TECHNICAL SPECIFICATIONS AND BASES FOR THE WASHINGTON STATE UNIVERSITY MODIFIED TRIGA REACTOR

This document constitutes the Technical Specifications for Facility License No. R-76 and supersedes all prior Technical Specifications. Included in these Technical Specifications are the "Bases" to support the selection and significance of the specification. These bases are included for informational purposes only and are not part of the Technical Specifications. The bases do not constitute limitations or requirements to which the licensee shall adhere.

1 DEFINITIONS

The following frequently used terms are listed to provide uniform interpretation of terms and phrases used in the Technical Specifications.

30/20 Fuel: 30/20 fuel is TRIGA fuel that contains a nominal 30 weight percent of uranium with a ^{235}U enrichment of less than 20% and erbium, a burnable poison.

Annual: Annual shall mean a time interval of 12 months, not to exceed 15 months.

Audit: An audit is a quantitative examination of records, procedures or other documents after implementation from which appropriate recommendations are made.

Biennial: Biennial shall mean a time interval of 24 months, not to exceed 30 months.

BNC Design Modification: The term BNC Design Modification as applied to the BNC Facility beam refers:

(1) to a change that is shown to alter the dose vs. depth profile of the fast neutron, thermal neutron or gamma rays in the beam as sensed by the calibration check and

(2) to a change that has the potential to increase the amount of activation products in the BNC Facility.

BNC Experiment: BNC experiment shall be defined as a boron neutron capture experiment which utilizes the BNC facility, including neutron irradiation of biological cells that have been enriched with boron.

BNC Facility: BNC facility shall mean the boron neutron capture facility that includes the BNC neutron beam, bridge moving system, beam monitoring equipment, beam shielding room, access door and experimental area viewing equipment. Experimental bench(s), experiment positioning equipment and other equipment used for neutron beam targets shall not be considered part of the BNC Facility for purposes of this definition except insofar as radiation safety (i.e., activation and/or contamination) is concerned.

BNC Neutron Beam Calibration: The term BNC neutron beam calibration shall be defined as the process of measuring the intensity and energy spectrum of a BNC neutron beam for the purpose of conducting a BNC Experiment.

BNC Radiation Fluence: The term "radiation fluence" means the total fluence of neutrons and gamma radiation that is emitted by the BNC facility beam. The determination of the ratio of gamma, fast neutron and thermal neutron fluences is part of the beam characterization, allowing the total radiation fluence to be monitored by the online detectors, which are neutron sensitive. Compliance with the limits specified for the radiation fluence is determined by reference to the fluence monitored by these detectors.

BNC Retracted Position: The retracted position is any position of the research reactor greater than 4 feet from the BNC beam port.

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

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

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior, or by comparison of the channel with other independent channels or systems measuring the same parameter.

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

Cold Critical: The reactor is in the cold critical condition when it is critical ($k_{eff} = 1$) with the fuel and pool water temperature both at ambient temperature.

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

Control Rod: A control rod is a device fabricated from neutron-absorbing material that is used to initiate neutron flux changes and to compensate for reactivity changes. A control rod can be coupled to its drive unit allowing it to perform a safety function when the coupling is disengaged. All such reactor control devices for the WSU reactor are referred to as control rods, irrespective of the specific geometry of the devices. The following types of control rods are in use:

(1) Regulating Rod: The regulating rod is a low-worth (low reactivity) control rod fabricated from stainless steel, and is used primarily to maintain an intended power level and does not have scram capability. Its position is varied by means of an electric motor-operated positioning system. The electric motor-operated positioning system moves the control rod into or out of the reactor core in response to a signal initiated by the reactor operator when the console mode selector switch is set in the manual or auto position or in response to a signal generated within the control console when the console mode switch is set to the auto position.

(2) Transient Rod: The transient rod is a control rod that has a scram capability and is capable of providing rapid reactivity insertion to produce a pulse. The transient rod is positioned by controlled movement of a pneumatic cylinder that moves the cylinder and control rod together when the console mode selector switch is in the manual or auto position and air pressure is applied, or the control rod can be rapidly moved by application of air pressure to move the control rod drive within the pneumatic cylinder when the console mode selector switch is in the pulse mode.

(3) Standard Control Rod: Standard control rod shall mean any control rod which has a scram capability, which is utilized to vary the reactivity of the core, and which is positioned by means of an electric motor-operated positioning system. The electric motor-operated positioning system moves the control rod into or out of the reactor core in response to a signal initiated by the reactor operator when the console mode selector switch is set to the manual or auto position.

Core Configuration: The core configuration includes the number, type, or arrangement of fuel rods, reflector elements and regulating, transient, or standard control rods occupying the core grid.

Core Lattice Position: Core lattice position refers to specific locations in the WSU reactor core. The core lattice positions are denoted by a letter-number sequence with the letters A through G and the numbers one through nine, where the letters denote rows and the numbers denote columns. Each letter-number sequence may be followed by a directional indicator, NE, SE, SW or NW, which are compass directional indicators denoting a particular quadrant in a core lattice position.

Excess Reactivity: Excess reactivity is the amount of reactivity which would be added if all reactivity control devices were moved to the maximum reactive condition from the point at which the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions or at a specified set of conditions.

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

(1) Movable experiment: A movable experiment is one in which it is intended that all or part of the experiment can be moved in or near the core or into and out of the reactor while the reactor is operating.

(2) Secured Experiment: A secured experiment is any experiment, experimental apparatus, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means or by gravity and is not readily removable from the reactor. The restraining forces shall be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the

operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

Fuel Assembly: A fuel assembly is a cluster of three or four fuel rods fastened together in a square array by a top handle and bottom grid plate adapter. A fuel assembly is also sometimes referred to as a fuel bundle.

Fuel Rod: A fuel rod is a single TRIGA-type fuel rod of either Standard TRIGA or 30/20 TRIGA fuel.

Irradiation Facilities: Any in-pool experimental facility which is not a normal part of the core and which is used to irradiate devices and materials.

Instrumented Fuel Rod: An instrumented fuel rod is a fuel rod in which thermocouples have been embedded for the purpose of measuring the fuel temperature during reactor operation. The Instrumented Fuel Rod or Instrumented Fuel Element is sometimes referred to by the acronyms "IFR" or "IFE."

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

Licensed Area: The licensed area is the part of the Dodgen Research Facility building which is subject to the requirements of the WSU license R-76. This area includes:

(1) the reactor pool room (also known as Room 201) and adjacent rooms that allow direct unrestricted access to the pool room

(2) the beam room (also known as Room 2) and adjacent rooms that allow direct unrestricted access to the beam room.

Licensee: An individual or organization holding a license.

Limiting Safety Systems Setting: Limiting safety systems settings are the settings for automatic protective devices related to those variables having significant safety functions.

Measured Value: The measured value is the value of a parameter as it appears on the output of a measuring channel.

Measuring Channel: A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a parameter.

Mixed Core: A mixed core is a core arrangement containing Standard and 30/20 TRIGA fuels.

Monthly: Monthly shall mean a time interval of 30 days, not to exceed 45 days.

Off-site: Off-site shall mean any location which is outside the site boundary.

On-site: On-site shall mean any location which is within the site boundary.

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

Operability Test: An operability test is a test to determine whether a component or system is capable of performing its intended function.

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

Operational Core: An operational core is any arrangement of TRIGA fuel that is capable of operating within the maximum licensed power level and that satisfies all the requirements of the Technical Specifications.

Pool Room Ventilation System: The pool room ventilation system is the combination of fans, dampers, filters, ductwork and controls that provides controlled movement of air into and out of the reactor pool room.

Protective Action: Protective action is the initiation of a signal or the operation of equipment within the reactor safety system in response to a parameter or condition of the reactor facility having reached a specified limit.

Pulse Mode: Pulse mode operation shall mean operation of the reactor with the mode selector switch in the pulse position.

Quarterly: Quarterly shall mean a time interval of three months, not to exceed four months.

Reactor Bridge: The bridge is the structure that spans the research reactor pool to provide a support structure from which the research reactor is suspended.

Reactivity Worth of an Experiment: The reactivity worth of an experiment is the value of the reactivity change which results from the experiment being inserted into or removed from its position.

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

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

Reactor Safety Systems: 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.

Reactor Secured: The reactor is secured when:

Either

(1) There is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality under optimum available conditions of moderation and reflection;

Or

(2) The following conditions exist:

(a) The reactor is shut down;

(b) All of the control rods are fully inserted;

(c) The console key switch is in the "off" position and the key is removed from the console lock;

(d) 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;

(e) No experiments are being moved or serviced that have a reactivity worth equal to or greater than $1.00.

Reactor Shutdown: The reactor is shut down if it is subcritical by at least $1.00 in the reference core condition with the reactivity worth of all installed experiments included.

Reference Core Condition: The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is less than $0.30.

Reportable Occurrence: Any of the following events is a reportable occurrence:

(1) violation of a safety limit;

(2) release of fission products into the environment;

(3) release radioactivity from the site above limits established by 10 CFR 20 or specified within these Technical Specifications;

(4) operation with actual safety system settings for required systems less conservative than specified in the Technical Specifications;

(5) operation in violation of a Limiting Condition of Operation listed in Section 3;

(6) operation with a required reactor or experiment safety system component in an inoperative or failed condition which renders or could render the system incapable of performing its intended safety function;

(7) an unanticipated or uncontrolled change in reactivity greater than $1.00;

(8) abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary where appropriate;

(9) an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

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

Research Reactor Facility: The research reactor facility includes all areas within which the owner or operator directs authorized activities associated with the reactor.

Review: A review is a qualitative examination and evaluation of records, procedures or other documents prior to implementation from which appropriate recommendations are made.

Safety Channel: A safety channel is a measuring channel in the reactor safety system.

Safety Limits: Safety limits are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity.

Scram time: Scram time is the elapsed time between the initiation of a scram signal and complete insertion of the slowest control rod.

Semi-Annual: Semi-annual shall mean a time interval of six months, not to exceed 7.5 months.

Senior Reactor Operator: A senior reactor operator is an individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

Shall, Should, and May: The word “shall” is used to denote a requirement; the word “should” is used to denote a recommendation; and the word “may” is used to denote permission, neither a requirement nor a recommendation.

Shutdown Margin: Shutdown margin 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 conditions with the most reactive control rod and the non-scrammable rod in the most reactive position and that the reactor will remain subcritical without further reactor operator action.

SRO: The term “SRO” is an acronym for Senior Reactor Operator.

Standard Fuel: Standard fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of less than 20%.

Steady-State Mode: Steady-state mode shall mean any operation of the reactor with the mode selector switch in the manual or auto position.

True Value: The true value is the actual value of a parameter.

Unrestricted Area: Unrestricted area shall mean any location that is off-site.

Unscheduled Shutdown: An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operations, not including shutdowns that occur during testing or checkout operations.

Vacant Core Position: A vacant core position is a core grid position which does not have a fuel assembly, reflector, or experimental apparatus installed in the grid position.

Weekly: Weekly shall mean a time interval of 7 days, not to exceed 10 days.

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Fuel Rod Temperature

Applicability: This specification applies to the temperature of the reactor fuel.

Objective: The objective is to define the maximum fuel temperature that can be permitted with confidence that a fuel cladding failure will not occur.

Specifications:

(1) The maximum temperature in a Standard TRIGA fuel rod shall not exceed 1000 °C under any condition of operation.

(2) The maximum temperature in a 30/20 TRIGA fuel rod shall not exceed 1150 °C under any condition of operation.

Basis: The important parameter for a TRIGA reactor is the fuel rod temperature. This parameter is well-suited as a single specification, especially since it can be measured. A loss in the integrity of the fuel rod cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding if the fuel temperature exceeds the safety limit. 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 and the ratio of hydrogen to zirconium in the alloy.

Specification (1) provides a safety limit for the Standard TRIGA fuel 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 because of hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided that the temperature of the fuel does not exceed 1000 °C and the fuel cladding is water cooled.

Specification (2) provides a safety limit for the 30/20 LEU fuel based on data that indicate the stress in the cladding due to the hydrogen pressure from the disassociation of zirconium hydride will remain below the ultimate stress, provided the temperature of the fuel does not exceed 1150 °C and the fuel cladding is water cooled.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the settings which prevent the safety limit from being reached or exceeded.

Objective: The objective is to prevent the safety limits from being reached or exceeded.

Specifications: The limiting safety system settings shall be 500 °C or less, as measured in an instrumented fuel rod located in the central region of the core. The instrumented fuel rod shall be located in one of the following core lattice positions in the region of the core containing the

30/20 fuel rods: D2NE, D2SE, C3 (except for C3NE), D3, E3 (except for E3SE), C4, E4NE, E4NW, C5 (except for C5SW), D5SE, E5NE, E5NW, C6NW, or D6.

Basis: The limiting safety system setting is the measured instrumented fuel rod temperature that, if exceeded, shall initiate a scram to prevent the fuel temperature safety limits from being exceeded. For both the hottest and coldest thermocouples, an IFE located in core positions D2NE, D2SE, C3 (except for C3NE), D3, E3 (except for E3SE), C4, E4NE, E4NW, C5 (except for C5SW), D5SE, E5NE, E5NW, C6NW, or D6 would protect the fuel temperature safety limit of 1150 °C for 30/20 LEU fuel at reactor power levels that are less than 1.7 MW and limit the maximum steady-state temperature in the 30/20 fuel region to less than 800 °C. This setting provides at least a 350 °C margin for 30/20 fuel and at least a 200 °C margin for Standard fuel. The analysis of acceptable locations for the Instrumented Fuel Rod can be found in the WSU Response to Request for Additional Information submitted to the U.S. NRC on August 22, 2008, under RAI 40, pages 3 – 9, U.S. NRC ADAMS Accession number ML082400522.

