod should remain in the fully withdrawn position. In addition, the 0.175% $\Delta k/k$ (~\$0.25) can be easily measured, and therefore verifying that the operation meets this specification.

3.1.4 Pulse Mode Operation

Applicability

These specifications apply to the energy generated in the reactor as a result of a pulse insertion of reactivity.

Objective

The objective is to assure that the safety limit will not be exceeded during pulse mode operation.

Specifications

a. The stepped reactivity insertion for pulse operation shall not exceed 2.45% $\Delta k/k$ (~\$3.50) and the maximum worth of the poison section of the transient rod shall be limited to 2.45% $\Delta k/k$ (~\$3.50).

b. Pulses shall not be initiated from power levels above 1 kw.

Bases

a. Experiments and analyses described in the Safety Analysis Report, Section IX.C., show that the peak pulse temperatures can be predicted for new 12 wt% fuel placed in any core position. These experiments and analyses show that the maximum allowed pulse reactivity of 2.45% $\Delta k/k$ (~\$3.50), prevents the maximum fuel temperature from reaching the safety limit (1150°C) for any core configuration that meets the requirements of 3.1.5.

The maximum worth of the pulse rod is limited to 2.45% $\Delta k/k$ (~\$3.50) to prevent exceeding the safety limit (1150°C) with an accidental ejection of the transient rod.

b. If a pulse is initiated from power levels below 1 kw, the maximum allowed full worth of the pulse rod can be used without exceeding the safety limit.

3.1.5 Core Configuration Limitation

Applicability

These specifications apply to all core configurations.

Objective

The objective is to assure that the safety limit (1150°C) will not be exceeded due to power peaking effects in the various core configurations.

Specifications

a. The critical core shall be an assembly of either 8.5 wt% U-ZrH stainless steel clad or a mixture of 8.5 wt% and 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator elements placed in water with a 1.7 inch center line grid spacing.

b. The maximum calculated MEPD shall be less that 24.7 kw per fuel element.

c. When the keff of the core is less than or equal to 0.99 with all control rods at their upper limit, the fuel may or may not be arranged in a close packed array. The source and detector shall be arranged such that the keff of the subcritical assembly shall always be monitored to assure compliance with keff ≤ 0.99 when all control rods are fully withdrawn.

d. The NP of any core loading with a maximum allowed pulse worth of 2.45% $\Delta k/k$ (~\$3.50) shall be limited to 2.2. If the maximum allowed pulse worth is less than 2.45% $\Delta k/k$ (\$~3.50) for any given core loading (i. e. the pulse can be limited by the total worth of the transient rod, by the core excess, or administratively), the maximum NP can be increased. The maximum NP can be increased above 2.2 as long as the calculated maximum fuel temperature does not exceed the safety limit with that maximum allowed pulse worth and NP. In addition, the Reactivity Accident in the Safety Analysis Report shall be evaluated to ensure that the safety limit is not exceeded with the new conditions (See Safety Analysis Report, Section IX.).

Bases

a. The safety analysis is based on an assembly of either 8.5 wt% U-ZrH stainless steel clad or a mixture of 8.5 wt% and 12 wt% U-ZrH stainless steel clad TRIGA fuel-moderator elements placed in water with a 1.7 inch center line grid spacing.

b. Limiting the MEPD to 24.7 kw per element places an upper bound on the elemental heat production and the source term of the PSBR to that used in the analysis of a Loss Of Coolant Accident (LOCA) and Maximum Hypothetical Accident (MHA) respectively. See Safety Analysis Report, Section IX.

c. When the keff of the core is less than or equal to 0.99 with all control rods at their upper limit, the core can not be taken critical. Hence, the requirement for close packed arrays is not necessary to prevent the core from attaining high fuel temperatures.

d. The maximum NP for a given core loading determines the peak pulse temperature with the maximum allowed pulse worth. If the maximum allowed pulse worth is reduced the maximum NP can be increased without exceeding the safety limit (1150°C). The amount of increase in the maximum NP allowed shall be calculated by an accepted method documented by an administratively approved procedure.

3.1.6 TRIGA Fuel Elements

Applicability

These specifications apply to the mechanical condition of the fuel.

Objective

The objective is to assure that the reactor is not operated with damaged fuel that might allow release of fission products.

Specifications

The reactor shall not be operated with damaged fuel except to detect and identify the fuel element for removal. A TRIGA fuel element shall be considered damaged and shall be removed from the core if:

a. In measuring the transverse bend, the bend exceeds the limit of 0.125 inch over the length of the cladding.

b. In measuring the elongation, its length exceeds its original length by 0.125 inch.

c. A clad defect exists as indicated by release of fission products.

Bases

a. The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots which cause damage to the fuel.

b. Experience with TRIGA reactors has shown that fuel element bending that could result in touching has occurred without deleterious effects. This is because (1) during steady state operation, the maximum fuel temperatures are at least 500°C degrees Centigrade below the safety limit (1150°C), and (2) during a pulse, the cladding temperatures remain well below their stress limit. The elongation limit has been specified to assure that the cladding material will not be subjected to strains that could cause a loss of fuel integrity and to assure adequate coolant flow

3.2 Reactor Control and Reactor Safety System

3.2.1 Reactor Control Rods

Applicability

This specification applies to the reactor control rods.

Objective

The objective is to assure that sufficient control rods are operable to maintain the reactor subcritical.

Specification

There shall be a minimum of three operable control rods in the reactor core.

Basis

The shutdown margin and excess reactivity specifications require that the reactor can be made subcritical with the most reactive control rod withdrawn. This specification helps assure it.

3.2.2 Manual Control and Automatic Control

Applicability

This specification applies to the maximum reactivity insertion rate associated with movement of a standard control rod out of the core.

Objective

The objective is to assure that adequate control of the reactor can be maintained during manual and 1, 2, or 3 rod automatic control.

Specification

The rate of reactivity insertion associated with movement of either the regulating, shim, or safety control rod shall be not greater than 0.63% $\Delta k/k$ (~90¢) per second when averaged over full rod travel. If the automatic control uses a combination of more than one rod, the sum of the reactivity of those rods shall be not greater than 0.63% $\Delta k/k$ (~90¢) per second when averaged over full travel.

Basis

The ramp accident analysis (refer to Safety Analysis Report, Chapter IX) indicates that the safety limit (1150°C) will not be exceeded if the reactivity addition rate is less than 1.75% $\Delta k/k$ (~\$2.50) per second, when averaged over full travel. This specification of 0.63% $\Delta k/k$ (~90¢) per second, when averaged over full travel, is well within that analysis.

3.2.3 Reactor Control System

Applicability

This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless the measuring channels listed in Table 1 are operable. (Note that MN, AU, and SW are abbreviations for manual control mode, automatic control mode, and square wave mode, respectively).

Table 1
Measuring Channels

Measuring Channel	Min. No. Operable	Effective Mode MN, AU	Pulse	SW
Fuel Element Temperature	1	X	X	X
Linear Power	1	X		X
Percent Power	1	X		X
Pulse Peak Power	1		X	
Count Rate	1	X		
Log Power	1	X		X
Reactor Period	1	X		

Basis

Fuel temperature displayed at the control console gives continuous information on this parameter which has a specified safety limit. The power level monitors assure that the reactor power level is adequately monitored for the manual control, automatic control, square wave, and pulsing modes of operation. The specifications on reactor power level and reactor period indications are included in this section to provide assurance that the reactor is operated at all times within the limits allowed by these Technical Specifications.

3.2.4 Reactor Safety System and Reactor Interlocks

Applicability

This specification applies to the reactor safety system channels, the reactor interlocks, and the watchdog circuit.

Objective

The objective is to specify the minimum number of reactor safety system channels and reactor interlocks that must be operable for safe operation.

Specification

The reactor shall not be operated unless all of the channels and interlocks described in Table 2a and Table 2b are operable.

Table 2a
Minimum Reactor Safety
System Channels

Channel	Number Operable	Function	Effective Mode MN, AU	Pulse	SW
Fuel Temperature	1	SCRAM ≤ 650°C*	X	X	X
High Power	2	SCRAM ≤ 110% of maximum reactor operational power not to exceed 1.1 MW	X		X
Detector Power Supply	1	SCRAM on failure of supply voltage	X		X
SCRAM Bar on Console	1	Manual SCRAM	X	X	X
Preset Timer	1	Transient Rod SCRAM 15 seconds or less after pulse		X	
Watchdog Circuit	1	SCRAM on software or self-check failure	X	X	X

* The limit of 650°C may be reduced based on specification 2.2.

