WASHINGTON STATE UNIVERSITY

Nuclear Radiation Center

August 10, 2010

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Washington, DC 20555-0001

Reference: Washington State University
Docket No. 50-27, License No. R-76

Subject: Submittal of Proposed Technical Specifications for the Washington State University Research Nuclear Reactor

Washington State University (WSU) has applied to renew operating License Number R-76 (Docket number 50-27). As part of the license renewal process, Washington State University is submitting proposed Technical Specifications. The proposed Technical Specifications are included as an attachment with this letter.

Technical Specifications
This letter and the attached proposed Technical Specifications are intended to supersede all previous Technical Specifications and all previous letters submitted as requests for modifications of the Technical Specifications. The version of the Technical Specifications that is included with this letter differs from the version submitted on July 30, 2010 in only one respect: a new subsection, titled "5.9 Site and Facility Description" has been added. The pagination of those sections following the addition of subsection 5.9 has changed to reflect the insertion of additional text. The table of contents has been changed to include the addition of subsection 5.9, and also to accurately depict the change in pagination of sections and subsections following subsection 5.9. There have been no additions, deletions, or changes of any kind to any other section of the Technical Specifications other than the insertion of subsection 5.9 and the consequent change in the table of contents and pagination.

Pool level monitoring
This discussion is intended to supplement the letter of April 7, 2010 (ADAMS Accession # ML101031097) in which the function of the pool level monitoring system was described. As described in the letter of April 7, 2010, there are three float switches that perform functions as the pool water level changes due to evaporation or addition of makeup water. The upper switch serves to trigger a high pool water level alarm. The middle switch activates and deactivates the pool water makeup feed system. The lower switch triggers a low pool water level alarm. WSU is required by the U.S. NRC to test the function of the upper and lower switches on a monthly basis to confirm proper operability. The monthly operability check may not be made less frequent or deleted without prior approval by the U.S. NRC.

I declare under penalty of perjury that to the best of my knowledge the foregoing is true and correct.

Date Executed 8/10/2010

Respectfully Submitted,

Donald Wall

Donald Wall, Ph.D.
Director

Attachments

cc Region IV Office

A020
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FACILITY LICENSE NO. R-76

TECHNICAL SPECIFICATIONS

AND BASES

FOR THE

WASHINGTON STATE UNIVERSITY

MODIFIED TRIGA REACTOR

DOCKET NO. 50-27

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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 information purposes only and they do not constitute limitations or requirements to which the licensee must adhere. Furthermore, the dimensions, measurements, and other numerical values given in these specifications may differ slightly from actual values because of normal construction and manufacturing tolerances, or normal degree of accuracy or instrumentation.

1.0 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.

Abnormal Occurrence: An abnormal occurrence is defined for the purposes of the reporting requirements of Section 208 of the Energy Reorganization Act of 1974 (PL 93-438) as an unscheduled incident or event which the Nuclear Regulatory Commission determines is significant from the standpoint of public health or safety.

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.

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

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 below 40 °C.

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

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

(1) Regulating Rod: The regulating rod is a low-worth (low reactivity) control rod used primarily to maintain an intended power level that need not have scram capability. Its position may be varied manually or by means of an electric motor-operated positioning.

(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.

(3) Standard Control Rod: Standard control rod shall mean any control rod that has a scram capability, that is utilized to vary the reactivity of the core, and that is positioned by means of an electric motor-operated positioning system.

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 H 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 that amount of reactivity that would exist if all reactivity control devices were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions or at a specified set of conditions.

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

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

(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 must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

(3) Unsecured Experiment: An unsecured experiment is an experiment that does not meet the definition of a Secured Experiment.

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.

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

Irradiation: Irradiation shall mean the insertion into an irradiation position of a device or material that is not a normal part of the reactor core or experimental facilities so that the device or material is exposed to a significant amount of radiation available in the irradiation position. The following limitations also apply:

(1) meets all the criteria of an experiment

(2) maximum dose equivalent rate of 10 rem/hour at 30 cm

(2) encapsulation in a suitable container

(3) reactivity worth less than $0.25

(4) maximum residence time in the reactor irradiation position of 15 days

Irradiation Facilities: Any in-pool experimental facility that is not a normal part of the core and that 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.

Licensee: An individual or organization holding a license.

Limiting Safety Systems Setting: Limiting safety systems settings are the settings for automatic protective devices that monitor parameters with 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 LEU-type 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 that is outside the site boundary.