The same limiting safety system setting will apply in the pulse mode of operation. However, the temperature channel will not limit the peak power generated during the pulse because of the relatively long response time of the temperature channel as compared with the duration of a pulse. The temperature scram would limit the total amount of energy generated in a pulse by cutting off the "tail" of the energy transient in the event that the fuel temperature limit is exceeded. Thus, the fuel temperature scram provides an additional degree of safety in the pulse mode of operation to protect the fuel in the event of such conditions as sticking of the transient control rod in the withdrawn position after a pulse.

3 LIMITING CONDITIONS OF OPERATION

3.1 Reactor Core Parameters

3.1.1 Steady-State Operation

Applicability: This specification applies to the power generated in the reactor during steady-state operation.

Objective: The objective is to ensure that the fuel rod temperature safety limit will not be exceeded during steady-state operation.

Specifications: The reactor power level shall not exceed 1.0 MW during steady-state operation.

Basis: Thermal hydraulic calculations were carried out during the course of preparation for conversion of the WSU reactor from high enriched uranium to low enriched uranium fuel. Calculations demonstrate that the reactor will be adequately cooled with coolant inlet temperatures as high as 50 °C with reactor power as high as 1.3 MW. The minimum departure from nucleate boiling ratio is 1.69, the peak fuel temperature is 540 °C, and the maximum fuel rod wall temperature is 165 °C at 1.3 MW reactor power with a coolant inlet temperature of 50 °C. Documentation of the calculations can be found in the WSU Response to Request for Additional Information submitted to the U.S. NRC on August 22, 2008, under RAI 40, pages 1 - 2, U.S. NRC ADAMS Accession number ML082400522.

3.1.2 Pulse Mode Operation

Applicability: This specification applies to the peak fuel temperature in the reactor as a result of a pulse insertion of reactivity.

Objective: The objective is to ensure that fuel rod damage does not occur in any fuel rod during pulsing.

Specification: The maximum reactivity inserted during pulse mode operation shall be such that the peak fuel temperature in any fuel rod in the core does not exceed 830 °C.

Basis: TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for 30/20 fuel and 1.65 for standard fuel. This yields delta phase zirconium hydride which has a high creep strength and undergoes no phase changes at temperatures over 1000 °C. However, after extensive steady-state operation at 1 MW, the hydrogen will redistribute due to migration from the central high temperature regions of the fuel to the cooler outer regions. When the fuel is pulsed, the instantaneous temperature distribution is such that the highest values occur at the surface of the fuel rod and the lowest values occur at the center. The higher temperatures in the outer regions occur in fuel with a hydrogen to zirconium ratio that has substantially increased above the nominal value. This produces hydrogen gas pressures considerably in excess of that expected for $ZrH_{1.6}$. If the pulse insertion is such that the temperature of the fuel exceeds 874 °C, then the pressure will be sufficient to cause expansion of microscopic holes in the fuel that grow larger with each pulse. This expansion of the fuel stresses and distorts the fuel rod material which, in turn, can cause overall swelling and distortion of the cladding and entire fuel rod. The pulsing limit of 830 °C is obtained by examining the equilibrium hydrogen pressure of zirconium hydride as a function of temperature. The decrease in temperature from 874 °C to 830 °C reduces hydrogen pressure by a factor of two, which provides an acceptable margin. This phenomenon does not alter the steady-state safety limit since the total hydrogen in a fuel rod does not change. Thus, the pressure exerted on the clad will not be significantly affected by the distribution of hydrogen within the fuel rod during steady state operation.

3.1.3 Shutdown Margin

Applicability: These specifications apply to the reactivity condition of the reactor and the reactivity worth of control rods and experiments for all modes of operation.

Objective: The objective is to ensure that the reactor can be shut down from any condition of operation and to ensure that the fuel temperature safety limit will not be exceeded.

Specifications: The reactor shall not be operated unless the shutdown margin provided by control rods is $0.25 or greater with:

(1) all experiments with positive reactivity in the most reactive state;

(2) the value of all experiments with negative reactivity not used in the shutdown margin determination;

(3) the highest worth scrammable control rod and the non-scrammable control rod fully withdrawn;

(4) the reactor in the reference core condition.

Basis: The value of the shutdown margin ensures that the reactor can be shut down from any permissible operating condition with the highest worth scrammable control rod in the fully withdrawn position. Control rods that are not scrammable are not used to determine the shutdown margin. The WSU HEU to LEU startup report documents a total rod worth of $12.48 with a shutdown margin of $0.91 with the highest reactivity control rod, Number 4, and the regulating rod fully withdrawn. The startup report was submitted to the U.S. NRC on April 17, 2009, and may be found at the U.S. NRC ADAMS Accession number ML092290631.

3.1.4 Maximum Excess Reactivity

Applicability: This specification applies to the maximum excess reactivity which may be loaded into the reactor core.

Objective: The objective is to ensure that the core analyzed in the safety analysis report closely approximates the operational core.

Specifications: The maximum excess reactivity based on the reference core condition shall not exceed 5.6% $\Delta k/k$.

Basis: Although maintaining a minimum shutdown margin at all times ensures that the reactor can be shut down, specification 3.1.3 does not address the total reactivity available within the core. Specification 3.1.4 ensures that fuel temperatures are maintained within the safety limits. The specified excess reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility. The core excess reactivity was reported in the WSU HEU to LEU conversion startup report as $7.44, using the dollar convention. The startup report was submitted to the U.S. NRC on April 17, 2009, and may be found at the U.S. NRC ADAMS Accession number ML092290631.

3.1.5 Core Configuration Limitation

Applicability: This specification applies to a core of standard fuel and 30/20 fuel.

Objective: The objective is to ensure that the fuel temperature safety limit will not be exceeded as a result of power peaking effects.

Specifications:

(1) The 30/20 fueled region in a mixed core shall contain at least 51 30/20 fuel rods in a contiguous block of fuel in the central region of the reactor core. Water holes in the 30/20 region shall be limited to nonadjacent single fuel rod holes.

(2) Instrumented fuel elements shall be placed in the core grid positions specified in Section 2.2, Limiting Safety System Settings.

Basis: The limitation on the allowable core configuration of the 30/20 fuel limits power peaking effects. The limitation on power peaking effects ensures that the fuel temperature safety limit will not be exceeded in a mixed core.

A 500 °C limiting safety system setting and the allowed locations for the IFE limit the peak fuel temperature to less than 800 °C.

3.1.6 Fuel Parameters

Applicability: This specification applies to all fuel rods.

Objective: The objective is to maintain the integrity of the fuel element cladding.

Specifications: The reactor shall not be operated with damaged fuel rods, except for the purpose of identifying damaged fuel rods. A fuel rod shall be considered damaged if any of the following occur:

(1) the sagitta of transverse bend exceeds 0.125 in. over the length of the cladding;

(2) the length exceeds the original length by 0.125 in.;

(3) a cladding defect exists as indicated by release of fission products;

(4) visual inspection reveals bulges, gross pitting, or corrosion.

Basis: Gross failure or obvious visual deterioration of a fuel rod is sufficient to warrant declaration of the fuel as damaged. The elongation and bend limits are the values found acceptable to the U.S. NRC in NUREG-1537 Part 1, Rev. 0, 2/96, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Appendix 14.1, Format and Content of Technical Specifications for Non-Power Reactors, Section 3.1(6).

3.2 Reactor Control and Safety System

3.2.1 Control Rods

Applicability: This specification applies to the function of the control rods and to the time required for the scrammable control rods to be fully inserted from the time that a scram signal is initiated.

Objective: The objective is to ensure that the control rods are operable and can perform rapid shutdown of the reactor.

Specifications:

(1) The reactor shall not be operated unless the control rods are operable. Control rods shall not be considered operable if damage is apparent to the rod or rod drive assemblies, or the scram time exceeds 2 seconds.

(2) The scram time from the time that a scram signal is initiated to the time that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds.

Basis: This specification ensures that the reactor can be rapidly shut down when a scram signal is initiated. Experience and analysis have indicated that the specified scram time is adequate to ensure the safety of the reactor for the range of transients anticipated for a TRIGA reactor.

3.2.2. Reactor Measuring Channels

Applicability: This specification applies to the information that shall be available to the reactor operator during reactor operation.

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

Specifications: The reactor shall not be operated in the specified mode of operation unless the channels listed in Table 3.1 are operable.

Table 3.1 Required Operable Measuring Channels

Channel	Minimum Number Operable	Operating Mode	
		Steady State	Pulse
Fuel rod temperature	1	X	X
Linear power level	1	X	-
Log power level	1	X	-
Integrated pulse power	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 ensure that the reactor power level is adequately monitored for both steady-state and pulsing modes of operation. The specifications on reactor power level indication are included in this section since the power level is related to the fuel temperature. The power level scram is initiated by either the log power or linear power measuring channel; operation is only permitted if the log and any one linear power level channel is operable, as stipulated in Tables 3.1 and 3.2.

3.2.3 Reactor Safety System

Applicability: This specification applies to the reactor safety system channels and interlocks.

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

Specifications: The reactor shall not be operated unless the safety channels described in Table 3.2 and interlocks described in Table 3.3 are operable.

Table 3.2 Minimum Reactor Safety Channels

Safety Channel	Function	Number operable in specified mode: Steady State	Number operable in specified mode: Pulse
Fuel temperature	Scram if fuel temperature exceeds 500 °C	1	1
Power level	Scram if power level exceeds 125% of full licensed power	2	-
Manual scram	Manually initiated scram	1	1
High-voltage monitor	Scram on loss of high voltage to power channels	1	1
Preset timer	Transient rod scram 15 seconds or less after pulse	-	1
Pool level alarm	Alarm if primary coolant level drops more than 8 inches below normal	1	1
Bulk primary coolant temperature	Manual scram if primary coolant temperature reaches 50 °C	1	1

Table 3.3 Minimum Interlocks

Interlock	Function	Number operable in specified mode: Steady State	Number operable in specified mode: Pulse
1 kW pulse interlock	Prevent initiation of a pulse above 1 kW	-	1
Startup count rate inhibit on the log power level channel	Prevent control rod withdrawal when neutron count rate is less than 2 counts per second	1	-
Pulse-mode switch	Prevent withdrawal of standard control and regulation rods in pulse mode	-	1
Transient rod control	Prevent application of air pressure unless fully inserted	1	-
Control element selector	Prevent withdrawal of more than one control rod at a time	1	1

Basis: The fuel temperature and power level scrams provide protection to ensure that the reactor can be shut down before the safety limit on the fuel rod temperature will be reached. The manual scram allows the operator to shut down the reactor if an unsafe or abnormal condition occurs. In the event of failure of the high voltage power supply for the power measuring channels, operation of the reactor without adequate instrumentation is prevented. The preset timer ensures that the reactor power level will return to a low level after pulsing. A primary coolant drop of 8 inches, which triggers the pool level alarm at the control console, would leave more than 16 feet of water above the reactor. The alarm provides a notification to the reactor operator to take appropriate action. Thermal hydraulic analyses were carried out with a maximum bulk primary coolant temperature of 50 °C, and demonstrated that the reactor can be safely operated with temperatures up to 50 °C.

The interlock to prevent startup of the reactor with neutron count rate of less than 2 counts per second on the log power level channel ensures that sufficient neutrons are available for monitored and controlled startup. The interlock to prevent the initiation of a pulse above 1 kW is to ensure that the magnitude of the pulse will not cause the fuel rod temperature safety limits to be exceeded. The interlock to prevent withdrawal of the standard and regulating control rods in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The interlock to prevent withdrawal of more than one control rod at a time is to prevent an inadvertently large reactivity insertion. The transient rod control interlock is to prevent rapid insertion of reactivity when the reactor is in the steady state mode.

3.2.4 Pool Level Alarm

Applicability: This specification applies to the pool water level at which the pool level alarm activates.

Objective: The pool level alarm is intended to alert the operator to a significant decrease in the pool water level.

Specification: The pool level sensor shall initiate an alarm signal if the reactor pool level falls 8 inches or more below the normal level. The pool alarm sensor shall initiate a signal at the reactor control console and at a monitored remote location.

Basis: The water in the reactor pool provides shielding from radiation generated by the reactor. A coolant level drop of 8 inches below the normal level will not adversely impact shielding, but is indicative of a loss of pool water due to evaporation or a leak.

3.3 Primary Coolant Conditions

Applicability: This specification applies to the quality and quantity of the primary coolant in contact with the fuel cladding.

Objectives: The objectives of this specification are to:

(1) minimize the possibility for corrosion of the cladding on the fuel rods and to minimize neutron activation of dissolved materials;

(2) limit the upper temperature of the primary coolant for operation of the reactor;

(3) limit the radionuclide content of the primary coolant;

(4) maintain the appropriate pressure of the reactor coolant.

Specifications:

(1) Conductivity of the primary coolant shall be no higher than 5×10^{-6} mhos/cm.

(2) The pH of the primary coolant shall be between 5.0 and 7.5.

(3) The bulk primary coolant temperature shall not exceed 50 °C.

(4) The radionuclide content of the primary coolant shall not exceed 10 CFR 20 effluent release limits.

(5) The reactor shall not be operated with less than 16 feet of water above the top of the core.

Basis:

Specification (1) limits the primary coolant conductivity to limit the rate of corrosion that occurs in a water-metal system and to control the neutron activation of dissolved minerals in the primary coolant. A water purification system is used to control primary coolant conductivity to limit the corrosion rate and thereby extend the longevity and integrity of the fuel cladding and to minimize the radioactivity of neutron activation products, which is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposures during maintenance and operations. Experience with water quality control at many reactor facilities has shown that maintenance of water conductivity within the specified limits provides acceptable control.

Specification (2) limits the acceptable value for primary coolant pH to minimize corrosion rates of metal components. The water purification system will maintain the primary coolant pH in a near neutral range, although the coolant pH will tend to be slightly more acidic than neutral due to dissolution of atmospheric carbon dioxide in the primary coolant water. Experience with water quality control at many reactor facilities has shown that maintenance of water pH within the specified limits provides acceptable control.

Specification (3) limits the primary coolant temperature to 50 °C because safety analysis for conversion of the WSU reactor from HEU to LEU fuel was carried out for a maximum pool water temperature of 50 °C and found to provide an acceptable level of cooling for the reactor.