Table 2b
Minimum Reactor Interlocks

Channel	Number Operable	Function	Effective Mode MN, AU	Pulse	SW
Source Level	1	Prevent rod withdrawal without a neutron induced signal on the startup channel	X		
Log Power	1	Prevent pulsing from levels above 1 kw		X	
Transient Rod	1	Prevent applications of air unless cylinder is fully inserted	X		
Shim, Safety, and Regulating Rod	1	Prevent movement of any rod except the transient rod		X	
Simultaneous Rod Withdrawal	1	Prevent simultaneous manual withdrawal of two rods	X		X

<u>Bases</u>

a. A temperature SCRAM and two power level SCRAMs assure the reactor is shutdown before the safety limit on the fuel element temperature is reached. The actual setting of the fuel temperature SCRAM depends on the LSSS for that core loading and the location of the instrumented fuel element (see Technical Specification section 2.2).

b. The maximum reactor operational power may be administratively limited to less than 1 MW depending on section 3.1.5.b of this Technical Specification. The high power SCRAMs shall be set to no more than 110% of the administratively limited maximum reactor operational power if it is less than 1 MW.

c. Operation of the reactor is prevented by SCRAM if there is a failure of the detector power supply for the reactor safety system channels.

d. The manual SCRAM allows the operator to shut down the reactor in any mode of operation if an unsafe or abnormal condition occurs.

e. The preset timer ensures that the transient rod will be inserted and the reactor will remain at low power after pulsing.

f. The watchdog circuit will SCRAM the reactor if the software or the self-checks fail (see Safety Analysis Report, Section VII).

g. The interlock to prevent startup of the reactor without a neutron induced signal assures that sufficient neutrons are available for proper startup in all allowable modes of operation.

h. The interlock to prevent the initiation of a pulse above 1 kw is to assure that fuel temperature is approximately pool temperature when a pulse is performed. This is to assure that the safety limit is not reached.

i. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the manual control or automatic control mode.

j. In the pulse mode, movement of any rod except the transient rod is prevented by an interlock. This interlock action prevents the addition of reactivity other than with the transient rod.

k. Simultaneous manual withdrawal of two rods is prevented to assure the reactivity rate of insertion is not exceeded.

3.2.5 Core Loading and Unloading Operation

Applicability

This specification applies to the source level interlock.

Objective

The objective of this specification is to allow bypass of the source level interlock during operations with a subcritical core.

Specification

During core loading and unloading operations when the reactor is subcritical, the source level interlock may be momentarily defeated using a spring loaded switch in accordance with the fuel loading procedure.

Basis

During core loading and unloading, the reactor is subcritical. Thus, momentarily defeating the source level interlock is a safe operation. Should the core become inadvertently supercritical, the accidental insertion of reactivity will not allow fuel temperature to exceed the 1150°C safety limit because no single TRIGA fuel element is worth more than 1% $\Delta k/k$ (~$1.43) in the most reactive core position.

3.2.6 SCRAM Time

Applicability

This specification applies to the time required to fully insert any control rod to a full down position from a full up position.

Objective

The objective is to achieve rapid shutdown of the reactor to prevent fuel damage.

Specification

The time from SCRAM initiation to the full insertion of any control rod from a full up position shall be less than 1 second.

Basis

This specification assures that the reactor will be promptly shut down when a SCRAM signal is initiated. Experience and analysis, Safety Analysis Report, Section IX, have indicated that for the range of transients anticipated for a TRIGA reactor, the specified SCRAM time is adequate to assure the safety of the reactor. If the SCRAM signal is initiated at 1.10 MW, while the control rod is being withdrawn, and the negative reactivity is not inserted until the end of the one second rod drop time, the maximum fuel temperature does not reach the safety limit.

3.3 Coolant System

3.3.1 Coolant Level Limits

Applicability

This specification applies to operation of the reactor with respect to a required depth of water above the top of the bottom grid plate.

Objective

The objective is to assure that water is present to provide adequate personnel shielding and core cooling when the reactor is operated, and during a LOCA.

Specification

The reactor shall not be operated with less than 18 ft. of water above the top of the bottom grid plate.

Basis

When the water is more than approximately 18 ft. above the top of the bottom grid plate, the water provides sufficient shielding to protect personnel during operation at 1 MW, and core cooling is achieved with natural circulation of the water through the core. Should the water level drop below approximately 18.25 ft. above the top of the bottom grid plate while operating at 1 MW, a low pool level alarm (see Technical Specifications 3.3.2) will alert the operator who is required by administratively approved procedure to shutdown the reactor. Once this alarm occurs it will take longer than 1300 seconds before the core is completely uncovered because of a break in the 6" pipe connected to the bottom of the pool. Tests and calculations show that, during a LOCA, 680 seconds is sufficient decay time after shutdown (see Safety Analysis Report, Section IX) to prevent the fuel temperature from reaching 950°C. To prevent cladding rupture, the fuel and the cladding temperature must not exceed 950 °C (it is assumed that the fuel and the cladding are the same temperature during air cooling).

3.3.2 Detection of Leak or Loss of Coolant

Applicability

This specification applies to detecting a pool water loss.

Objective

The objective is to detect the loss of a significant amount of pool water.

Specification

A pool level alarm shall be activated and corrective action taken when the pool level drops 26 cm from a level where the pool is full.

Basis

The alarm occurs when the water level is approximately 18.25 ft. above the top of the bottom grid plate. The point at which the pool is full is approximately 19.1 ft. above the top of the bottom grid plate. The reactor staff shall take action to keep the core covered with water according to existing procedures. The alarm is also transmitted to the Police Services annunciator panel which is monitored 24 hrs. a day. The alarm provides a signal that occurs at all times (see Safety Analysis Report, Section VII,). Thus, the alarm provides time to initiate corrective action before the radiation from the core poses a serious hazard.

3.3.3 Fission Product Activity

Applicability

This specification applies to the detection of fission product activity.

Objective

The objective is to assure that fission products from a leaking fuel element are detected to provide opportunity to take protective action.

Specification

An air particulate monitor shall be operating in the reactor bay whenever the reactor is operating. An alarm on this unit shall activate a building evacuation alarm.

Basis

This unit will be sensitive to airborne radioactive particulate matter containing fission products and fission gases and will alert personnel in time to take protective action.

3.3.4 Pool Water Supply for Leak Protection

Applicability

This specification applies to pool water supplies for the reactor pool for leak protection.

Objective

The objective is to assure that a supply of water is available to replenish reactor pool water in the event of pool water leakage.

Specification

A source of water of at least 100 GPM shall be available either from the University water supply or by diverting the heat exchanger secondary flow to the pool.

Basis

Provisions for both of these supplies are in place and will supply more than the specified flow rate. This flow rate will be more than sufficient to handle leak rates that have occurred in the past or any anticipated leak that might occur in the future.

3.3.5 Coolant Conductivity Limits

Applicability

This specification applies to the conductivity of the water in the pool.

Objectives

The objectives are:

a. To prevent activated contaminants from becoming a radiological hazard.

b. To help preclude corrosion of fuel cladding and other primary system components.

Specification

The reactor shall not be operated if the conductivity of the bulk pool water is greater than 5 microsiemens/cm (5 micromhos/cm).

Basis

Experience indicates that 5 microsiemens/cm is an acceptable level of water contaminants in an aluminum/stainless steel system such as that at the PSBR. Based on experience, activation at this level does not pose a significant radiological hazard, and significant corrosion of the stainless steel fuel cladding will not occur when the conductivity is below 5 microsiemens/cm.

3.3.6 Coolant Temperature Limits

Applicability

This specification applies to the pool water temperature.

Objective

The objective is to maintain the pool water temperature at a level that will not cause damage to the demineralizer resins.

Specification

An alarm shall annunciate and corrective action shall be taken if during operation the bulk pool water temperature reaches 100°F (37.8°C).

Basis

This specification is primarily to preserve demineralizer resins. Information available indicates that temperature damage will be minimal up to this temperature.

3.4 Confinement

Applicability

This specification applies tc reactor bay doors.

Objective

The objective is to assure that no large air passages exist to the reactor bay during reactor operation.

Specification

The reactor bay truck door shall be closed when the reactor is operating. Personnel doors to the reactor bay shall not be blocked open and left unattended when the reactor is operating.

Basis

This specification helps to assure that the air pressure in the reactor bay is lower than the remainder of the building and the outside air pressure. Controlled air pressure is maintained by the air exhaust system and assures controlled release of any airborne radioactivity.

3.5 Engineered Safety Features - Facility Exhaust System and Emergency Exhaust System

Applicability

This specification applies to the operation of the facility exhaust system and the emergency exhaust system.

Objective

The objective is to mitigate the consequences of the release of airborne radioactive materials resulting from reactor operation.

Specification

The facility exhaust system shall be operating and the emergency exhaust system shall be maintained in an operable condition when the reactor is operating except for periods of time less than 48 hrs. when it is necessary to permit maintenance and repairs.

Basis

During normal operation, the concentration of airborne radioactivity in unrestricted areas is below effluent release limits as described in the Safety Analysis Report, Section IX. In the event of a substantial release of airborne radioactivity, an air radiation monitor and/or an area radiation monitor will sound a building evacuation alarm which will automatically cause the facility exhaust system to close and the exhausted air to be passed through the emergency exhaust system filters before release. This reduces the radiation within the building. The filters will reduce to <10% all of the particulate fission products that escape to the atmosphere. Radiation monitors, Section 3.6.1 of these Technical Specifications and Safety Analysis Report, Section VII, within the building, independent of the exhaust systems, will give warning of high levels of radiation that might occur during operation with the exhaust systems out of service.

3.6 Radiation Monitoring System

3.6.1 Radiation Monitoring Information

Applicability

This specification applies to the radiation monitoring information which must be available to the reactor operator during reactor operation.

Objective

The objective is to assure that sufficient radiation monitoring information is available to the operator to assure personnel radiation safety during reactor operation.

Specification

The reactor shall not be operated unless the radiation monitoring channels listed in Table 3 are operating.

Table 3
Radiation Monitoring Channels

Radiation Monitoring Channels	Function	Number
Area Radiation Monitor	Monitor radiation levels in the reactor bay.	1
Continuous Air (Radiation) Monitor	Monitor radioactive particulates in the reactor bay air.	1
Beam Laboratory Monitor	Monitor radiation in the Beam Laboratory required only when the laboratory is in use.	1

Bases

a. The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and to take the necessary steps to control the spread of radioactivity to the surroundings.

b. The area radiation monitor in the Beam Laboratory provides information to the user and to the reactor operator when this laboratory is in use.

3.6.2 Evacuation Alarm

Applicability

This specification applies to the evacuation alarm.

Objective

The objective is to assure that all personnel are alerted to evacuate the PSBR building when a potential radiation hazard exists within this building.

Specification

The reactor shall not be operated unless the evacuation alarm is operable and audible to personnel within the PSBR building when activated by the radiation monitoring channels in Table 3 or a manual switch.

Basis

The evacuation alarm produces a loud pulsating sound throughout the PSBR building when there is any impending or existing danger from radiation. The sound notifies all personnel within the PSBR building to evacuate the building as prescribed by the PSBR emergency procedure.

3.6.3 Argon-41 Discharge Limit

Applicability

This specification applies to the concentration of Argon-41 that may be discharged from the PSBR.

Objective

The objective is to insure that the health and safety of the public is not endangered by the discharge of Argon-41 from the PSBR.

Specification

All Argon-41 concentrations produced by the operation of the reactor shall be below the limits imposed by 10 CFR Part 20 when averaged over a year.

Basis

The maximum allowable concentration of Argon-41 in air in unrestricted areas as specified in Appendix B, Table 2 of 10 CFR Part 20 is 1.0×10^{-8} µCi/ml. Measurements of Argon-41 have been made in the reactor bay when the reactor operates at 1 MW. These measurements show that the concentrations averaged over a year produce less than 1.0×10^{-8} µCi/ml in an unrestricted area (see Environmental Impact Appraisal, December 12, 1996).

3.6.4 As Low As Reasonably Achievable (ALARA)

Applicability

This specification applies to all reactor operations that could result in occupational exposures to radiation or the release of radioactive effluents to the environs.

Objective

The objective is to maintain all exposures to radiation and release of radioactive effluents to the environs ALARA.

Specification

An ALARA program shall be in effect.

Basis

Having an ALARA program will assure that occupational exposures to radiation and the release of radioactive effluents to the environs will be ALARA. Having such a formal program will keep the staff cognizant of the importance to minimize radiation exposures and effluent releases.

3.7 Limitations of Experiments

Applicability

These specifications apply to experiments installed in the reactor and its experimental facilities.

Objective

The objective is to prevent damage to the reactor and to prevent excessive release of radioactive materials in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist.

a. The reactivity of a movable experiment and/or movable portions of a secured experiment plus the maximum allowed pulse reactivity shall be less than 2.45% $\Delta k/k$ (~\$3.50). However, the reactivity of a movable experiment and/or movable portions of a secured experiment shall have a reactivity worth less than 1.4% $\Delta k/k$ (~\$2.00). When a movable experiment is used, the maximum allowed pulse shall be reduced below the allowed pulse reactivity insertion of 2.45% $\Delta k/k$ (~\$3.50) to assure that the sum is less 2.45% $\Delta k/k$ (~\$3.50).

b. A single secured experiment shall be limited to a maximum of 2.45% $\Delta k/k$ (~\$3.50). The sum of the reactivity worth of all experiments shall be less than 2.45% $\Delta k/k$ (~\$3.50).

c. When the keff of the core is less than 1 with all control rods at their upper limit and no experiments in or near the core, secured negative reactivity experiments may be added without limit.

d. An experiment may be irradiated or an experimental facility may be used in conjunction with the reactor provided its use does not constitute an unreviewed safety question. The failure mechanisms that shall be analyzed include, but are not limited to corrosion, overheating, impact from projectiles, chemical, and mechanical explosions.

   Explosive material sufficient to cause such damage shall not be stored or used in the facility without proper safeguards to prevent release of fission products or loss of reactor shutdown capability.

   If an experimental failure occurs which could lead to the release of fission products or the loss of reactor shutdown capability, physical inspection shall be performed to determine the consequences and the need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Director or a designated alternate and determined to be satisfactory before operation of the reactor is resumed.

e. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment and reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment, shall be limited in activity such that the airborne concentration of radioactivity averaged over a year shall not exceed the limit of Appendix B Table 2 of 10 CFR Part 20.

   When calculating activity limits, the following assumptions shall be used:

   1) If an experiment fails and releases radioactive gases or aerosols to the reactor bay or atmosphere, 100% of the gases or aerosols escape.

   2) If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.

   3) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these vapors can escape.

   4) For materials whose boiling point is above 130°F and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.

f. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies. In addition, any fueled experiment which would generate an inventory of more than 5 millicuries (mCi) of I-131 through I-135 shall be reviewed to assure that in the case of an accident, the total release of iodine will not exceed that postulated for the MHA (see Safety Analysis Report, Section IX,).

<u>Bases</u>

a. This specification limits the sum of the reactivities of a maximum allowed pulse and a movable experiment to the specified maximum reactivity of the transient rod. This limits the effects of a pulse simultaneous with the failure of the movable experiment to the effects analyzed for a 2.45% $\Delta k/k$ (~\$3.50) pulse. In addition, the maximum power attainable with the ramp insertion of 1.4% $\Delta k/k$ (~\$2.00) is less than 500 kW starting from critical.

b. The maximum worth of all experiments is limited to 2.45% $\Delta k/k$ (~\$3.50) so that their inadvertent sudden removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the temperature safety limit (1150°C). The worth of a single secured experiment is limited to the allowed pulse reactivity insertion as an increased measure of safety. Should the 2.45% $\Delta k/k$, (~\$3.50) reactivity be inserted by a ramp increase, the maximum power attainable is less than 1 MW.

c. Since the initial core is subcritical, adding and then inadvertently removing all negative reactivity experiments leaves the core in its initial subcritical condition.

d. The design basis accident is the MHA (See Safety Analysis Report, Section IX). A chemical explosion, such as detonated TNT or a mechanical explosion, such as a steam explosion or a high pressure gas container explosion may release enough energy to cause release of fission products or loss of reactor shutdown capability. A projectile with a large amount of kinetic energy could cause release of fission products or loss of reactor shutdown capability. Accelerated corrosion of the fuel cladding due to material released by a failed experiment could also lead to release of fission products. No experiment shall be conducted that is an unreviewed safety question

If an experiment failure occurs a special investigation is required to ensure that all effects from the failure are known before operation proceeds.

e. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B Table 2 of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary.

f. The 5 mCi limitation on I-131 through I-135 assures that in the event of failure of a fueled experiment, the exposure dose at the exclusion area boundary will be less than that postulated for the MHA (See Safety Analysis Report, Section IX) even if the iodine is released in the air.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Reactor Parameters

4.1.1 Reactor Power Calibration

Applicability

This specification applies to the surveillance of the reactor power calibration.

Objective

The objective is to verify the performance and operability of the power measuring channel.

Specification

A thermal power channel calibration shall be made on the linear power level monitoring channel annually, not to exceed 15 months.