Specification (4) stipulates that the radioactive content of the reactor pool water shall remain below 10 CFR 20 release limits, which ensures that a pool water leak cannot under any condition exceed 10 CFR 20 effluent release limits. At this limit the entire pool could be emptied into the WSU sewage system without taking advantage of the dilution factor associated with the discharge volume of the WSU sewage system.

Specification (5) ensures that the appropriate pressure exists for the reactor coolant.

3.4 Ventilation System

Applicability: This specification applies to the operation of the facility ventilation system.

Objective: The objective is to ensure that the ventilation system is operable to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation.

Specifications:

(1) The reactor shall not be operated unless the facility ventilation system is operable and operating, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system. The ventilation system is operable when flow rates, dampers and fans are functioning normally. The normal, dilute and isolation modes shall be operable for the ventilation system to be considered operable.

   (a) The exhaust flow rate of the ventilation system in the normal mode, from the reactor pool room, shall be not less than 4000 cfm.

   (b) The exhaust flow rate of the ventilation system in the dilute mode, from the reactor pool room, shall be 300 cfm.

(2) The reactor pool room atmospheric pressure shall be maintained negative with respect to the areas outside the pool room when the ventilation system is in the normal or dilute mode.

(3) The ventilation system shall automatically switch to dilute mode upon a high activity alarm from the Continuous Air Monitor.

(4) The ventilation system shall be switched to the isolate mode upon initiation of a reactor scram.

(5) The dilute mode air filter shall be changed whenever the pressure drop across the filter increases by 1 in. of water above the initial level.

Basis: The ^{41}Ar gaseous radioactive effluent release under normal operations of the reactor facility is limited to 20 Ci of ^{41}Ar per year. The environmental monitoring program, as reported in annual reports to the U.S. NRC shows that there are no measurable exposures in the environment arising from operation of the reactor. Therefore, operating the reactor with the ventilation system in the dilute or isolate mode presents no increased risk to the public or the environment since, when in the dilute or isolate mode, the amount of effluent air released from the reactor pool room is less than when the ventilation system is operating normally. Operation of the reactor with the ventilation system shut down or in isolate mode for short periods of time to make system repairs or tests does not compromise the control over the release of airborne radioactive materials. Moreover, radiation monitors within the building, independent of the ventilation system, can give warning of high levels of radiation when the ventilation system is shut down or in isolate mode.

A high activity alarm from the Continuous Air Monitor would be the first indication of a loss of cladding integrity of a fuel rod. The high activity alarm from the Continuous Air Monitor causes the ventilation system to automatically switch into the dilute mode.

All reactor scrams trigger the ventilation system controls to shift the ventilation system into the isolate mode, irrespective of the cause of the reactor scram.

3.5 Radiation Monitoring System and Effluents

3.5.1 Radiation Monitoring Systems

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

Objective: The objective is to ensure that sufficient radiation monitoring information is available to the reactor operator to ensure safe operation of the reactor.

Specifications: The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.4 are operable. Each channel shall have a readout in the reactor control room and be capable of sounding an audible alarm that can be heard in the reactor control room.

Basis: The radiation monitors inform the reactor operator about danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings.

Table 3.4 Minimum Radiation Monitoring Channels

Channel*	Number
Reactor bridge radiation monitor	1
Beam room radiation monitor	1
Continuous air monitor	1
Exhaust gas monitor	1

*During maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma radiation sensitive instruments with alarms or that shall be kept under visual observation.

3.5.2 Effluents

Applicability: This specification applies to the concentration of ^{41}Ar in air effluent and to liquid effluents that may be discharged from the WSU TRIGA reactor facility.

Objective: The objective of this specification is to protect the health and safety of the public by limiting discharge of ^{41}Ar from the WSU research reactor facility and to limit the annual population radiation exposure due to operation of the WSU reactor.

Specifications:

(1) The concentration of ^{41}Ar in the effluent gas discharged from the facility into the unrestricted area, after environmental dilution shall not exceed 1×10^{-8} μCi/mL averaged over one year.

(2) An environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the facility.

(3) The annual radiation exposure due to reactor operation, at the closest off-site point of extended occupancy, shall not, on an annual basis, exceed the average local off-site background radiation by more than 20%.

(4) The total annual discharge of ^{41}Ar into the environment shall not exceed 20 Ci per year.

(5) The reactor shall be shut down if a fission product leak from a fuel rod or an airborne radioactive release from an irradiated sample is detected by the continuous air monitor, and the reactor shall remain shut down until the source of the leak is located and eliminated. However, the reactor may continue to be operated on a short-term basis as needed to assist with identification of the source of the leak provided that occupational values listed in Table 1 of 10 CFR 20 Appendix B are not exceeded and effluent concentrations listed in Table 2 of 10 CFR 20 are not exceeded.

(6) The quantity of radioactivity in liquid effluents released to the sewer system shall not exceed the limits stipulated in 10 CFR 20 Appendix B, Table 3.

Basis:

The maximum allowable concentration of ^{41}Ar in air released to unrestricted areas as specified in Appendix B, Table II of 10 CFR 20 is 1×10^{-8} μCi/mL.

The environmental monitoring program requirement is intended to provide data to measure the impact of reactor operations on the surrounding environment.

The maximum allowable annual radiation exposure to the public as stipulated in 10 CFR 20, Subpart D, 20.1301 due to reactor operations is 0.1 rem in one year. An increase of 20% over background radiation levels is less than the 0.1 rem per year limit. The environmental radiation monitoring program is used to determine average background radiation by monitoring background radiation levels in locations distant from the WSU reactor. The monitoring program is also used to measure radiation levels in close proximity to the nearest occupied dwelling. The closest off-site point of extended occupancy is used as a conservative baseline because it would be the point of greatest exposure.

Section 6.5 of the safety analysis report for conversion of the WSU TRIGA reactor to FLIP fuel establishes a 3.4×10^{-3} atmospheric dilution factor for an average 4.4 mph wind speed. Given a ventilation system exhaust discharge rate of 4000 cubic feet per minute, a dilution factor of 3.4×10^{-3} and a maximum discharge of 20 Ci/year, the maximum concentration of ^{41}Ar released into the unrestricted area would be 1.1×10^{-9} μCi/mL.

A fission product leak would first be detected by the continuous air monitor. The reactor may only be operated for purposes of finding the leak as long as 10 CFR 20 limits for occupational exposure and effluent concentrations are not exceeded.

Monthly average release limits for radioactive materials are provided in 10 CFR 20, Appendix B, Table 3. These values will be used for purposes of determining liquid effluent release limits.

3.6 Limitations on Experiments

Applicability: This specification applies to experiments installed in the research reactor and its experimental facilities (defined in Section 1).

Objective: The objective is to prevent damage to the reactor or 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:

(1) The reactivity worth of a moveable experiment shall be less than $1.00.

(2) The reactivity worth of a secured experiment shall not exceed \$2.00.

(3) The sum of the absolute values of all individual experiments shall not exceed \$5.00.

(4) Explosive materials, such as TNT, or its equivalent, shall be limited to 25 mg, for irradiation in the reactor or experimental facilities. Explosive materials in quantities less than 25 mg may be irradiated in the reactor or experimental facilities, provided prior testing of explosive material encapsulation is shown to ensure no reactor damage in the event of a detonation and that the pressure produced upon detonation of the explosive has been demonstrated to be less than half the design pressure of the container.

(5) Experimental materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under:

(a) normal operating conditions of the experiment or reactor;

(b) credible accident conditions in the reactor;

(c) possible accident conditions in the experiment;

shall be limited in radioactivity so that if 100% of the gaseous radioactivity or radioactive aerosols produced escaped to the reactor room or the atmosphere of the unrestricted area outside the facility, the airborne concentration of radioactivity would not exceed the limits of 10 CFR 20, Appendix B, Table 1 or Table 2 averaged over one year. An atmospheric dilution factor of 3.4×10^{-3} for gaseous discharges from the facility shall be used in calculations of unrestricted area effluent discharges.

(6) Pursuant to specification (5) above, the following conditions shall be shown to exist:

(a) at least 90% of the particles will be retained if the effluent from an experiment is designed to exhaust through a filter installation designed for greater than 99% filtration efficiency for 0.3 micrometer particles;

(b) at least 90% of the vapors will be retained in the experiment or in the reactor pool for materials whose boiling point is above 60 °C and the materials are exposed to conditions in which the material can boil, and vapors formed by boiling this material can escape only through an undisturbed column of water above the core.

(7) Each fueled experiment shall be controlled so that the total radioactive inventory of iodine isotopes 131 through 135 in the experiment is less than 1.5 Ci.

(8) The experimental material and potentially damaged components shall be inspected to determine the consequences and need for corrective action if a capsule fails and releases material that could damage the reactor fuel or structure by corrosion or other means.

(9) Corrosive materials shall be doubly encapsulated. All liquid and gas samples shall be analyzed to determine whether they require double encapsulation.

Basis:

Specification (1) limits the worth of moveable experiments to less than $1.00 to provide assurance that the worth of a single moveable experiment will be limited to such a value that the reactor will not reach a state of prompt criticality if the positive worth of the experiment were to be suddenly inserted.

Specification (2) limits the maximum worth of a secured experiment to $2.00 so that the sudden addition of $2.00 of positive reactivity to the reference core will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. It was shown in the WSU HEU to LEU conversion Safety Analysis Report that a $2.00 pulse from any permissible power level will not exceed the fuel temperature safety limit.

Specification (3) limits the sum of the absolute values of reactivity worth of all experiments to ensure that the reactor will remain subcritical when in a shutdown condition in the event of a simultaneous removal of all of the experiments with one control rod and the regulatory rod withdrawn. The minimum required shutdown margin is $0.25 with all experiments with positive reactivity in the most reactive state, and the value of all experiments with negative reactivity not used in the shutdown margin determination and the non-scrammable regulating rod and the highest worth control rod in the fully withdraw position. The value of experiments with positive reactivity is included in the shutdown margin calculation to ensure that insertion of experiments with positive reactivity cannot cause the reactor to become critical. Excluding the value of experiments with negative reactivity assures that the reactor cannot become critical with simultaneous withdrawal of all experiments with negative reactivity.

Specification (4) limiting the use of explosive materials is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials. NUREG 1537 Part 1, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Appendix 14.1, Format and Content of Technical Specifications for Non-Power Reactors, Section 3.8.2 describes a limiting condition for irradiation of explosive materials, i.e. a limit of 25 mg, and that prior testing of the explosive material encapsulation ensures that no reactor damage would result in the event of a detonation.

Specification (5) is intended to reduce the likelihood that a release of airborne radioactive material in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere within or outside the facility. The atmospheric dilution factor of 4×10^{-3} for an elevated release has been documented in the Safety Analysis Report of May, 1974 for conversion of the WSU reactor to HEU fuel, with the Technical Specifications approved by the U.S. NRC in a letter dated June 26, 1975. The atmospheric dilution factor calculation was also demonstrated in the WSU Safety Analysis Report of 1979 which was approved by the U.S. NRC as Amendment No. 10 in a letter from the U.S. NRC in a letter dated August 11, 1982.

Specification (6)(a) applies to experiments which are set up to exhaust through a filter system. The specification is intended to prevent particulate matter from escaping if the experiment is set up such that any material released from the experiment shall first pass through a filter before reaching the pool room atmosphere or the unrestricted area outside the facility.

Specification (6)(b) applies only to materials with a boiling point that is greater than 60 °C that are irradiated under conditions that cause the sample to exceed the boiling point.

Specification (7) provides the 1.5-Ci limitation on iodine isotopes 131 through 135 to ensure that in the event of failure of a fueled experiment leading to total release of the iodine, the dose at the exclusion area boundary will be less than that allowed by 10 CFR 20 for an unrestricted area. The most restrictive limit for an iodine isotope is 2×10^{-10} μCi/mL for ^{131}I, averaged over one year. At a ventilation system flow rate of 4000 ft^3/minute, 1.5 Ci of ^{131}I would result in an air effluent release of 8.6×10^{-11} μCi/mL, averaged over one year, which is well below the 10 CFR 20 Appendix B Table 2 release limit.

Specification (8) stipulates inspection in case of a capsule failure because operation of the reactor with the reactor fuel or structure damaged is prohibited.

Specification (9) provides for double encapsulation of corrosive or potentially hazardous liquid or gas samples to reduce the likelihood and consequences of an encapsulation failure.

3.7 Sealed Sources in the Reactor Pool

Applicability: This specification applies to sealed radioactive sources stored or used in the reactor pool.

Objective: The objectives of this requirement are to ensure that:

(1) sealed radioactive sources that are stored or used in the pool do not constitute a hazard to the reactor;

(2) sealed radioactive sources do not create an environmental or occupational hazard;

(3) sealed radioactive sources do not compromise the ALARA criteria of the facility.

Specifications:

(1) Sealed sources shall not be stored or used closer than five (5) feet from an operating reactor core.

(2) The total radioactivity of all sealed sources stored in the reactor pool shall not exceed 100,000 curies.

(3) All sealed source configurations shall be designed so that a loss of pool water accident shall not cause a loss of sealed source encapsulation integrity and the sources shall be stored in an appropriate shield to prevent a significant radiation hazard in the event of a loss of reactor pool water.

(4) The storage and use of sealed sources shall be considered an experiment subject to all applicable provisions for experiments in the Technical Specifications.

(5) A written Standard Operating Procedure for the storage and use of sealed sources in the reactor pool shall be in effect.

Basis:

The 10 CFR 20 Appendix B, Table 3 limit is 3×10^{-5} µCi/mL for ^{60}Co, which is the WSU sealed source material. Limits of the pool water radionuclide content are provided in the Limiting Conditions of Operation in Section 3.3 for Primary Coolant Conditions and the Surveillance Requirement in Section 4.3 for Primary Coolant Conditions.