Basis

The thermal power level channel calibration will assure that the reactor is to be operated at the authorized power levels.

4.1.2 Reactor Excess Reactivity

Applicability

This specification applies to surveillance of core excess reactivity.

Objective

The objective is to assure that the reactor excess reactivity does not exceed the Technical Specifications and the limit analyzed in Safety Analysis Report, Section IX.F.

Specification

The excess reactivity of the core shall be measured annually, not to exceed 15 months, and following core or control rod changes equal to or greater than 0.7% $\Delta k/k$ (~$1.00).

Basis

Excess reactivity measurements on this schedule assure that no unexpected changes have occurred in the core and the core configuration does not exceed excess reactivity limits established in the Technical Specifications.

4.1.3 TRIGA Fuel Elements

Applicability

This specification applies to the surveillance requirements for the TRIGA fuel elements.

Objective

The objective is to verify the continuing integrity of the fuel element cladding.

Specification

All fuel elements and control rods with fuel followers shall be inspected visually for damage or deterioration and measured for length and bend before being placed in the core for the first time and at intervals not to exceed the sum of 3,500 dollars in pulse reactivity or two years, not to exceed 30 months, whichever comes first.

Basis

The frequency of inspection and measurement schedule is based on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known.

4.2 Reactor Control and Safety System

4.2.1 Reactivity Worth

Applicability

This specification applies to the reactivity worth of the control rods.

Objective

The objective is to assure that the control rods are capable of maintaining the reactor subcritical.

Specification

The reactivity worth of each control rod and the shutdown margin for the core loading in use shall be determined annually, not to exceed 15 months, or following core or control rod changes equal to or greater than 0.7% $\Delta k/k$ (~\$1.00).

Basis

The reactivity worth of the control rod is measured to assure that the required shutdown margin is available and to provide an accurate means for determining the core excess reactivity, maximum reactivity, insertion rates, and the reactivity worth of experiments inserted in the core.

4.2.2 Reactivity Insertion Rate

Applicability

This specification applies to control rod movement speed.

Objective

The objective is to assure that the reactivity addition rate specification is not violated and that the control rod drives are functioning.

Specification

The rod drive speed both up and down and the time from SCRAM initiation to the full insertion of any control rod from the full up position shall be measured annually, not to exceed 15 months, or when any significant work is done on the rod drive or the rod.

Basis

This specification assures that the reactor will be promptly shut down when a SCRAM signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified SCRAM time is adequate to assure the safety of the reactor. It also assures that the maximum reactivity addition rate specification will not be exceeded.

4.2.3 Reactor Safety System

Applicability

The specifications apply to the surveillance requirements for measurements, channel tests, and channel checks of the reactor safety systems and watchdog circuit.

Objective

The objective is to verify the performance and operability of the systems and components that are directly related to reactor safety.

Specifications

a. A channel test of the SCRAM function of the high power, fuel temperature, manual, and preset timer safety channels shall be made on each day that the reactor is to be operated, or prior to each operation that extends more than one day.

b. A channel test of the detector power supply SCRAM function and the watchdog circuit shall be performed annually, not to exceed 15 months.

c. Channel checks for operability shall be performed daily on fuel element temperature, linear power, count rate, log power and reactor period when the reactor is to be operated, or prior to each operation that extends more than one day.

d. The percent power channel shall be compared with other independent channels for proper channel indication, when appropriate, each time the reactor is operated.

e. The pulse peak power channel shall be compared to the fuel temperature each time the reactor is pulsed, to assure proper peak power channel operation.

Bases

System components have proven operational reliability.

a. Daily channel tests insure accurate SCRAM functions and insure the detection of possible channel drift or other possible deterioration of operating characteristics.

b. An annual channel test of the detector power supply SCRAM will assure that this system works, based on past experience as recorded in the operation log book. An annual channel test of the watchdog circuit is sufficient to assure operability.

c. The channel checks will make information available to the operator to assure safe operation on a daily basis or prior to an extended run.

d. Comparison of the percent power channel with other independent power channels will assure the detection of channel drift or other possible deterioration of its operational characteristics.

e. Comparison of the peak pulse power to the fuel temperature for each pulse will assure the detection of possible channel drift or deterioration of its operational characteristics.

4.2.4 Reactor Interlocks

Applicability

These specifications apply to the surveillance requirements for the reactor control system interlocks.

Objective

The objective is to insure performance and operability of the reactor control system interlocks.

Specifications

a. A channel check of the source interlock shall be performed each day that the reactor is operated or prior to each operation that extends more than one day.

b. A channel test shall be performed semi-annually, not to exceed 7 1/2 months, on the log power interlock which prevents pulsing from power levels higher than one kilowatt.

c. A channel check shall be performed semi-annually, not to exceed 7 1/2 months, on the transient rod interlock which prevents application of air to the transient rod unless the cylinder is fully inserted.

d. A channel check shall be performed semi-annually, not to exceed 7 1/2 months, on the rod drive interlock which prevents movement of any rod except the transient rod in pulse mode.

e. A channel check shall be performed semi-annually, not to exceed 7 1/2 months, on the rod drive interlock which prevents simultaneous manual withdrawal of more than one rod.

Basis

The channel test and checks will verify operation of the reactor interlock system. Experience at the PSBR indicates that the prescribed frequency is adequate to insure operability.

4.2.5 Overpower SCRAM

Applicability

This specification applies to the high power and fuel temperature SCRAM channels.

Objective

The objective is to verify that high power and fuel temperature SCRAM channels perform the SCRAM functions.

Specification

The high power and fuel temperature SCRAM's shall be tested annually, not to exceed 15 months.

Basis

Experience with the PSBR for more than a decade, as recorded in the operation log books, indicates that this interval is adequate to assure operability.

4.2.6 Transient Rod Test

Applicability

These specifications apply to surveillance of the transient rod mechanism.

Objective

The objective is to assure that the transient rod drive mechanism is maintained in an operable condition.

Specifications

a. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient rod system shall be performed. The transient rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary annually, not to exceed 15 months.

b. The reactor shall be pulsed annually, not to exceed 15 months, to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value or the reactor shall not be pulsed until such comparative pulse measurements are performed.

Basis

Functional checks along with periodic maintenance assure repeatable performance. The reactor is pulsed at suitable intervals and a comparison made with previous similar pulses to determine if changes in transient rod drive mechanism, fuel, or core characteristics have taken place.

4.3 Coolant System

4.3.1 Fire Hose Inspection

Applicability

This specification applies to the dedicated fire hoses used to supply water to the pool in an emergency.

Objective

The objective is to assure that these hoses are operable.

Specification

The two (2) dedicated fire hoses that provide supply water to the pool in an emergency shall be visually inspected for damage and wear annually, not to exceed 15 months.

Basis

This frequency is adequate to assure that significant degradation has not occurred since the previous inspection.

4.3.2 Pool Water Temperature

Applicability

This specification applies to pool water temperature.

Objective

The objective is to limit pool water temperature.

Specification

The pool temperature alarm shall be calibrated annually, not to exceed 15 months.

Basis

Experience has shown this instrument to be drift-free and that this interval is adequate to assure operability.

4.3.3 Pool Water Conductivity

Applicability

This specification applies to surveillance of pool water conductivity.

Objective

The objective is to assure that pool water mineral content is maintained at an acceptable level.

Specification

Pool water conductivity shall be measured and recorded daily when the reactor is to be operated, or at monthly intervals when the reactor is shut down for this time period.

Basis

Based on experience, observation at these intervals provides acceptable surveillance of limits that assure that fuel clad corrosion and neutron activation of dissolved materials will not occur.

4.3.4 Pool Water Level Alarm

Applicability

This specification applies to the surveillance requirements for the pool level alarm.

Objective

The objective is to verify the operability of the pool-water level alarm.

Specification

The pool-water level alarm shall be channel checked monthly, not to exceed 6 weeks, to assure its operability.

Basis

Experience, as exhibited by past periodic checks, has shown that monthly checks of the pool-water level alarm assures operability of the system during the month.

4.4 Confinement

Applicability

This specification applies to reactor bay doors.

Objective

The objective is to assure that reactor bay doors are kept closed as per Specification 3.4.

Specification

Doors to the reactor bay shall be locked or under supervision by an authorized keyholder.

Basis

A keyholder is authorized by the Director or his designee.

4.5 Facility Exhaust System and Emergency Exhaust System

Applicability

These specifications apply to the facility exhaust system and emergency exhaust system.

Objective

The objective is to assure the proper operation of the facility exhaust system and emergency exhaust system in controlling releases of radioactive material to the uncontrolled environment.