Specification (1) limits the proximity of sealed sources to five or more feet away from the surface of the reactor core which minimizes the effect of radioactive sources on the reactor and the operation of the reactor upon the sources. The neutron flux at a distance of five feet from the core surface is insignificant and thus could not cause activation of the sources and associated shielding.

Specification (2) limits the level of radioactivity of sealed sources to a value at which the presence of the sources in the pool would have no impact upon the maximum hypothetical accident, which is the rupture of the cladding of one fuel rod in air. However, the presence of sources in the pool could contribute to the radiation hazard associated with a loss of pool water. The dose rate 25 feet above an unshielded core in the event of a loss of coolant accident would be increased by less than 2% with the presence of 100,000 curies of ^{60}Co stored in the irradiation unit in the reactor pool.

Specification (3) requires shielding of sealed sources to be provided in a manner which` will limit the radiation exposure in the reactor pool room in the event of a loss of coolant accident.

Specifications (4) and (5) provide limitations on the storage and use of sealed sources to ensure that the sealed sources are used safely. Classifying the storage of sealed sources as an experiment mandates that the storage be reviewed by the Reactor Safeguards Committee.

3.8 Boron Neutron Capture Facility

Applicability: This specification applies solely to the generation of a BNC neutron beam for BNC experiments.

Objective: The objective of the Technical Specifications in this section is to provide assurance that use of the BNC facility and a BNC neutron beam does not present a danger to any person, the reactor, or the reactor facility.

Specifications:

(1) It shall be possible to initiate a scram of the reactor from a control panel located in the BNC facility area. In the event that the BNC facility scram is inoperable, it shall be acceptable to use one of the control room scrams via communication with the reactor

operator as a temporary means of satisfying this provision. Use of this temporary provision is limited to seven consecutive working days.

(2) Access to the BNC facility shall be controlled by means of a single access door located at the entrance to the BNC facility.

(3) The following BNC facility features and controls shall be operable:

(a) The reactor bridge shall automatically move to the retracted position upon initiation of a reactor scram.

(b) A BNC experiment may not be conducted if the BNC room radiation monitor reading exceeds 50 mrem/hr 30 seconds or longer after the initiation of the scram and bridge retraction.

(c) An interlock shall prevent movement of the reactor bridge from the retracted position unless the BNC facility access gate is closed.

(d) The reactor shall scram and the reactor bridge shall move to the retracted position automatically upon opening the BNC facility room access door.

(e) The reactor bridge shall move to the retracted position automatically upon failure of BNC facility electric power or low voltage on backup batteries that are used to power the reactor bridge movement motor.

(f) There shall be a means to manually move, without the use of electric power, the reactor bridge to the retracted position.

(g) It shall be possible to initiate a signal from within the BNC facility to cause the reactor bridge to move to the retracted position.

(h) A key operated switch near the BNC facility access door shall prevent reactor control rod withdrawal when the key is not inserted and turned to the locked position.

(4) The reactor bridge shall be equipped with a sensor that shows the position of the reactor bridge. The BNC facility control panel shall include a reactor bridge position indicator.

(5) It shall be acceptable to use an alternate means of verifying reactor bridge position, such as a video camera in the reactor pool room providing a signal to a video monitor at the control panel of the BNC facility in the event of a bridge position readout malfunction at the BNC facility control panel. Use of an alternate means of bridge position verification is limited to seven consecutive working days.

(6) The BNC facility shall be equipped with a display of the reactor linear power and log power channels on a BNC facility control panel.

(7) The BNC facility shall be equipped with a display that provides an indication of the radiation level within the facility that indicates both within the facility and at the local control panel and provides an audible alarm within the facility and at the BNC facility control panel.

(a) The radiation monitor shall be equipped with a backup power supply.

(b) The radiation monitor audible alarm shall be set at or below 50 mrem/hr. The monitor and/or its alarm may be disabled once the BNC beam room has been searched and secured by closing and locking the BNC facility access door. If the radiation monitor and/or the audible alarm are disabled, both the monitor and the audible alarm shall automatically become functional upon opening of the BNC facility access door.

(c) Personnel entering the BNC facility shall use portable radiation detection instruments and audible alarm personal dosimeters if the radiation monitor becomes inoperable during use of the BNC facility. Use of portable radiation detection instruments and audible alarm personal dosimeters as a temporary means of satisfying this provision shall be limited to seven consecutive working days.

(8) An intercom or other means of two-way communication shall be operable between the BNC facility control panel and the reactor control room, and also between the BNC facility control panel and the interior of the BNC facility radiation shielding.

(9) It shall be possible for personnel monitoring a BNC experiment to open the BNC facility access door.

(10) It shall be possible to observe the BNC experiment by means of two independent closed-circuit television (CCTV) cameras.

(a) Each camera shall be operable at the beginning of a BNC experiment. A BNC experiment may be continued at the discretion of the experimenter if one camera fails during a BNC experiment. The BNC experiment shall be immediately stopped if both cameras fail during a BNC experiment.

(b) Emergency lighting and backup power shall be provided for one BNC facility CCTV camera.

(11) Maintenance, repair, and modification of the BNC facility shall be performed under the supervision of a Senior Reactor Operator. All modifications shall be reviewed pursuant to the requirements of 10 CFR 50.59.

(12) Personnel who are not licensed to operate the WSU research reactor may operate the controls for the BNC facility provided compliance is maintained with all technical specifications and that:

(a) instructions have been posted at the BNC facility control panel to ensure that only the appropriate target is in the irradiation facility before turning on the primary beam of radiation to begin an irradiation;

(b) training has been provided, proficiency satisfactorily demonstrated and documented on the design of the facility, the controls, and the use of the controls;

(c) the procedure for conduct of the BNC experiment shall be posted at the control panel of the BNC facility with instructions to notify the reactor operator if the BNC facility operator is unable to turn the BNC neutron beam off with BNC facility controls, or if any abnormal condition occurs; a directive shall be included with this procedure to notify the reactor console operator if an abnormality occurs;

(d) personnel who are not licensed on the WSU research reactor but who have been trained under this provision may initiate bridge movement provided that verbal permission is requested and received from the reactor console operator immediately prior to such action. Emergency scrams causing a bridge retraction are an exception and may be made without first requesting permission.

(13) Personnel who are not licensed to operate the WSU research reactor shall not take any action that affects the reactivity of the research reactor without approval of a senior reactor operator.

(14) The following characterizations of the BNC neutron beam shall be carried out to prepare the BNC facility for a BNC experiment:

(a) the intensity of the beam shall be measured;

(b) the neutron energy spectrum shall be determined;

(c) the beam diameter and divergence shall be determined;

(d) the dose vs. depth profile in phantoms, evaluated from the surface of the phantom to a depth at least equivalent to the total thickness of the BNC experimental target as irradiated on a central axis. Thermal and fast neutrons and gamma ray components shall be determined in this characterization.

<u>Basis</u>:

Specification (1) provides a requirement to have a capability to initiate a scram from a control panel located in the BNC facility area to quickly shut down the reactor in case of an emergency.

Specification (2) requires that access to the BNC facility be limited to a single door to control access and ensure that there will be no uncontrolled entry of personnel into the BNC neutron beam area.

Specifications (3)(a) and (3)(c) establish a requirement that the reactor bridge remain in or return to the retracted position upon initiation of a reactor scram, or when the BNC facility door is not closed to provide radiation protection to personnel who may be working in the vicinity of the BNC neutron beam port.

Specification (3)(b) establishes a radiation protection requirement to provide for the safety of operating personnel.

Specification (3)(d) establishes a requirement that the reactor shall scram and the reactor bridge shall move to the retracted position to provide for radiation protection of personnel.

Specification (3)(e) provides a requirement that the reactor bridge automatically retract upon loss of power to the BNC facility to provide radiation protection to personnel who may have to enter the BNC neutron beam area during a power outage to perform procedures to provide for safe operation and shutdown of the BNC facility.

Specification (3)(f) provides a requirement for a means to manually move the reactor bridge so that it is possible to move the reactor bridge to the retracted position if building power is lost.

Specification (3)(g) provides a requirement to initiate a signal that will cause the reactor bridge to move to the retracted position to control the amount of radiation entering the BNC facility through the BNC neutron beam port if an urgent need arises to quickly enter the beam area.

Specification (3)(h) puts in place a provision that would prevent startup of the reactor while personnel are working in the area of the BNC neutron beam port.

Specification (4) requires a reactor bridge position indicator to notify personnel of the beam status with respect to reactor proximity to the beam port.

Specification (5) provides for alternate means of positively verifying the reactor bridge position via video link, which provides a great degree of certainty with regard to information on the reactor position.

Specification (6) requires that the reactor power is displayed on the BNC facility control panel so that personnel performing work in the BNC facility will have accurate live-time information about reactor operational status so that work may be performed safely.

Specification (7) provides for radiation monitors and audible alarms to provide personnel with information on radiation levels within the BNC facility and to alert personnel to the presence of elevated radiation levels. The monitor and alarm may be disabled only after the facility has been searched, closed, and locked so that it will not act as a distraction to personnel operating the facility once BNC neutron beam use commences.

Specification (8) requires an intercom system to provide a means for prompt communication between BNC experimenter(s) and the reactor operator in the reactor control room.

Specification (9) establishes a requirement for operation of the access door of the BNC facility to ensure access to the experimental area under all circumstances.

Specification (10) places requirements on closed circuit television monitoring to provide the experimenter(s) with the capability to visually monitor the BNC target area. The emergency lighting and the television camera and monitor backup power will permit observation of the BNC target area if building electrical power is lost.

Specification (11) requires that a senior reactor operator supervise all maintenance, repair, or modification of the BNC facility to provide assurance that the BNC facility remains compliant with Technical Specifications and is operated safely.

Specification (12)(a) provides for posted instructions on conduct and use of the BNC facility to ensure that only the appropriate target is in the irradiation facility before turning the primary beam of radiation on to begin an irradiation.

Specification (12)(b) provides a training requirement.

Specification (12)(c) calls for posting of procedures to be followed in the event of the inability to turn off the BNC neutron beam or if an abnormality occurs.

Specification (12)(d) requires that movement of the reactor bridge shall first be approved by the reactor console operator, except in an emergency when it is necessary for the individual to initiate a reactor scram.

Specification (13) prohibits non-licensed personnel from carrying out any action that affects the reactivity of the reactor without Senior Reactor Operator approval because license conditions prohibit a non-licensed person from performing reactor operation actions without the supervision of a licensed reactor operator and Senior Reactor Operator oversight is required for use of the research reactor.

Specification (14) requires BNC neutron beam characterization prior to performing a BNC experiment to determine the parameters that are necessary to take into consideration for safe design and performance of the experiments.

4 SURVEILLANCE REQUIREMENTS

4.0 General

Applicability: This specification applies to the surveillance requirements of systems related to reactor safety.

Objective: The objective is to verify the proper operation of systems related to reactor safety.

Specifications: Additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the control rod drive mechanisms, or the reactor safety systems shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safeguards Committee. A system shall not be considered operable until after it has been successfully tested.

Basis: This specification relates to changes in reactor systems that could directly affect the safety of the reactor or the health and safety of personnel. As long as changes or replacements to these systems continue to meet the original design specifications, it can be assumed that they meet the presently accepted operating criteria.

4.1 Reactor Core Parameters

4.1.1 Steady State Operation

Applicability: This specification applies to the surveillance requirements for reactor power level.

Objective: The objective of this specification is to verify the operability of the reactor power level monitoring system to ensure that the maximum power level is not exceeded.

Specifications: The surveillance requirements for the reactor safety systems that monitor reactor power level are described in Section 4.2.3.

Basis: The power level information and control is provided by the power level monitoring system. This specification ensures the proper functioning of the reactor power level monitoring systems providing a means to ensure that the maximum reactor power level is not exceeded.

4.1.2 Pulse Mode Operation

Applicability: This specification applies to the authorized limits for fuel temperature during pulsing operation.

Objective: The objective of this surveillance requirement is to assure that the peak fuel temperature limit is not reached or surpassed during pulsing.

Specification:

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(1) The maximum safe allowable reactivity insertion shall be calculated annually for an existing core and prior to pulsing a new or modified core arrangement.

(2) The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with previous pulses of the same reactivity.

Basis:

Specification (1) requires an annual calculation for the maximum allowable reactivity insertion for pulsing because changing core configurations or fuel burnup could cause differences in core power peaking behavior, which shall be examined and accounted for by determining the safe allowable reactivity worth for pulsing operation.

Specification (2) requires semiannual pulsing to determine by generation of data whether changes in fuel or core characteristics have taken place.

4.1.3 Shutdown Margin

Applicability: This technical specification applies to the surveillance requirement for determination of reactor shutdown margin.

Objective: The objective of this specification is to verify that the reactor does not exceed the authorized limits for shutdown margin.

Specifications:

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(1) The reactivity worth of each control rod and the shutdown margin shall be determined annually.

(2) The reactivity worth of each control rod and the shutdown margin shall be determined after a change to the type or location of fuel, reflector, or control rods in the reference core, or after any change to the reference core that results in or could result in a change of reactivity of $0.25 or more.

(3) The reactivity worth of an experiment shall be estimated or measured, as appropriate, before reactor operation with the experiment.

Basis:

Specification (1) requires the reactivity worth of the control rods to be measured to ensure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worths of experiments inserted in the core. Experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis will detect changes in the shutdown margin.

Specification (2) is provided because core changes could also cause changes in shutdown margin. Therefore, it is necessary to measure the shutdown margin after core changes to confirm that the Technical Specification limits are not exceeded.

Specification (3) requires that the reactivity worth of experiments shall be determined to provide assurance that no single experiment or assemblage of experiments can exhibit sufficient reactivity to violate a technical specification limit on shutdown margin.

4.1.4 Maximum Excess Reactivity

Applicability: This specification applies to the surveillance requirement of reactor excess reactivity.

Objective: The objective is to verify that the reactor excess reactivity does not exceed the allowed excess reactivity of 5.6% $\Delta k/k$.