Specifications

a. It shall be verified monthly, not to exceed 6 weeks, whenever operation is scheduled, that the emergency exhaust system is operable with correct pressure drops across the filters (as specified in procedures).

b. It shall be verified monthly, not to exceed 6 weeks, whenever operation is scheduled, that the facility exhaust system secures when the emergency exhaust system activates during an evacuation alarm (See Technical Specification 3.6.2 and 5.5).

Basis

Experience, based on periodic checks performed over years of operation, has demonstrated that a test of the exhaust systems on a monthly basis, not to exceed 6 weeks, is sufficient to assure the proper operation of the systems. This provides reasonable assurance on the control of the release of radioactive material.

4.6 Radiation Monitoring System and Effluents

4.6.1 Radiation Monitoring System and Evacuation Alarm

Applicability

This specification applies to surveillance requirements for the area radiation monitor, the beam laboratory radiation monitor, the air radiation monitor, and the evacuation alarm.

Objective

The objective is to assure that the radiation monitors and evacuation alarm are operable and to verify the appropriate alarm settings.

Specification

The area radiation monitor, the beam laboratory radiation monitor, the continuous air (radiation) monitor, and the evacuation alarm system shall be channel tested monthly not to exceed 6 weeks. They shall be verified to be operable by a channel check daily when the reactor is to be operated, and shall be calibrated annually, not to exceed 15 months.

Basis

Experience has shown this frequency of verification of the radiation monitor set points and operability and the evacuation alarm operability is adequate to correct for any variation in the system due to a change of operating characteristics. Annual channel calibration insures that units are within the specifications defined by procedures.

4.6.2 Argon-41

Applicability

This specification applies to surveillance of the Argon-41 produced during reactor operation.

Objective

To assure that the production of Argon-41 does not exceed the limits specified by 10 CFR Part 20.

Specification

The production of Argon-41 shall be measured and/or calculated for each new experiment or experimental facility that is estimated to produce a dose greater than 1 mrem at the exclusion boundary.

Basis

One (1) mrem dose per experiment or experimental facility represents 1% of the maximum 10 CFR Part 20 annual dose. It is considered prudent to analyze the Argon-41 production for any experiment or experimental facility that exceeds 1% of the annual limit.

4.6.3 ALARA

Applicability

This specification applies to the surveillance of all reactor operations that could result in occupational exposures to radiation or the release of radioactive effluents to the environs.

Objective

The objective is to provide surveillance of all operations that could lead to occupational exposures to radiation or the release of radioactive effluents to the environs.

Specification

As part of the review of all operations, consideration shall be given to alternative operational modes that might reduce staff exposures, release of radioactive materials to the environment, or both.

Basis

Experience has shown that experiments and operational requirements can, in many cases, be satisfied with a variety of combinations of facility options, core positions, power levels, time delays, and effluent or staff radiation exposures. Similarly, overall reactor scheduling achieves significant reductions in staff exposures. Consequently, ALARA must be a part of both overall reactor scheduling and the detailed experiment planning.

4.7 Experiments

Applicability

This specification applies to surveillance requirements for experiments.

Objective

The objective is to assure that the conditions and restrictions of Specification 3.7 are met.

Specification

Those conditions and restrictions listed in Specification 3.7 shall be considered by the PSBR authorized reviewer before signing the irradiation authorization for each experiment.

Basis

Authorized reviewers are appointed by the facility director.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

Specifications

The individual unirradiated TRIGA fuel elements shall have the following characteristics:

a. The total uranium content shall be either 8.5 wt% or 12.0 wt% nominal and enriched to less than 20% uranium-235.

b. The hydrogen-to-zirconium atom ratio (in the ZrHx) shall be a nominal 1.65 H atoms to 1.0 Zr atom.

c. The cladding shall be 304 stainless steel with a nominal 0.020 inch thickness.

5.2 Reactor Core

Specifications

a. The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plates.

b. The reflector, excluding experiments and experimental facilities, shall be water, or D_2O, or graphite, or any combination of the three moderator materials.

5.3 Control Rods

Specifications

a. The shim, safety, and regulating control rods shall have SCRAM capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in stainless steel or aluminum cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

b. The transient control rod shall have SCRAM capability and contain borated graphite, B_4C powder, or boron, and its compounds in a solid form as a poison in an aluminum or stainless steel clad. When used as a transient rod, it shall have an adjustable upper limit to allow a variation of reactivity insertions. This rod may incorporate a voided or a solid aluminum follower.

5.4 Fuel Storage

Specifications

a. All fuel elements shall be stored in a geometrical array where the keff is less than 0.8 for all conditions of moderation.

b. Irradiated fuel elements shall be stored in an array which shall permit sufficient natural convection cooling by water such that the fuel element temperature shall not reach the safety limit as defined in Section 2.1 of the Technical Specifications.

5.5 Reactor Bay and Exhaust Systems

Specifications

a. The reactor shall be housed in a room (reactor bay) designed to restrict leakage. The minimum free volume (total bay volume minus occupied volume) in the reactor bay shall be 1900 m^3.

b. The reactor bay shall be equipped with two exhaust systems. Under normal operating conditions, the facility exhaust system exhausts unfiltered reactor bay air to the environment releasing it at a point at least 24 feet above ground level. Upon

initiation of a building evacuation alarm, the previously mentioned system is automatically secured and an emergency exhaust system automatically starts. The emergency exhaust system is also designed to discharge reactor bay air at a point at least 24 feet above ground level.

5.6 Reactor Pool Water Systems

Specification

The reactor core shall be cooled by natural convective water flow.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The University Vice President for Research (level 1) has the responsibility for the reactor facility license. The management of the facility is the responsibility of the Director (level 2), who reports to the Vice President for Research through the Head of the Nuclear Engineering Department and the Dean of the College of Engineering. Administrative and fiscal responsibility is within the offices of the Department Head and the Dean.

The minimum qualifications for the position of Director of the PSBR are an advanced degree in science or engineering, and 2 years experience in reactor operation. Five years of experience directing reactor operations may be substituted for an advanced degree.

The Director can at any time temporarily delegate his authority to the Manager of Operations and Training (level 3) who can in-turn further delegate his authority to a qualified Senior Reactor Operator (level 4).

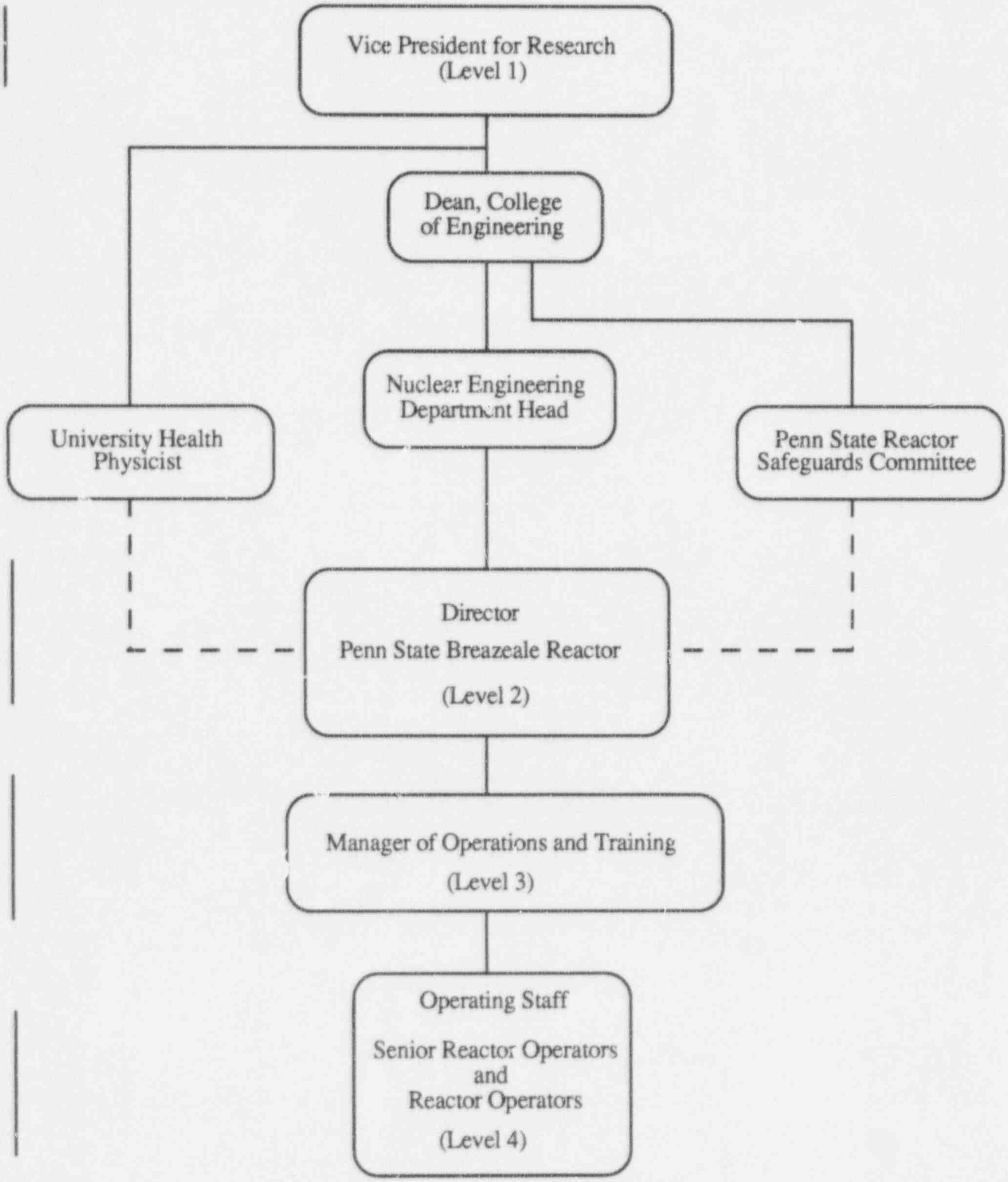
The Operating Staff (level 4) report to the Manager of Operations and Training (level 3) for day-to-day operational matters.