Specifications: The core excess reactivity shall be determined annually or following a change to the core that causes a change in reactivity greater than \$0.25. This surveillance requirement may be postponed during periods of reactor shutdown. If this surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

Basis: The core excess reactivity is limited to 5.6% $\Delta k/k$; annual determination of core excess reactivity will assure that the limiting condition of operation has not been violated.

4.1.5 Core Configuration Limitation

Applicability: This specification applies to a core consisting of standard and 30/20 fuel.

Objective: The objective is to verify that the fuel temperature safety limit will not be exceeded as a result of power peaking effects.

Specifications:

(1) Proposed changes in core configuration shall be analyzed to determine whether amendments to the reactor license or Technical Specifications are required.

(2) Changes in fuel configuration shall be documented by a Safety Analysis Report and recorded in a fuel inventory log.

(3) Each change to the core configuration shall be evaluated to determine the allowed locations for the Instrumented Fuel Element.

Basis:

Specification (1) and (2) require core configuration documentation in a safety analysis report for a proposed new core configuration as a means to present analysis to determine whether a license or Technical Specification amendment is required and to determine power peaking and heat transfer characteristics to provide a high degree of confidence that the new core configuration will not lead to a safety limit violation.

Specification (3) requires a determination of acceptable positions for an instrumented fuel element for a new core configuration. Examining the acceptable instrumented fuel rod locations will provide assurance that core behavior will not lead to a violation of the fuel temperature limiting safety system setting and safety limits.

4.1.6 Fuel Parameters

Applicability: This specification applies to all fuel rods.

Objective: The objective of this surveillance requirement is to verify that the reactor is not operated with damaged fuel rods.

Specifications:

(1) At least 20% of the fuel rods comprising the core shall be visually inspected annually for damage or deterioration and annually measured for bowing or elongation such that each fuel rod in the core is inspected at least once over a five year period.

(2) The failure of a single fuel element to pass inspection shall trigger a required inspection of all fuel elements in the reactor core.

Basis:

Specification (1) provides for an inspection frequency based upon time rather than the total worth of all reactor pulses, as was previously the case for the WSU research reactor. The increased inspection frequency will provide information of the condition of the research reactor fuel rods. The limit of transverse bend has been shown to result in no difficulty disassembling the core. The elongation limit has been specified to ensure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to ensure adequate coolant flow.

Specification (2) provides for a trigger event for inspection of all fuel rods if a single rod fails to pass inspection because of the need to provide a high degree of confidence that the remaining fuel can be safely used.

4.2 Reactor Control and Safety System

4.2.1 Control Rods

Applicability: These specifications apply to the surveillance requirements for control rod operability.

Objective: The objective is to verify the performance and operability of those systems and components which are directly related to reactor safety.

Specifications:

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(1) The control rods shall be visually inspected at biennial intervals.

(2) The scram time shall be measured annually.

(3) The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated semiannually.

Basis:

Specification (1) requires biennial inspection of the control rods, which has been shown to provide sufficient information on the condition of the control rods.

Specification (2) requires measurement of the scram time on an annual basis as a check not only of the scram system electronics, but also as an indication of the capability of the control rods to perform properly.

Specification (3) requires inspection, cleaning, and lubrication of the transient rod cylinder and air supply because these parts undergo more movement than the other control rods, and have parts which may need to be serviced more frequently. As a result, the inspection interval has been set on a semiannual schedule.

4.2.2 Reactor Measuring Channels

Applicability: This surveillance requirement applies to reactor measuring channel operability.

Objective: The objective of this surveillance requirement is to provide assurance that the reactor measuring channels are operable.

Specifications:

(1) A channel test of each of the required operable measuring channels listed in Table 3.1 for

the intended mode of operation shall be performed before each day's operation or before each operation extending more than one day.

(2) A channel check of the fuel rod temperature measuring channel shall be made each time the reactor is operated in the steady state mode by comparing the indicated instrumented fuel rod temperature with previous indicated temperature values for the same core configuration and power level.

Basis:

Specification (1) requires a channel test to confirm that the reactor measuring channels are operable.

Specification (2) provides for a channel check of the fuel temperature measuring channel to be carried out by comparing indicated fuel temperatures over time to determine whether the reactor fuel temperature measuring channel is behaving consistently.

4.2.3 Reactor Safety System

Applicability: This surveillance requirement applies to reactor safety system operability

Objective: The objective of this surveillance requirement is to provide assurance that the reactor safety systems are operable.

Specifications:

The surveillance requirements in this section may be postponed during periods of reactor shutdown. If the surveillance requirement occurs during a period of reactor shutdown the surveillance shall be completed upon resumption of reactor operation.

(1) A channel test of each of the safety channels listed in Table 3.2, except for the bulk primary coolant temperature, for the intended mode of operation (steady-state or pulse) shall be performed before each day's operation or before each operation extending more than one day.

(2) A channel check of the bulk primary coolant temperature shall be performed before each day’s operation or before operation extending more than one day.

(3) A test of the interlocks in Table 3.3 for the intended mode of operation (steady-state or pulse) shall be performed before each day's operation or before each operation extending more than one day.

(4) A channel calibration of the fuel rod temperature measuring channel shall be performed semiannually by the substitution of a thermocouple simulator in place of the instrumented fuel rod thermocouple.

(5) A channel calibration shall be made of the power level monitoring channels annually or after a core configuration change, by the calorimetric method.

Basis:

Specification (1) requires a channel test of each safety channel listed in Table 3.2, except the bulk primary coolant temperature, to provide verification of the operability of each safety channel.

Specification (2) requires a channel check of the bulk primary coolant temperature to provide assurance that the bulk pool temperature limit will not be exceeded.

Specification (3) requires a test of the interlocks listed in Table 3.3 to provide verification of the operability of each interlock.

Specification (4) requires a semiannual channel calibration because operational experience with the TRIGA system gives assurance that thermocouple measurements are sufficiently reliable and stable to provide an accurate indication of fuel rod temperature.

Specification (5) specifies the calorimetric method, which is the standard method to perform the calibration of the reactor power measuring channels. Annual calibration has been observed to provide assurance that the power level measuring channels are providing accurate power level indications. A recalibration is required after core configuration changes due to the possibility that a change in flux distribution could lead to inaccurate power level readings.

4.2.4 Pool Level Alarm

Applicability: This surveillance requirement applies to the surveillance of the reactor pool level alarm.

Objective: The objective of this surveillance requirement is to provide assurance that the reactor pool level alarm is operable.

Specification: The reactor pool level alarm shall be monthly tested for operability.

Basis: Monthly testing of the reactor pool level alarm has demonstrated that monthly surveillance is adequate to confirm operability.

4.3 Primary Coolant Conditions

Applicability: This specification applies to the surveillance of primary coolant water quality.

Objective: The objective is to ensure that primary coolant water quality does not deteriorate.

Specifications:

(1) The conductivity and pH of the primary coolant water shall be measured at least once every 2 weeks.

(2) The radionuclide content of the reactor pool water shall be monitored monthly. Steps shall be taken to isolate the source of the radioactivity and to mitigate the problem if the radionuclide content of the pool water in the reactor pool exceeds one-third (1/3) of the 10 CFR 20 Appendix B, Table 3 value.

Basis: These surveillance requirements ensure that primary coolant water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

Specification (1) provides for monitoring of primary coolant water conductivity and pH to provide timely information of possible changes in primary coolant water chemistry. The primary coolant water purification system and buffering of water pH due to atmospheric carbon dioxide act to stabilize primary coolant chemistry against sudden changes, and as a result the surveillance interval has been shown to provide assurance that primary coolant chemistry lies within the acceptable ranges.

Specification (2) provides for monthly monitoring of the radionuclide content in the pool water to provide information as a means to detect, in a timely fashion, a leak of radioactive fission products from fuel or a leak of a sealed source. Leakage of the primary coolant from the reactor pool is an analyzed event in terms of the potential impact of a pool water leak on effluent release limits. As long as the radionuclide content of the pool water remains below 10 CFR 20 effluent release limits it would be possible to release the entire contents of the pool directly into the sewer system, without dilution, and not violate the 10 CFR 20 release limits. As a result, the specification limiting radionuclide content of the pool water is intended to prevent the possibility of exceeding the 10 CFR 20 release limits under any circumstances or condition of operation.

4.4 Ventilation System

Applicability: This specification applies to surveillance requirements for the pool room ventilation system.

Objective: The objective is to ensure the proper operation of the pool room ventilation system in all operational modes.

Specifications:

(1) The operation of the pool room ventilation system shall be checked monthly by cycling the system from the “normal” to the “isolate” and “dilution” modes of operation. The positions of the associated dampers, indicator display, and fan operation shall be visually checked to ensure correspondence between the device performance and selected mode of operation.

(2) The pressure drop across the absolute filter in the pool room ventilation system shall be measured semiannually.

(3) The air flow rates in the ventilation system shall be measured biennially.

Basis:

Specification (1) requires visual confirmation of ventilation system operability by cycling the system through the operational modes while observing the fans and dampers. This is to be done to provide a high degree of confidence that the ventilation system is operable.

Specification (2) requires the pressure drop across the pool room ventilation filter to be measured to determine whether the filter needs to be changed.

Specification (3) requires air flow rates to be measured to provide data which can be used to determine whether the ventilation system is operating as designed.

4.5 Radiation Monitoring System and Effluents

4.5.1 Radiation Monitoring System

Applicability: This specification applies to the surveillance monitoring for the area monitoring equipment, argon-41 monitoring system, and continuous air monitoring system.

Objectives: The objectives are to ensure that the radiation monitoring equipment is operating properly and capable of performing its intended function, and that the alarm points are set correctly.

Specifications:

(1) All radiation monitoring systems shall be verified to be operable by a monthly channel test.

(2) The following surveillance activities shall be performed on an annual basis:

 (a) The reactor bridge and beam room radiation monitoring system shall be calibrated using a certified radioactive source.

 (b) A calibration shall be performed on the continuous air monitor in terms of counts per unit time per unit of radioactivity using calibrated beta-particle emitting sources.

 (c) A calibration of the exhaust gas monitor system shall be done using at least two different calibrated gamma-ray sources.

Basis:

Specification (1) requires monthly verification of radiation monitoring systems operability to confirm that the systems are performing as designed.

Specification (2) requires calibration of the bridge and beam room monitoring system, the continuous air monitor, and the exhaust gas monitor because these radiation detection systems provide important information about the working environment within the licensed areas of the Dodgen Research Facility and about the gas effluent releases due to reactor operation.

4.5.2 Effluents

Applicability: This surveillance requirement applies to monitoring of gaseous and liquid effluents.

Objective: The objective of this surveillance requirement is to ensure that gaseous and liquid effluents have radionuclide contents that are below 10 CFR 20 limits.

Specifications:

(1) The level of ^{41}Ar in the effluent gas shall be continuously monitored during operation of the reactor.

(2) The environmental radiation monitoring program required by Section 3.5.2(2) shall measure the integrated radiation exposure in and around the Dodgen Research Facility on a quarterly basis.

(3) The radiation levels determined by the environmental monitoring program shall be tabulated and examined annually.

(4) The annual discharge of ^{41}Ar shall be calculated annually.

(5) The continuous air monitor shall be continuously monitored during operation of the reactor.

(6) Before discharge, the facility liquid effluents collected in the holdup tanks shall be analyzed for their radioactivity content.

Basis:

Specification (1) requires continuous monitoring of ^{41}Ar releases to determine compliance with 10 CFR 20 release limits, and the limits imposed by these Technical Specifications.

Specification (2) requires quarterly measurement of radiation in and around the Dodgen Research Facility to demonstrate compliance with limits of radiation exposure to the public and to personnel in the Dodgen Research Facility.

Specification (3) requires annual tabulation and examination of radiation levels determined by the environmental monitoring program to provide data for annual reporting requirements.

Specification (4) requires annual determination of ^{41}Ar releases to provide data for reporting requirements and for determination of compliance with release limits imposed by regulation and by these Technical Specifications.

Specification (5) requires that the continuous air monitor be used during the course of reactor operation because this detector can provide information about a leak of fission products from the reactor and about the level of radiation in the reactor pool room air.

Specification (6) requires analysis of the radioactivity content of liquid effluents before release to assure compliance with 10 CFR 20 liquid effluent release limits.

4.6 Limitations on Experiments

Applicability: This specification applies to the surveillance requirements for experiments installed in the reactor and its experimental facilities.

Objective: The objective is to provide assurance that experiments are adequately planned, reviewed and carried out in order to protect the reactor, facilities, personnel and the environment.

Specifications:

(1) The reactivity worth of moveable experiments shall be shown by measurement, testing, calculation, or comparison to other experiments, to be less than $1.00. This surveillance requirement shall be considered to be satisfied for subsequent movable experiments that exhibit the same characteristics as a previously analyzed moveable experiment.

(2) The reactivity worth of a secured experiment shall be shown by measurement, testing, calculation, or comparison to other experiments, to be less than $2.00. This surveillance requirement shall be considered to be satisfied for subsequent secured experiments when a measurement of an initial secured experiment is applied to subsequent secured experiments that are the same as the initially analyzed secured experiment.

(3) The sum of absolute values of all individual experiments shall be shown to be less than $5.00.

(4) The following surveillance requirements apply to use of explosive materials in experiments:

  (a) The quantity of explosive materials (if any) used in an experiment shall be documented and shown to be less than 25 mg.

  (b) Testing of explosive material encapsulation shall be documented and shown to be in accordance with Section 3.6 (4).

(5) A safety analysis shall document conformance to the requirements of Section 3.6(5).

(6) A safety analysis shall document conformance to the requirements of Section 3.6(6).

(7) A safety analysis shall document that the total radioactive inventory of the iodine isotopes 131 through 135 in a fueled experiment is less than 1.5 Ci.

(8) The results of an inspection and any corrective action taken following a sample failure that releases material that could damage reactor fuel or the reactor structure shall be reviewed by the facility Director and the Reactor Safeguards Committee and shall be determined to be satisfactory before operation of the reactor is resumed.

(9) Minor modifications to a reviewed and approved experiment may be made at the discretion the Reactor Supervisor, provided that the hazards associated with the modifications have been reviewed and a determination has been made and documented that the modifications do not create a significantly different, a new, or a greater hazard than the originally approved experiment.

Basis:

Specification (1) requires a determination of the reactivity worth of moveable experiments to provide a high degree of confidence that the reactor cannot be inadvertently brought to a prompt critical state by installation or removal of a moveable experiment and that compliance is maintained with these Technical Specifications.

Specification (2) requires a determination of the reactivity worth of secured experiments to provide a high degree of confidence that the experiment cannot cause undesirable influences on reactor behavior, such as inducing unacceptable changes in shutdown margin, and that compliance is maintained with these Technical Specifications.

Specification (3) requires a determination of the reactivity worth of all experiments to provide a high degree of confidence that the assembly of experiments cannot cause undesirable influences on reactor behavior, such as inducing unacceptable changes in shutdown margin, and that compliance is maintained with these Technical Specifications.

Specification (4) provides surveillance requirements on the use of explosives in experiments to remove the risk of damage to the reactor due to an inadvertent detonation of the explosive material.

Specifications (5) and (6) provides for a safety analysis for experiments involving materials which could off-gas, sublime, volatilize or produce aerosols and provides specifications for acceptance criteria.

Specification (7) provides a surveillance requirement for the permitted inventory of radioactive isotopes of iodine in an experiment to protect the health and safety of personnel and the public if the experiment suffers a catastrophic failure and releases all of the radioactive iodine.

Specification (8) provides a requirement for conduct of a review of inspection and corrective actions following a sample failure to provide a high degree of assurance that the research reactor may be safely operated.

Specification (9) permits minor modifications to previously approved experiments, but only if the modifications do not present hazards that are significantly different than the previous experiment.

4.7 Sealed Sources in the Reactor Pool

Applicability: This specification applies to the surveillance requirements for sealed radioactive sources in the reactor pool.

Objective: The objective is to provide assurance that sealed sources are stored and used in such a way so that they do not present a hazard to the reactor, reactor facility personnel, the public or the environment.

Specifications:

(1) An inventory of sealed sources in the reactor pool shall be kept to demonstrate that the total radioactivity remains below 100,000 curies.

(2) The written Standard Operating Procedure for storage and use of sealed sources shall be reviewed and approved by the Reactor Safeguards Committee.

Basis:

Specification (1) requires an up-to-date inventory to provide data to show the amount of sealed source radioactivity stored in the reactor pool.

Specification (2) requires that a written Standard Operating Procedure be reviewed by the Reactor Safeguards Committee to provided additional oversight over the use of sealed sources.

4.8 Boron Neutron Capture Facility

Applicability: This specification applies to the surveillance requirements for the BNC facility.

Objective: The objective is to assure the proper operation of the BNC facility.

Specifications:

(1) Operability of the BNC facility reactor scram mechanism shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment. A scram test shall be carried out from full power before BNC facility use if more than 6 months have elapsed during which the BNC facility was not used for BNC experiments, or after a component or system modification which could affect the scram system.

(2) Single door access to the facility shall be confirmed before each day of operation of the BNC facility.

(3) The operability of each system listed in Specifications 3.8(3)(a) through 3.8(3)(h) shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(4) Operability of the reactor bridge position sensor shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(5) Operability of an alternate means of monitoring reactor bridge position in the event of a failure of the bridge position readout, as permitted in Technical Specification 3.8(5), shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(6) Operability of the display of the reactor linear power and log power display channels shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(7) Operability of the radiation monitor, radiation monitor alarm, portable radiation detection instruments and personal dosimeters shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment. The radiation monitor shall be calibrated quarterly for every calendar quarter that the BNC facility is in use to perform a BNC experiment.

(8) Operability of the intercom system or two way communication system shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(9) Operability of the BNC facility access door shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(10) Operability of the closed circuit television monitoring system and emergency lighting and backup power system shall be checked and documented before operation of the experiment commences each day that the BNC facility is used to carry out a BNC experiment.

(11) Maintenance, repair and modification of the BNC facility, including identification of the supervising Senior Reactor Operator, shall be documented.

(12) Before each day's operation it shall be verified that:

(a) the instructions have been posted;

(b) training of personnel who operate the controls of the BNC facility has been documented before being permitted to operate the BNC facility controls.

(c) the procedure for conduct of the BNC experiment has been posted.

(d) instruction has been given to BNC facility personnel that the reactor bridge shall be moved only with permission of a Reactor Operator or Senior Reactor Operator and under the direct supervision of a Senior Reactor Operator.

(13) It shall be documented before use of the BNC facility commences that the approval of a senior reactor operator is required before performing any action that affects the reactivity of the research reactor.

(14) Characterization of the BNC neutron beam and beam monitors shall be carried out before initiation of use of the neutron beam for BNC experiments. The following surveillances apply:

(a) beam intensity, diameter and divergence shall be measured according to the following schedule:

(i) before initial use in a BNC experiment, weekly for any week that the beam will be used and semiannually for any six month interval that the BNC neutron beam will be used for BNC experiments;

(ii) prior to performance of a BNC experiment if a beam component has been modified or replaced in the interim since a prior beam characterization was carried out;

(b) beam monitors shall be calibrated at least once every two years for any two year interval that the neutron beam is used for BNC experiments. An initial calibration shall be made prior to carrying out a BNC experiment if more than two years have elapsed during which the BNC facility was not used to perform BNC experiments. The beam monitors shall be calibrated by a means that is traceable to the National Institute of Standards and Technology, and shall measure dose or dose rate.

Basis:

Surveillance requirement (1) requires operability of the BNC facility to be checked and documented before each day of use of the BNC facility to provide assurance that the scram mechanism will be operable during the course of a BNC experiment.

Surveillance requirement (2) requires that one access door be available for access to the BNC facility at all times to maintain access control.

Surveillance requirements (3) provides for checks to BNC facility and research reactor systems to provide assurance that the BNC facility and research reactor can be safely used in the conduct of a BNC experiment.

Surveillance requirement (4) requires a daily check and documentation of the operability of the reactor bridge position sensor to ensure that reactor bridge position information is continuously available to BNC facility personnel.

Surveillance requirement (5) allows a check for operability of alternate means of monitoring reactor bridge position if the reactor bridge position sensor is not operable.

Surveillance requirement (6) provides for a check and documentation of the operability of the linear power and log power displays to give assurance that research reactor power level information will be readily available to BNC facility personnel.

Surveillance requirement (7) specifies checks of the operability of the radiation monitor, radiation monitor alarm, portable radiation detection instruments and personal dosimeters before each day to provide assurance that the required monitors and alarms are operable to maintain a safe work environment for BNC personnel.

Surveillance requirement (8) requires checking of the communication system between the BNC facility and the reactor control room to provide assurance that personnel have adequate two way communication.

Surveillance requirement (9) requires an operability check of the BNC facility access door because the door is part of the reactor positioning system.

Surveillance requirement (10) requires a check for operability of the closed circuit television, emergency lighting, and backup power systems to provide assurance that the video monitoring system will be operable during the course of a BNC experiment.

Surveillance requirement (11) provides for documentation of oversight of maintenance, repair and modification of the BNC facility to provide assurance that that all such activities are performed under the supervision of personnel cognizant of radiation safety, system function and reliability, and quality assurance.

Surveillance requirements 12(a) through 12(d) provide assurance that the required posting of instructions, training of BNC facility personnel and posting of procedures have been accomplished.

Surveillance requirement (13) limits actions and requires documentation of actions that influence reactivity of the research reactor because changes in reactivity, which can change reactor power levels and power distribution, may only be done with the approval of a senior reactor operator.

Surveillance requirement (14) provides assurance that the BNC neutron beam characteristics will be well understood when the beam is used for BNC experiments.

5 DESIGN FEATURES

5.1 Site and Facility Description

Applicability: This specification applies to the Washington State University research reactor site location and facility description. The research reactor is located within the Dodgen Research Facility, which is a concrete building located approximately one mile east of the main portion of the WSU campus.

Objective: The objective is to specify the site and the facility of the Washington State University research reactor.

Specifications:

(1) The site is that area bound by the perimeter that encloses the Nuclear Radiation Center building, (also known as the Dodgen Research Facility), the fenced area immediately outside the east pool room loading dock door and the fenced area immediately outside the beam room west loading dock door.

(2) The Washington State University research reactor shall be located in the licensed area of the Dodgen Research Facility.

(3) The facility shall be the following:

   (a) the room in which the WSU research reactor is located, also known as Room 201, the reactor control room which is within Room 201, the pump room, primary coolant water purification room, primary coolant and makeup water valve manifold room;

   (b) the research reactor beam room, also known as Room 2.

(4) The facility shall be a restricted area.

Basis:

Descriptions of the Nuclear Radiation Center, Dodgen Research Facility, reactor building, and site are provided in detail in the Washington State University Safety Analysis Report.

Specification (1) provides a description of the site.

Specifications (2) and (3) provide a description of the location of the research reactor and the licensed area, and that the location of the research reactor shall be within the licensed area of the Dodgen Research Facility.

Specification (4) requires that the facility be a restricted area.

5.2 Reactor Fuel

Applicability: This specification applies to the fuel rods used in the reactor core.

Objective: The objective is to ensure that the fuel rods are designed and fabricated to provide confidence that the fuel has a high degree of reliability with respect to physical and nuclear characteristics.

Specifications:

(1) The unirradiated 30/20 fuel rods shall have the following characteristics:

(a) the uranium content shall be a maximum of 30% by weight uranium, enriched to less than 20% ^{235}U;

(b) the hydrogen to zirconium ratio (in the ZrH_x) shall be a nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65;

(c) the erbium content shall be homogeneously distributed with a nominal 0.90% by weight;

(d) the cladding shall be 304 stainless steel with a nominal thickness of 0.020 inches.

(2) The unirradiated standard fuel rods shall have the following characteristics:

(a) the uranium content shall be a maximum of 9.0% by weight enriched to less than 20% ^{235}U;

(b) the hydrogen to zirconium atom ratio (in the ZrH_x) shall be between 1.5 and 1.8;

(c) the cladding shall be 304 stainless steel with a nominal thickness of 0.020 inches.

Basis:

The fuel specification permits a maximum uranium enrichment of less than 20% in the 30/20 LEU fuel. This is about 1% greater than the design value of 19.75% enrichment ^{235}U . Such an increase in loading would result in an increase in the power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2%.

The fuel specification for a single fuel element permits a minimum erbium content of 5.6% less than the design value of 0.9% by weight, (however, the quantity of erbium in the full core shall not deviate from the design value by more than -3.3%). This variation for a single fuel element would result in an increase in the fuel element power density from 1% to 2%. Such a small increase in local power density would reduce the safety margin by less than 2%

The maximum hydrogen to zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel rod clad about a factor of two greater than that for a hydrogen-to-zirconium ratio of 1.6. This increase in the clad stress during an accident would not exceed the rupture strength of the clad.

A maximum uranium content of 9% by weight ^{235}U for the standard TRIGA elements is about 6% greater than the design value of 8.5% by weight ^{235}U. Such an increase in loading would result in an increase in the power density of 6% and reduces the safety margin by 10% at most. The maximum hydrogen to zirconium ratio of 1.8 could result in the maximum stress under accident conditions in the fuel rod clad being about a factor of 2 greater than the value resulting from a hydrogen to zirconium ratio of 1.60. However, this increase in the clad stress during an accident would not exceed the rupture strength of the clad.

5.3 Reactor Core

Applicability: This specification applies to the configuration of fuel and in-core experiments.

Objective: The objective of this specification is to ensure that provisions are made to restrict the arrangement of fuel rods and experiments to provide assurance that excessive power densities will not be produced.

Specifications:

(1) The core shall be an arrangement of TRIGA uranium zirconium hydride fuel moderator assemblies positioned in the reactor grid plate.

(2) The TRIGA core may be composed of 30/20 fuel or a combination of standard and 30/20 fuel (mixed cores) provided that the 30/20 fuel region contains at least 51 30/20 fuel rods located in a contiguous block in the central region of the core.

(3) A reactor core fueled with a mixture of fuel types shall not be operated with a vacant core lattice position in the 30/20 fuel region. Water holes in the 30/20 fuel region shall be limited to single-rod holes. Lattice positions in the fueled region of the core that are not occupied by fuel assemblies, reflectors, or experiments shall be occupied by fixtures that will prevent the installation of a fuel assembly into a position not occupied by a fuel assembly, reflector or experiment.

(4) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite, aluminum and water.

Basis:

Specification (1) calls for the use of TRIGA fuel in the WSU research reactor. Standard TRIGA cores have been used for years and their characteristics are well-documented. Mixed cores of 30/20 fuel and standard fuel have been tested by WSU and General Atomics and operated successfully. Calculations, as well as measured performance of mixed cores in the WSU reactor, have shown that such cores can be safely operated.

Specification (2) describes possible reactor core compositions. In mixed cores, it is necessary to arrange 30/20 fuel rods in a contiguous, central region of the core to control flux peaking and power generation peak values in individual fuel rods.

Specification (3) describes conditions that apply to core lattice positions. Core lattice positions in the standard fuel region may contain experiments or an experimental facility which prevent accidental fuel additions to the reactor core. Vacant core lattice positions are not permitted in the 30/20 fuel region.

Specification (4) describes types of permissible core reflectors. The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

5.4 Control Rods

Applicability: This specification applies to the control rods used in the reactor core.

Objective: The objective is to ensure that the control rods have a high degree of reliability with respect to physical and nuclear characteristics.

Specifications:

(1) Standard control rods shall have scram capability and contain borated graphite, B_4C powder, boron or boron compounds in solid form within aluminum or stainless steel cladding.

(2) The regulating control rod does not have scram capability and shall be stainless steel.

(3) The transient control rod shall have scram capability and contain borated graphite or boron compounds in a solid form within aluminum or stainless steel. The transient rod shall have an adjustable upper limit to allow variation of reactivity insertions. The transient control rod does not incorporate a fueled follower.