The University Health Physicist reports through the Director of Intercollege Programs to the Office of the Vice President for Research. The qualifications for the University Health Physicist position are the equivalent of a graduate degree in radiation protection, 3 to 5 years experience with a broad byproduct material license, and certification by The American Board of Health Physics or eligibility for certification.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be within the chain of command shown in the organization chart. Individuals at the various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license and technical specifications.

In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.



ORGANIZATION CHART

6.1.3 Staffing

a. The minimum staffing when the reactor is not secured shall be:

1) A licensed operator present in the control room, in accordance with applicable regulations.

2) A second person present at the facility able to carry out prescribed written instructions.

3) If a senior reactor operator is not present at the facility, one shall be available by telephone and able to be at the facility within 30 minutes.

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

1) Management personnel.

2) Radiation safety personnel.

3) Other operations personnel.

c. Events requiring the direction of a Senior Reactor Operator shall include:

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

2) Relocation of any in-core experiment with a reactivity worth greater than one dollar.

3) Recovery from unplanned or unscheduled shutdown (in this instance, documented verbal concurrence from a Senior Reactor Operator is required).

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of all applicable regulations and the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1977, Sections 4-6.

6.2 Review and Audit

6.2.1 Safeguards Committee Composition

A Penn State Reactor Safeguards Committee (PSRSC) shall exist to provide an independent review and audit of the safety aspects of reactor facility operations. The committee shall have a minimum of 5 members and shall collectively represent a broad spectrum of expertise in reactor technology and other science and engineering fields. The committee shall have at least one member with health physics expertise. The committee shall be appointed by and report to the Dean of the College of Engineering. The PSBR Director shall be an ex-officio member of the PSRSC.

6.2.2 Charter and Rules

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

a. Meeting frequency - not less than once per calendar year not to exceed 15 months.

b. Quorums - at least one-half of the voting membership shall be present (the Director who is ex-officio shall not vote) and no more than one-half of the voting members present shall be members of the reactor staff.

c. Use of Subgroups - the committee chairman can appoint ad-Hoc committees as deemed necessary.

d. Minutes of the meetings - shall be recorded, disseminated, reviewed, and approved in a timely manner.

6.2.3 Review Function

The following items shall be reviewed:

a. 10 CFR Part 50.59 reviews of:

1) Proposed changes in equipment, systems, tests, or experiments.

2) All new procedures and major revisions thereto having a significant effect upon safety.

3) All new experiments or classes of experiments that could have a significant effect upon reactivity or upon the release of radioactivity.

b. Proposed changes in technical specifications, license, or charter.

c. Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance.

d. Operating abnormalities having safety significance.

e. Special reports listed in 6.6.2.

f. Audit reports.

6.2.4 Audit

The audit function shall be performed annually, not to exceed 15 months, preferably by a non-member of the reactor staff. The audit function shall be performed by a person not directly involved with the function being audited. The audit function shall include selective (but comprehensive) examinations of operating records, logs, and other documents. Discussions with operating personnel and observation of operations should also be used as appropriate. Deficiencies uncovered that affect reactor safety shall promptly be reported to the Nuclear Engineering Department Head and the Dean of the College of Engineering. The following items shall be audited:

a. Facility operations for conformance to Technical Specifications, license, and procedures (at least once per calendar year with interval not to exceed 15 months).

b. The requalification program for the operating staff (at least once every other calendar year with the interval not to exceed 30 months).

c. The results of action taken to correct deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety (at least once per calendar year with the interval not to exceed 15 months).

d. The reactor facility emergency plan and implementing procedures (at least once every other calendar year with the interval not to exceed 30 months).

6.3 Operating Procedures

Written procedures shall be reviewed and approved prior to the initiation of activities covered by them in accordance with Section 6.2.3. The procedures in this section preceded by an asterisk shall also be approved and initialed by a representative of the University Health Physics Office. Written procedures shall be adequate to assure the safe operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect and shall be followed for at least the following items:

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

b. Core loading, unloading, and fuel movement within the reactor.

c. Routine maintenance of major components of systems that could have an effect on reactor safety.

d. Surveillance tests and calibrations required by the technical specifications (including daily checkout procedure).

e. Radiation, evacuation, and alarm checks.

*f. Release of Irradiated Samples.

*g. Evacuation.

*h. Fire or Explosion.

*i. Gaseous Release.

*j. Medical Emergencies.

*k. Civil Disorder.

*l. Bomb Threat.

*m. Threat of Theft of Special Nuclear Material.

*n. Theft of Special Nuclear Material.

*o. Industrial Sabotage.

*p. Experiment Evaluation and Authorization.

*q. Reactor Operation Using a Beam Port.

*r. D_20 Handling.

*s. Health Physics Orientation Requirements.

*t. Hot Cell Entry Procedure.

u. Implementation of emergency and security plans.

v. Radiation instrument calibration

w. Loss of pool water.

6.4 Review and Approval of Experiments

a. All new experiments shall be reviewed for Technical Specifications compliance, 10 CFR Part 50.59 analysis, and approved in writing by level 2 management or designated alternate prior to initiation.

b. Substantive changes to experiments previously reviewed by the PSRSC shall be made only after review and approval in writing by level 2 management or designated alternate.

6.5 Required Action

6.5.1 Action to be Taken in the Event the Safety Limit is Exceeded

In the event the safety limit (1150°C) is exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.

b. The safety limit violation shall be promptly reported to level 2 or designated alternates.

c. An immediate report of the occurrence shall be made to the Chairman, PSRSC and reports shall be made to the USNRC in accordance with Specification 6.6.

d. A report shall be prepared which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the PSRSC for review.

6.5.2 Action to be Taken in the Event of a Reportable Occurrence

In the event of a reportable occurrence, (1.1.33) the following action shall be taken:

a. The reactor shall be returned to normal or shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operations shall not be resumed unless authorized by level 2 or designated alternates.

b. The Director or a designated alternate shall be notified and corrective action taken with respect to the operations involved.

c. The Director or a designated alternate shall notify the Nuclear Engineering Department Head who, in turn, will notify the office of the Dean of the College of Engineering and the office of the Vice President for Research.

d. The Director or a designated alternate shall notify the Chairman of the PSRSC.

e. A report shall be made to the PSRSC which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be reviewed by the PSRSC at their next meeting.

f. A report shall be made to the USNRC, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator.

6.6 Reports

6.6.1 Operating Reports

An annual report shall be submitted within 6 months of the end of The Pennsylvania State University fiscal year to the USNRC, Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, including at least the following items:

a. A narrative summary of reactor operating experience including the energy produced by the reactor, and the number of pulses ≥ \$2.00 but less than or equal to \$2.50 and the number greater than \$2.50.

b. The unscheduled shutdowns and reasons for them including, where applicable, corrective action taken to preclude recurrence.

c. Tabulation of major preventive and corrective maintenance operations having safety significance.

d. Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests and experiments, that are significantly different from those performed previously and are not described in the Safety Analysis Report, including a summary of the analyses leading to the conclusions that no unreviewed safety questions were involved and that 10 CFR Part 50.59 was applicable.

e. A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed or recommended, only a statement to this effect need be presented.

f. A summarized result of environmental surveys performed outside the facility.

6.6.2 Special Reports

Special reports are used to report unplanned events as well as planned major facility and administrative changes. These special reports shall contain and shall be communicated as follows:

a. There shall be a report no later than the following working day by telephone and confirmed in writing by telegraph or similar conveyance to the USNRC, Operations Center, Washington, DC 20555, to be followed by a written report to the USNRC, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Regional Administrator, that describes the circumstances of the event within 14 days of any of the following:

   1) Violation of safety limits (See 6.5.1)

   2) Release of radioactivity from the site above allowed limits (See 6.5.2)

   3) A reportable occurrence (Section 1.1.35)

b. A written report shall be made within 30 days to the USNRC, and to the offices given in 6.6.1 for the following:

   1) Permanent changes in the facility organization involving level 1-2 personnel.

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

6.7 Records

To fulfill the requirements of applicable regulations, records and logs shall be prepared, and retained for the following items:

6.7.1 Records to be Retained for at Least Five Years

a. Log of reactor operation and summary of energy produced or hours the reactor was critical.

b. Checks and calibrations procedure file.

c. Preventive and corrective electronic maintenance log.

d. Major changes in the reactor facility and procedures.

e. Experiment authorization file including conclusions that no unreviewed safety questions were involved for new tests or experiments.

f. Event evaluation forms (including unscheduled shutdowns) and reportable occurrence reports.

g. Preventive and corrective maintenance records of associated reactor equipment.

h. Facility radiation and contamination surveys.

i. Fuel inventories and transfers.

j. Surveillance activities as required by the Technical Specifications.

k. Records of PSRSC reviews and audits.

6.7.2 Records to be Retained for at Least One Training Cycle

a. Requalification records for licensed reactor operators and senior reactor operators.

6.7.3 Records to be Retained for the Life of the Reactor Facility

a. Radiation exposure for all facility personnel and visitors.

b. Environmental surveys performed outside the facility.

c. Radioactive effluents released to the environs.

d. Drawings of the reactor facility including changes.

ENVIRONMENTAL IMPACT APPRAISAL
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Bonnie L. Burris

Notarial Seal
Bonnie L. Burris, Notary Public
State College Boro, Centre County
My Commission Expires Nov. 22, 1999

ENVIRONMENTAL IMPACT APPRAISAL

1.0 Facility, Environmental Effects of Construction

This Environmental Impact Appraisal (EIA) is submitted for the continued operation of The Penn State Breazeale Reactor (PSBR) under the license R-2. Because the PSBR has a power level less than 2 MW(th), an Environmental Impact Statement (EIS) is not needed.

The PSBR core is a mixture of 8.5 wt% and 12 wt% standard TRIGA fuel and is supported on a movable bridge structure in an open concrete pool of water containing 71,000 gallons of demineralized water.

The building housing the reactor is a standard industrial-type steel building. When the evacuation alarm sounds, the normal roof fan exhaust system closes and an emergency exhaust system is activated automatically.

Supporting facilities include a Cobalt-60 irradiation facility, two hot cells, various laboratories, evaporator building, pool water storage tank, and liquid waste storage tanks.

No plans for construction exist that would affect the environment, since the reactor is located in an existing building at a developed site.

2.0 Environmental Effects of Facility Operation

2.1 Thermal Discharges

The PSBR heat exchanger limits the temperature of the PSBR pool water. Maintaining a low pool temperature decreases pool water evaporation losses and a temperature below 100°F is needed to prevent deterioration of the demineralizer anion resins.

The system is comprised of two loops. In the primary loop, pool water is pumped through the baffled shell side of two double pass heat exchangers connected in series. In the secondary loop, cooling water is pumped from Thompson Pond (650 yards from the PSBR) through the tube bundle side of the two heat exchangers and then to a storm sewer which returns the water to Thompson Pond.

Relatively small variations in cooling water temperatures are noted (55° ± 2°F) but other factors such as flow rates, pool temperature, and system cleanliness affect the system's heat removal capacity.

A check on the efficiency of the heat exchanger was performed in August 1984. During this check, the pool recirculation system was secured and two mixing pumps were installed on the pool divider wall to provide a more uniform pool temperature. The pool was cooled from 99°F to 73.5°F and periodic measurements were taken of the heat exchanger pool in and pool out temperatures. Calculations show that with a pool inlet temperature of 99°F the heat exchanger is removing 1 MW of heat.

Thompson Pond is fed by a spring whose output averages 4.5×10^6 gallons per day on a yearly basis, with a dry season low of 3×10^6 gallons per day and a rainy season high of 7×10^6 gallons per day. Assuming a 1 MW heat input from the heat exchanger, the temperature rise of the 4.5×10^6 gallons per day would be 2.2°F (3.3°F in the dry season and 1.4°F in the wet season). However, the reactor pool seldom reaches a temperature as high as 99°F where the heat exchanger heat removal capacity is 1 MW. Records for the 1983 calendar year show the highest recorded pool temperature as 80.6°F, resulting in a heat removal rate of 0.6 MW. So in the worst case dry season, the temperature rise of the 3×10^6 gallons per day would be about 2°F at a pool temperature of 80.6°F.

The above discussion assumes that the heat exchanger effluent contributes its heat directly to Thompson Pond. As earlier noted, the heat exchanger effluent travels to Thompson Pond by way of a storm sewer where some heat loss would take place making the calculated temperature rise even less than calculated.

PSBR thermal discharge presents no significant hazard to the environment.

2.2 Radioactive Discharges

2.2.1 Argon-41

Gaseous effluent Ar-41, is released from dissolved air in the reactor pool water, dry irradiation tubes, and air leakage from pneumatic sample transfer systems. The amount of Ar-41 released from the reactor pool is very dependent upon the operating power level and the length of time at power. The release per MWH is highest for extended high power runs and lowest for intermittent low power runs. The concentration of Ar-41 in the reactor bay and the bay exhaust was measured by the Health Physics staff during the summer of 1986. Measurements were made for conditions of low and high power runs simulating typical operating cycles. Based on these measurements, an annual release of between 186 mCi and 564 mCi of Ar-41 is calculated for July 1, 1995, to June 30, 1996, resulting in an average concentration at ground level outside the reactor building that is 0.3% to 0.9% of the effluent concentration limit in Appendix B to 10 CFR 20.1001 to 20.2402. The concentration at ground level is estimated using only dilution by a 1 m/s wind into the lee of the 200 m^2 cross section of the reactor bay.

During the report period, several irradiation tubes were used at high enough power levels and for long enough runs to produce significant amounts of Ar-41. The calculated annual production was 55 mCi. Since this production occurred in a stagnant volume of air confined by close fitting shield plugs, most of the Ar-41 decayed in place before being released to the reactor bay. The reported releases from dissolved air in the reactor pool are based on measurements made, in part, when a dry irradiation tube was in use at high power levels; the Ar-41 released from the tubes are part of rather than in addition to the release figures quoted in the previous paragraph. The use of the pneumatic transfer system was minimal during this period and any Ar-41 release would be insignificant since the system operates with CO_2 as the fill gas.

2.2.2 Tritium

Tritium release for the reactor pool is another gaseous release. The evaporation rate of the reactor pool was checked recently by measuring the loss of water from a flat plastic dish floating in the pool. The dish had a surface area of 0.38 ft^2 and showed a loss of 139.7 grams of water over a 71.9 hour period giving a loss rate of 5.11 g $ft^{-2}hr^{-1}$. Based on a pool area of about 395 ft^2 the annual evaporation rate would be 4,680 gallons. This is of course dependent upon relative humidity, temperature of the air and water, air movement, etc. For a pool ^3H concentration of 25,720 pCi/l (the average for July 1995 to June 1996) the tritium activity released from the ventilation system would be 455 μCi. A dilution factor of 2 x 10^8ml s^{-1} was used to calculate the unrestricted area concentration. This is from 200 m^2 (cross-section of the building) times 1 m s^{-1} (wind velocity). These are the values used in the safety analysis in the reactor license. A sample of air conditioner condensate showed no detectable ^3H. Thus, there is probably very little ^3H recycled into the pool by way of the air conditioner condensate and all evaporation can be assumed to be released.

^3H released	455 μCi
Average concentration, unrestricted area	7.2 x 10^{-14}μCi/ml
Permissible concentration, unrestricted area	1 x 10^{-7}μCi/ml
Percentage of permissible concentration	7.2 x 10^{-5}%
Calculated effective dose, unrestricted area	3.6 x 10^{-5} mrem

During August of 1996, approximately 1.1 gallon of D_2O was lost from the D_2O tank's expansion tank into the reactor pool. This increased the tritium concentration level to 74,000 pCi/l. Projecting this average for the July 1996 to June 1997 year, would increase the above calculated effective dose by a factor of approximately 2.9.