Basis:

Specification (1) describes the requirement that the standard control rods have scram capability, which allows the reactor operator to quickly shut down the research reactor. The poison requirements for the control rods are satisfied by using neutron-absorbing borated graphite, B_4C powder, boron or boron compounds. These neutron poisons shall be contained in a suitable cladding material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the neutron poison from the pool water.

Specification (2) describes the scram capability and composition of the regulating rod. Since the regulating rod is a low worth control rod, its function is satisfied by using stainless steel.

Specification (3) describes the requirements for the transient rod. The transient control rod, which is used to produce a reactor pulse, shall have provisions for variable reactivity insertion.

5.5 Fuel Storage

Applicability: This specification applies to the storage of reactor fuel and fueled devices at times when it is not in the reactor core.

Objective: The objective is to ensure that reactor fuel rods and fueled devices in storage will not become critical ($k_{eff} = 1$) and will not reach an unsafe temperature.

Specifications:

(1) All fuel rods and fueled devices shall be stored in a geometrical array where the k_{eff} is less than 0.8 for all conditions of moderation and reflection.

(2) Irradiated fuel rods and fueled devices shall be stored in an array which will permit sufficient natural convective cooling by water or air, so that the fuel rod or fueled device temperature will not exceed design values.

Basis: The limits imposed by Specifications 5.5(1) and 5.5(2) are conservative and ensure safe storage.

5.6 Radiation Monitoring System

Applicability: This specification describes the functions and components of the area radiation monitoring equipment, the continuous air monitor and the exhaust gas monitor.

Objective: The objective is to describe the radiation monitoring equipment that shall be operable to ensure safe operation of the reactor.

Specifications:

(1) The area radiation monitors shall be sensitive to gamma radiation, shall monitor radiation fields in key locations, and shall alarm and readout at the reactor control console.

(2) The Continuous Air Monitor shall:

- (a) be capable of particulate collection, and detection of beta and gamma radiation;
- (b) monitor particulate radioactivity in the pool room air, alarm and readout at the reactor control console;
- (c) be capable of causing the building ventilation system to switch from the normal mode into the dilution mode upon initiation of a high continuous air monitor alarm signal when the reactor is operating.

(3) The exhaust gas monitor shall be capable of detecting gamma radiation, and shall monitor ^{41}Ar content in ventilation system exhaust air, and shall alarm and readout at the reactor control console.

Basis:

The area radiation monitoring system described in specification (1) is intended to provide information of the level of radiation to operating personnel to maintain a safe work environment and if necessary, to evacuate the facility and take the necessary steps to prevent the spread of radioactive materials to the surroundings.

The continuous air monitor described in specification (2) samples air above the reactor core, and would be the most likely monitor to first detect a leak of fission products from the reactor.

The exhaust gas monitor described in specification (3) is required to sample and monitor ventilation system exhaust air to provide a live time indication and a record of the level of radioactivity in building exhaust gas effluent. The primary effluent component that is monitored is Ar-41.

5.7 Reactor Building and Ventilation System

Applicability: This specification applies to the building that houses the reactor.

Objective: The objective is to ensure that provisions are made to restrict the amount of radioactivity released into the environment.

Specifications:

(1) The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be at least 10^9 cm^3.

(2) The reactor building shall be equipped with a ventilation system designed to filter and exhaust air or other gases from the reactor building and release them from a stack at a height of 46 ± 2 feet from the ground level in the front of the Dodgen Research Facility building.

(3) A set of controls for the ventilation system shall be located outside the reactor pool room and control room areas. The controls shall be capable of changing the ventilation system mode of operation into the dilute or isolate mode.

(4) The reactor pool room ventilation system shall have a dilution mode of operation with the following characteristics:

   (a) air from the reactor pool room shall be mixed and diluted with outside air before being discharged from the facility when the ventilation system is operated in the dilution mode;

(b) the exhaust air from the reactor pool room shall pass through a filter before being discharged from the facility when the ventilation system is operated in the dilution mode.

Basis:

Specification (1) addresses facility design. The facility is designed so that the ventilation system will maintain a negative pressure with respect to the atmosphere outside the reactor pool room to minimize uncontrolled leakage to the environment. The reactor pool room air would be maintained at a negative pressure with respect to the surroundings in both the normal and dilute modes, with respect to limiting the release of effluents through controlled pathways. The free air volume within the reactor building is isolated when the ventilation system is operated in the isolation mode.

Specification (2) provides a requirement for the air exhaust system stack height to be above ground level to provide a greater dilution effect for exhaust effluent gases than would be the case for ground level releases.

Specification (3) provides a requirement that a set of controls for isolation, dilution, and normal operation of the ventilation system are located external to the control and reactor pool rooms, so that proper handling of airborne radioactive materials in emergency situations can be managed with a minimum of exposure to personnel.

Specification (4) provides a requirement that the reactor pool room ventilation system have a dilution mode to minimize release rates of airborne radioactive material, and that the exhaust air be passed through an air filter before release. This is to control the rate at which airborne effluents can be released from the facility. The air filtration system will remove most particulate material, restricting releases to gases that can pass though an air filter, such as argon, krypton and xenon.

5.8 Reactor Pool Water System

Applicability: This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective: The objective is to ensure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding.

Specifications:

(1) The reactor core shall be cooled by natural convection water flow.

(2) All piping extending more than 5 feet below the surface of the pool shall have adequate provisions to prevent inadvertent siphoning of the pool.

(3) A pool level alarm shall be provided to indicate in the reactor control room and at a remote location of a loss of coolant if the pool level drops more than 8 inches below the normal level.

(4) The reactor primary coolant pool shall provide for at least 16 feet of water above the top of the core.

Basis:

Specification (1) is based on thermal hydraulic calculations which show that the WSU research reactor can operate in a safe manner with natural convection flow of the primary coolant at power levels up to 1.3 MW with primary coolant temperatures as high as 50 °C.

Specification (2) requires a means to prevent accidental siphoning through pipes so the pool water level cannot drop due to a siphon discharge to a level that would not provide sufficient shielding from radiation when the reactor is shut down.

Specification (3) requires that a loss of coolant alarm initiates a signal to provide a rapid means of alerting the reactor staff to a drop in primary coolant level.

Specification (4) requires a minimum value for water height over the reactor core to ensure that the reactor will not be operated without the appropriate pressure for the reactor coolant.

6 ADMINISTRATIVE CONTROL

6.1 Responsibility and Organization

(1) The Washington State University research reactor shall be operated by the Nuclear Radiation Center of Washington State University. The organization of the research reactor facility management and operation shall be as shown in Figure 6.1. The responsibilities and authority of the Level 2, Level 3, and Level 4 operating staff shall be defined in writing in these Technical Specifications.

(2) The following organizational levels and responsibilities shall exist:

(a) Vice President for Research (Level 1): The Vice President for Research is the head of the WSU Office of Research.

(b) Director of the Nuclear Radiation Center (Level 2): The Director of the Nuclear Radiation Center shall report to the Vice President for Research. The Director is responsible for ensuring that regulatory requirements and implementation are in accordance with requirements of the U.S. Nuclear Regulatory Commission, the Code of Federal Regulations, the State of Washington, and Washington State University regulations and the requirements of the WSU Reactor Safeguards Committee.

(c) Reactor Supervisor (Level 3):

(i) The Reactor Supervisor shall report to the Director of the Nuclear Radiation Center and is responsible for guidance, oversight, and technical support of reactor operations.

(ii) The Reactor Supervisor shall report to the Director of the Nuclear Radiation Center and to the Reactor Safeguards Committee in matters of radiation protection.

(d) Reactor Operating Staff (Level 4): The reactor operating staff shall report to the Reactor Supervisor. Reactor operating staff shall include one or more licensed Senior Reactor Operator, Reactor Operator or Reactor Operator trainee.

(e) Radiation Protection

(i) Radiation protection activities shall be carried out by Level 3 or Level 4, with supervisory function performed by the Level 3, Reactor Supervisor.

(ii) The Reactor Safeguards Committee shall perform the review and audit function over the radiation protection activities within the facility.

(iii) The Director of the Radiation Safety Office, as an ex-officio member of the Reactor Safeguards Committee, shall provide communication regarding radiation safety to the Director of the Nuclear Radiation Center.

(iv) The Director of the Radiation Safety Office shall have oversight, through the Reactor Safeguards Committee, of activities utilizing radioactive material.

(3) Responsibilities of one level may be assumed by higher levels or by alternates designated by a higher level, conditional upon meeting all requirements for the position.

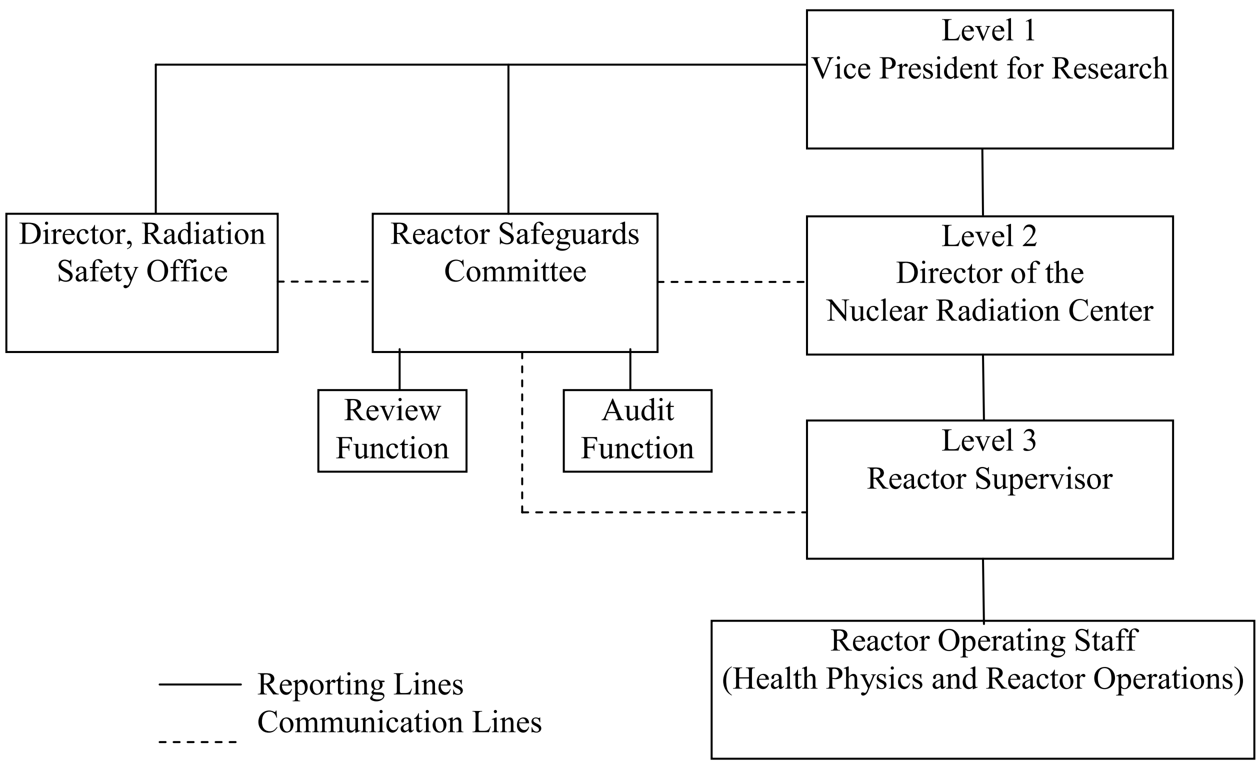

Figure 6.1 Facility organization

6.2 Staffing

6.2.1 Minimum Staffing Levels

(1) When the reactor is not secured, the minimum staffing level shall consist of:

(a) a licensed Reactor Operator or Senior Reactor Operator in the control room;

(b) a second designated person present at the facility complex able to carry out written instructions;

(c) a designated Senior Reactor Operator who shall be readily available in the Dodgen Research Facility or on call.

(2) A Senior Reactor Operator who is "on call" shall be defined as an individual who:

(a) has been specifically designated and this designation is known to the Reactor Operator on duty;

(b) keeps the Reactor Operator on duty informed of where he/she can be rapidly contacted and the contact telephone number;

(c) is capable of getting to the reactor facility in less than 30 minutes and shall remain within a 15 mile radius of the facility;

(3) It is not necessary to have a Senior Reactor Operator on call if the Reactor Operator in the control room is a Senior Reactor Operator. If the Reactor Operator in the control room is a Senior Reactor Operator a second person shall be present in the facility as described in Section 6.2(1)(b).

6.2.2 Contact Information

(1) A list of personnel including name and telephone number shall be readily available in the control room for use by the Reactor Operator. The list shall include:

(a) facility Director;

(b) Reactor Supervisor;

(c) all licensed Reactor Operators and Senior Reactor Operators.

6.2.3 Events Requiring the Direction of an SRO

A licensed senior reactor operator shall be present at the facility for:

(1) Initial startup and approach to power;

(2) All fuel movement or relocation;

(3) All control rod relocations within the core region;

(4) Relocation of any in-core experiments or irradiation facilities with a reactivity worth greater than $1.00;

(5) Recovery from unplanned or unscheduled shutdown; or

(6) Recovery from unplanned or unscheduled significant power reduction.

6.3 Selection and Training of Personnel

The selection, training and requalification of each member of operations personnel shall meet or exceed the requirements of ANSI/ANS 15.4 – 2007, "Standard for the Selection and Training of Personnel for Research Reactors," for comparable positions.

6.4 Reactor Safeguards Committee

6.4.1 Function

The Reactor Safeguards Committee shall function to provide an independent review and audit of the Nuclear Radiation Center activities including:

(1) reactor operations;

(2) radiological safety;

(3) general safety;

(4) testing and experiments;

(5) licensing and reports;

(6) quality assurance.

6.4.2 Composition and Qualifications

(1) The Reactor Safeguards Committee shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety.