2.2.3 Nitrogen-16

During 1 MW(th) steady-state operation, the gamma level directly above the reactor core near the pool surface is 25 mR/hr. Just over the edge of the pool wall, the maximum gamma level is 15 mR/hr. Neither of these areas are ones where personnel spend much time. The radiation levels on the reactor bay floor areas at shoulder level range from 0.2 to 1.5 mR/hr. Control room radiation levels are much less than 1 mR/hr.

The main source of the radiation in the reactor bay is the production of radioactive Nitrogen-16 from the action of fast neutrons on the oxygen in the pool water. The problem is minimized by a diffuser pump that prolongs the amount of time for the nitrogen-16 to reach the surface, thus allowing for decay of much of the nitrogen-16 (7.14 second half-life). This short half-life also makes the release of nitrogen-16 to the environment negligible.

2.2.4 Radioactive Waste

Radioactive material produced by the PSBR comes under the control of the University Isotopes Committee and the broad byproduct material license number 37-185-4 upon removal from the reactor pool. This includes samples and activated components and equipment, excluding fuel elements. Radioactive waste from these materials is disposed of by the Health Physics office in the same manner as radioactive waste from other campus laboratories. The disposal techniques include shipment to licensed disposal facilities, storage of short-lived radioisotopes for decay, and release of low-concentration liquid and gaseous material. Volume reduction techniques include compaction, incineration, and evaporation.

The largest potential volume of liquid waste from the PSBR would be from the regeneration of the PSBR pool recirculation system demineralizer. This liquid would be collected along with waste from some floor drains and pump gland leakage in holdup tanks. Although the liquid could be released with little or no dilution, most of it would be evaporated and the distillate used as makeup water for the reactor pool. Presently, the demineralizer is not being regenerated and the resins are changed as necessary. The evaporator resins are solidified for disposal. Any solid dry waste in the evaporator would be compacted into 55 gallon drums and shipped to commercial burial sites for disposal.

2.3 ALARA

It is the policy of the University that environmental releases of radioactive material and exposure of individuals to ionizing radiation be kept as far below regulatory limits as is reasonably achievable (reference: Rules and Procedures for the Use of Radioactive Material at The Pennsylvania State University by The University Isotopes Committee- July 1980).

2.3.1 Personnel Training

PSBR administrative procedures call for all students, staff, faculty, or other users working independently in the PSBR facility to participate in a health physics orientation program. Participants must pass a written exam at the end of the program.

Before individuals can receive radioisotopes released from the reactor pool, they or their supervisor must possess an authorization issued by the University Isotopes Committee. Authorization approval requires the completion of the above-mentioned health physics orientation program. Authorized limits of the isotopes and their quantity depend upon the individual's qualifications and experience in handling radioactive materials as determined by the University Isotopes Committee.

Proper training of personnel in health physics procedures and licensing of individuals to possess radioisotopes are effective ways of minimizing personnel exposure within the facility and limiting the release of radioactive materials to the environment.

2.3.2 Administrative Policy

Where practical, steps are taken to minimize radiation levels in both restricted and unrestricted areas. Several years ago, a pneumatic transfer system used to transport samples between laboratories and the reactor core was converted from air to CO_2 as the working fluid to minimize Argon-41 production and release to the environment. Formerly a newly designed nozzle on the Nitrogen-16 diffuser pump caused a decrease in radiation levels due to Nitrogen-16 in the reactor bay. Floor drains have been clearly marked as to their destination to minimize chances of accidentally releasing radioactivity to the environment.

The Health Physics office uses appropriate methods to limit radiation workers internal and external exposure to radiation. Experimenters using loose radioactive material or performing neutron beam laboratory irradiations work under guidelines developed by the Health Physics office. Public areas and laboratories are monitored for transferrable contamination by periodic smear surveys. Monitoring for airborne activity is used if needed. Bioassays are used in appropriate situations. Radioisotopes Laboratory Rules are posted in all areas where radioactive materials are used.

Disposal of solid and liquid radioactive wastes are handled under the strict procedures of the Health Physics office to comply with 10 CFR 20 regulations.

2.4 Radiation Control

The radiation monitoring devices at the PSBR help to ensure compliance with the radiation limits in 10 CFR 20 and help to keep the radiation exposure of individuals in both restricted and unrestricted areas as low as reasonably achievable.

2.4.1 Environmental Radiation Monitoring

An environmental radiation monitoring program is conducted continuously by the Health Physics office. Integrated radiation measurements are made for successive 90-day periods using thermoluminescent dosimeters (TLD's).

2.4.2 Fixed Radiation Monitoring System

The principal PSBR fixed radiation monitoring system consists of six devices. Geiger-Mueller detectors on the east and west sides of the reactor bridge have alarms at 40 mR/hr and 200 mR/hr, respectively. Geiger-Mueller detectors are located in the reactor neutron beam laboratory and in the Cobalt facility, and both devices alarm at 6 mR/hr. Particulate air monitors are located along the east and west walls of the reactor bay. Both monitors use Geiger-Mueller detectors. Both monitors alarm at 10,000 CPM, a setting determined by 10 CFR 20 ALI considerations and detector efficiencies. All six devices provide readouts to the control room. A building evacuation alarm and reactor SCRAM occur when any monitor reaches its alarm point.

The reactor demineralizer room and hot cell clean and control areas have Geiger-Mueller detectors which provide local alarms at 5 mR/hr. The demineralizer room provides an alarm to the reactor console.

The lobby, lunch room, and various laboratories are equipped with rate meters with appropriate Geiger-Mueller probes.

All facility fixed monitors are checked for proper operation monthly and calibrated as a minimum annually.

2.4.3 Portable Survey Instruments

A variety of portable survey instruments are available for general radiation surveys and emergency situations. A typical inventory is listed below.

Instrument	Range	Location
Eberline RO-2-IC (4)	0-5 R/hr	Reactor Bay
Eberline E51OG-GM	0-1 R/hr	Cobalt Bay
Eberline E500B-GM	0-2 R/hr	Emergency Support Center
Eberline E520-GM	0-2 R/hr	Emergency Support Center
Eberline E520-GM	0-2 R/hr 0-250,000cpm	Emergency Support Center
Eberline E120(4)-GM	0-50 mR/hr 0-50,000cpm	Reactor Bay and Various Laboratories
Ludlum Model 3 (2)-GM	0-200 mR/hr 0-200,000 cpm	Various Laboratories
Eberline RO-20-IC	0-50 R/hr	Emergency Support Center

Instrument		Range	Location
Dosimeters	(4)	0-2 R/hr	Lobby and Reactor Bay
	(15)	0-200 mR/hr	Lobby
	(12)	0-200 mR/hr	Emergency Support Center
	(2)	0-5 R/hr	Emergency Support Center
	(2)	0-100 R/hr	Emergency Support Center
	(2)	0-1 R/hr	Emergency Support Center

These instruments are checked for proper operation monthly and calibrated as a minimum annually.

A variety of similar portable survey instruments are also available in the Academic Projects Building at the Health Physics office.

2.4.4 Personnel Monitoring

Faculty, staff, students, and other facility users are issued thermoluminescent dosimeters by the Health Physics office to monitor their radiation exposure. Other facility visitors such as public tours, are issued pocket dosimeters.

3.0 Enviromental Effects of Accidents

Accidents ranging from failure of experiments to core damage and fission product release are considered in the PSBR Safety Analysis Report. Effects are considered negligible with respect to the environment.

4.0 Unavoidable Effects of Facility Construction and Operation

The unavoidable effects of construction and operation of the facility involve the materials used in construction that cannot be recovered and the fissionable material used in the reactor. No adverse impact on the environment is expected from either of the unavoidable effects.

5.0 Alternatives to Construction and Operation of the Facility

Some of the activities conducted at the PSBR could be done by using particle accelerators, radioactive sources, or other means; but these alternatives are more costly and less efficient. Much of the research and educational activities using the PSBR can not be done by other suitable or economic means. Therefore, there is no reasonable alternative to a research reactor such as the PSBR for conducting the wide variety of research and education activities presently done.

6.0 Long-Term Effects, Costs and Benefits, and Alternatives of Facility Construction and Operation

Since the facility is an existing one, the capital costs are low. Environmental impact due to the facility's operation is minimal, and no long-term environmental damage is anticipated.

Beneficial long-term effects are numerous and include (1) the education of students, the public, and power plant personnel; and (2) research activities, including neutron radiography, activation analysis, and isotope production, which serve the national interest in areas of health, nuclear power production, national defense, and many other areas.

No reasonable alternatives exist to the wide versatility of research reactors such as the PSBR in contributing to education and scientific knowledge.

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