(2) The members of the Committee shall include:

   (a) one Senior Reactor Operator who may be the Director of the Nuclear Radiation Center. The presence of Nuclear Radiation Center staff members shall not be counted to constitute a quorum. Nuclear Radiation Center staff members shall not be voting members of the Committee.

   (b) WSU faculty and staff members designated to serve on the Committee in accordance with the procedures specified by the WSU committee manual.

(3) The Director of the WSU Radiation Safety Office shall be an ex-officio member of the Committee.

(4) The Reactor Safeguards Committee is a WSU Presidential Committee which performs reviews and audits of the WSU Nuclear Radiation Center. The Reactor Safeguards Committee reports to the WSU Vice President for Research.

6.4.3 Reactor Safeguards Committee Operation

The Reactor Safeguards Committee shall operate in accordance with a written charter, including provisions for:

(1) semiannual meetings of the full committee;

(2) voting rules;

(3) quorums: the committee chair or a designate and two voting members;

(4) method of submission and content of presentations to the committee;

(5) use of subcommittees;

(6) review, approval and dissemination of minutes.

6.4.4 Reviews

The responsibilities of the Reactor Safeguards Committee or designated subcommittee shall include the following:

(1) review and approval of new experiments utilizing the research reactor;

(2) review and approval of proposed changes to the following:

  (i) the operating license (R-76) by amendment;

  (ii) Standard Operating Procedures;

  (iii) Technical Specifications.

(3) review of the operation and operational records of the Nuclear Radiation Center;

(4) review of operating abnormalities or deviations from normal and expected performance of equipment with safety significance;

(5) review in accordance with 10 CFR 50.59 whether proposed changes in equipment, systems, tests, experiments or Standard Operating Procedures would be allowed without prior authorization by the U.S. Nuclear Regulatory Commission.

(6) review of reportable occurrences and the reports filed with the U.S. Nuclear Regulatory Commission for reportable occurrences;

(7) biennial review and approval of all standard operating procedures and changes to the standard operating procedures;

(8) biennial review of the emergency plan and the security plan;

(9) annual review of the radiation protection program;

(10) review audit reports.

6.4.5 Audits

(1) The RSC or a subcommittee shall audit reactor operations semiannually. The semiannual audit shall include at least the following:

  (a) review of the reactor operating records;

  (b) inspection of the reactor operating areas;

  (c) review of reportable occurrences;

  (d) radiation exposures within and outside the facility;

  (e) operations for conformance to the Technical Specifications and license conditions.

(2) The RSC or a subcommittee shall audit the following at biennial intervals:

  (a) emergency plan and implementing procedures;

  (b) retraining and requalification program;

  (c) security plan.

6.4.6 Records

The activities of the RSC shall be documented by the secretary of the Committee and distributed as follows:

(1) A written report of all audits performed under Section 6.4.5 shall be prepared and forwarded to Level 1 and Level 2 management within 3 months after the audit has been complete.

(2) A written report of all reviews performed under Section 6.4.4 shall be prepared and forwarded to the Level 1 and Level 2 management within 30 days following the completion of the review.

(3) The secretary of the Reactor Safeguards Committee shall maintain a file of the minutes of all meetings.

6.4.7 Experiment Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures. The following provisions shall be stated in a Standard Operating Procedure for review and approval of experiments:

(1) All new experiments or classes of experiments shall be:

a. installed in the reactor or in its irradiation facilities only after a safety analysis has been performed;

and

b. reviewed and approved by at least 2 Senior Reactor Operators, including written approval by Level 2 or Level 3 management for compliance with the Technical Specifications;

and

c. reviewed and approved by the Reactor Safeguards Committee.

(2) Substantive Changes to previously approved experiments shall be made only after review by the Reactor Safeguards Committee and approved in writing by Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

(3) An experiment shall not be installed in the reactor or in its irradiation facilities until after a safety analysis has been performed and reviewed for compliance with Section 3.6 by the Reactor Safeguards Committee in accordance with Section 6.4.7 of these Technical Specifications.

6.5 Radiation Safety

(1) The Reactor Supervisor (Level 3) shall have responsibility for implementing the radiation protection program using guidelines of ANSI/ANS-15.11-1993 (R2004). The Reactor Supervisor shall report to Level 2 management and shall communicate with the Reactor Safeguards Committee on matters of radiation safety.

(2) Radiation Protection

(a) The Reactor Safeguards Committee shall have oversight responsibility as defined in Section 6.1(2)(e)(ii) and 6.1(2)(e)(iv).

(b) The Reactor Operating Staff (Level 4) shall conduct radiation protection procedures in licensed areas, and shall report to the Reactor Supervisor (Level 3).

6.6 Action To Be Taken if a Safety Limit Is Exceeded

The following actions shall be taken if a safety limit is exceeded:

(1) The safety limit violation shall be reported within 24 hours by telephone to the U.S. Nuclear Regulatory Commission Operations Center.

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

(3) The safety limit violation shall be promptly reported to Level 1 management or designated alternates, to Level 2 management or designated alternates, to Level 3 management and to the Chair of the Reactor Safeguards Committee.

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

(a) applicable circumstances leading to the violation, the cause and contributing factors;

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

(c) corrective action to be taken to prevent recurrence.

(5) The report shall be submitted to the Reactor Safeguards Committee for review.

(6) A report shall be submitted in writing, within 10 days, to the U.S. Nuclear Regulatory Commission Document Control Desk.

6.7 Required Actions for Reportable Occurrences other than Safety Limit Violations

The following actions shall be taken as required by regulations or for a Reportable Occurrence, as defined in Section 1 for events that are reportable to the U.S. Nuclear Regulatory Commission within 24 hours. Reports are to be made to the U.S. Nuclear Regulatory Commission Operations Center for:

(1) Reactor conditions shall be returned to normal, or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operation of the reactor shall not be resumed unless authorized by Level 2 or designated alternates and the Chair of the Reactor Safeguards Committee.

(2) The occurrence shall be reported to Level 1 management, Level 2 management or designated alternates.

(3) The occurrence shall be reviewed by the Reactor Safeguards Committee at its next scheduled meeting.

(4) An immediate report of the occurrence shall be made to the Chair of the WSU Reactor Safeguards Committee.

(5) A report shall be prepared that includes an analysis of the causes and extent of possible damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. The report shall be submitted to the Reactor Safeguards Committee for review.

(6) A report shall be submitted in writing to the U.S. Nuclear Regulatory Commission Document Control Desk within 10 days.

6.8 Standard Operating Procedures

(1) Written procedures shall be prepared, reviewed, and approved prior to initiating any of the activities listed in this section. The procedures shall be reviewed by at least 2 Senior Reactor Operators. The procedures shall be reviewed and approved by the Reactor Safeguards Committee after approval by Level 2 management or designated alternates, and such reviews and approvals shall be documented.

(2) Written operating procedures shall be adequate to ensure the safe operation of the reactor, but shall not preclude the use of independent judgment and action if required to protect the health and safety of the public. Operating procedures shall be in effect for the following:

   (a) startup, operation and shutdown of the reactor;

   (b) fuel loading, unloading, and movement within the reactor;

   (c) maintenance of major components of systems which could influence reactor safety;

   (d) surveillance checks, calibrations, and inspections required by Technical Specifications or those that could have an influence on reactor safety;

   (e) personnel radiation protection, consistent with applicable regulations or guidelines. The procedures shall include management commitment and programs to maintain exposures and releases as low as reasonably achievable in accordance with guidelines of ANSI/ANS-15.11-1993 (R2004).

   (f) performing irradiations and experiments using the reactor;

   (g) implementation of emergency and security plans;

   (h) use, receipt, and transfer of radioactive material;

(i) control rod removal and replacement;

(j) reactor power calibration;

(k) performing maintenance and/or calibration on the reactor and associated equipment.

(3) Substantive changes to the previous procedures shall be made effective only after documented review by the review group of the Reactor Safeguards Committee and approval by Level 2 or designated alternates or if necessary, by a review under the regulations established by 10 CFR 50.59. Modifications to the procedures that do not change their original intent may be made by Level 3 or higher, but the modifications shall be approved by Level 2 or a designated alternate. Minor changes, such as corrections of typographical errors, editing for clarity or formatting that do not change the execution of the procedure may be made by any Senior Reactor Operator but the modifications shall be approved by Level 2 or a designated alternate. Temporary deviations from the original procedures may be made by the responsible Senior Reactor Operator or higher individual present to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported within 24 hours to the Level 2 or designated alternate.

6.9 Facility Operating Records

Records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination of single or multiple records. In addition to the requirements of applicable regulations, records and logs shall be prepared for at least the following items and retained for the periods of time indicated in Sections 6.9.1, 6.9.2 and 6.9.3:

6.9.1 Five Year Record Retention

Records of the following shall be kept for at least five years:

(1) normal reactor operation, including supporting documents such as pre-startup checklists and reactor operating log sheets;

(2) principal maintenance operations;

(3) reportable occurrences;

(4) surveillance activities required by the Technical Specifications;

(5) experiments performed with the reactor;

(6) approved changes in operating procedures;

(7) facility radiation and contamination surveys;

(8) Reactor Safeguards Committee meeting records and audit reports.

6.9.2 Life of the Facility Records Retention

Records of the following components or items shall be kept for the life of the facility:

(1) gaseous and liquid radioactive effluents released to the environs;

(2) off-site environmental monitoring surveys required by the Technical Specifications;

(3) radiation exposures for all personnel monitored;

(4) updated, corrected and as-built drawings of the reactor facility;

(5) fuel inventories, receipts, and shipments;

(6) reviews and reports of violations of Safety Limits;

(7) reviews and reports of violations of a Limiting Safety Systems Setting;

(8) reviews and reports of violations of a Limiting Condition of Operation.

6.9.3 Training Records

Record of training, retraining and requalification of licensed personnel shall be maintained at all times the individual is employed or until the operator license is renewed.

6.10 Reports to the U.S. Nuclear Regulatory Commission

All reports in this Section shall be submitted to the U.S. Nuclear Regulatory Commission Document Control Desk.

6.10.1 Written Reports Due Within 10 Days

Written reports of the following shall be submitted to the U.S. NRC within 10 days:

(1) A release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure. The written report (and, to the extent possible, the preliminary telephone report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event.

(2) a violation of a safety limit;

(3) a reportable occurrence as defined in Section 1, “Reportable Occurrence,” of these Technical Specifications.

6.10.2 Written Reports Due Within 30 Days

Written reports of the following shall be submitted to the U.S. NRC within 30 days:

(1) a significant variation of measured values from a corresponding predicted or previously measured value of safety related operating characteristics occurring during operation of the reactor;

(2) a significant change in the transient or accident analysis as described in the Safety Analysis Report;

(3) permanent changes in the facility organization involving Level 1 or Level 2 management personnel.

6.10.3 Written Report Due Within 60 Days

A report shall be submitted within 60 days after completion of startup testing of the reactor upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level. The report shall describe the measured values of the operating conditions, including:

(1) an evaluation of facility performance in comparison with design predictions and specifications;

(2) a reassessment of the safety analysis submitted with the license application which discusses measured operating parameters when measurements indicate a substantial variation from prior analysis.

6.10.4 Written Report to the U.S. NRC Within 60 days after June 30 of Each Year

The annual report shall provide the following information:

(1) a brief narrative summary of

 (a) operating experience (including experiments performed),

 (b) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and

 (c) results of surveillance tests and inspections;

(2) tabulation of the energy output (in megawatt-days) of the reactor, the number of hours that the reactor was critical, the cumulative total energy output since initial criticality, and number of pulses greater than $1.00;

(3) the number of emergency shutdowns and inadvertent scrams, including reasons for them and actions taken to prevent recurrence;

(4) discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

(5) a brief description, including a summary of the safety evaluations of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR 50.59;

(6) a summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge;

(7) liquid waste (summarized on a monthly basis):

   (a) monthly radioactivity discharged;

   (b) total estimated quantity of radioactivity released (in curies);

   (c) an estimation of the specific quantity for each detectable radionuclide in the monthly release;

   (d) fraction of 10 CFR 20 Table 3, Appendix B limit for each detectable radionuclide taking into account the dilution factor from the total volume of sewage released by the licensee into the sewage system;

   (e) sum of the fractions for each radionuclide reported above;

   (f) total quantity of radioactive material released by the facility into the sewage system during the reporting period.

(8) gaseous waste (summarized on a monthly basis) radioactivity discharged during the reporting period, including:

   (a) total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method;

   (b) total estimated quantity of ^{41}Ar released (in curies) during the reporting period based on data from an appropriate monitoring system;

   (c) estimated average atmospheric diluted concentration of ^{41}Ar released during the reporting period in terms of μCi/mL and fraction of the applicable DAC value;

(d) total estimated quantity of radioactivity in particulate form with half-lives greater than 8 days (in curies) released during the reporting period as determined by an appropriate particulate monitoring system;

(e) average concentration of radioactive particulates with half-lives greater than 8 days released in μCi/mL during the reporting period;

(f) an estimate of the average concentration of other significant radionuclides present in the gaseous waste discharge in terms of μCi/mL and fraction of the applicable DAC value for the reporting period if the estimated release is greater than 20% of the applicable DAC.

(9) solid waste (summarized on an annual basis):

(a) total amount of solid waste packaged (in cubic feet),

(b) total radioactivity in solid waste in curies,

(c) the dates of shipment and disposal (if shipped off-site).

(10) an annual summary of the radiation exposure received by facility personnel and visitors in terms of the average radiation exposure per individual and the greater exposure per individual in the two groups. Each exposure in excess of the limits of 10 CFR 20 shall be reported, including the time and date of the exposure as well as the circumstances that led to the exposure.

(11) an annual summary of the radiation levels including contamination levels observed during routine surveys performed at the facility including a summary of the average and highest levels;

(12) an annual summary of environmental surveys performed outside the facility.

6.11 Written Communications

All written communications with the U.S. Nuclear Regulatory Commission shall be made in accordance with the requirements of 10 CFR 50.4 “Written Communications.”