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

Operable: Operable means 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 at the maximum licensed power level and that satisfies all the requirements of the Technical Specifications.

Owner or operator: See licensee.

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.

Reactivity worth of an experiment: The reactivity worth of an experiment is the value of the reactivity change that results from the experiment being inserted into or removed from its intended 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, that 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 minimum number of neutron-absorbing control devices is fully inserted or other safety devices are in shutdown position, as required by technical specifications;

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

(c) 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;

(d) No experiments are being moved or serviced that have a reactivity worth exceeding the maximum value allowed for a single experiment, or $1.00, whichever is smaller.

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 and the following conditions exist:

(a) No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless the control rod drives are physically decoupled from the control rods;

(b) No experiments are moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or $1.00, whichever is smaller.

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

Research Reactor: A research reactor is defined as a device designed to support a self-sustaining

neutron chain reaction for research, development, educational, training, or experimental purposes and that may have provisions for the production of radioisotopes.

Research Reactor Facility: Includes all areas within which the owner or operator directs authorized activities associated with the reactor.

Reportable Occurrence: A reportable occurrence is any of the following events that occur during reactor operation:

(1) operation with any safety system setting less conservative than specified in Section 2.2, "Limiting Safety System Settings"

(2) operation in violation of a limiting condition of operation listed in Section 3.0

(3) operation with a required reactor or experiment safety system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function. No report is required if the condition arises due to or during the course of performing maintenance activities

(4) an unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips from known or identifiable causes are excluded

(5) an observed inadequacy in the implementation of either administrative or procedural controls, to such degree that the inadequacy causes or could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits

(6) release of fission products into the environment

(7) abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks which do not create a potential to exceed 10 CFR 20 release limits)

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 on parameters that are necessary to protect the integrity of the physical barriers which 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 a control or safety device.

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 nonscrammable rod(s) in the most reactive position and that the reactor will remain subcritical without further reactor operator action.

Site Boundary: The site boundary is 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.

Standard Core: A standard core is any arrangement of all-Standard TRIGA fuel.

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 steady-state 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.

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

2.0 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 LEU-type TRIGA fuel rod shall not exceed 1150 °C under any condition of operation.

Bases: 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. The safety limit for the 30/20 LEU fuel is based on data that indicate that 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. The safety limit for the Standard TRIGA fuel is based on data, including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding 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. Further information may be found in the document GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, submitted under docket number 50-227.

2.2 Limiting Safety System Settings

Applicability: This specification applies to the settings that prevent the safety limit from being reached.

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

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. For a mixed core, the instrumented rod shall be located in one of the following core lattice positions in the region of the core containing the 30/20 LEU-type 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.

Bases: 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 limit 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 LEU fuel region to less than 800 °C. This setting provides at least a 350 °C margin for 30/20 LEU fuel and at least a 200 °C margin for Standard fuel.

In the pulse mode of operation, the same limiting safety system setting will apply. 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 width 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.

3.0 LIMITING CONDITIONS OF OPERATION

3.1 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 performed by General Atomics indicate that TRIGA fuel may be safely operated up to power levels of at least 2.0 MW with adequate convective heat transfer due to natural convection.

3.2 Reactivity Limitations

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) the highest worth unsecured experiment in its most reactive state

(2) the highest worth scrammable control rod and all non-scrammable control rods fully withdrawn

(3) the reactor in the cold critical condition without xenon

Basis: The value of the shutdown margin ensures that the reactor can be shut down from any operating condition even if the highest worth scrammable control rod should remain in the fully

withdrawn position. Control rods that are not scrammable are not used to determine the shutdown margin.

3.3 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.

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

Basis: TRIGA fuel is fabricated with a nominal hydrogen to zirconium ratio of 1.6 for 30/20 LEU 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. The 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.

3.4 Maximum Excess Reactivity

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

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

Specifications: The maximum excess reactivity under conditions of cold, critical ($k_{eff} = 1$) xenon-free 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, that specification does not address the total reactivity available within the core. This specification, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures, and temperature peaks are maintained within the evaluated safety limits. The specified excess reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility.

3.5 Core Configuration Limitation

Applicability: This specification applies to mixed cores of 30/20 LEU and Standard types of fuel.

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

Specifications:

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

(2) Each new mixed core configuration shall be evaluated to determine the allowed locations for the IFE.

Bases: The limitation on the allowable core configuration of the 30/20 LEU 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.

3.6 Reactor Control and Safety System

3.6.1 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 moment that a safety channel parameter reaches the limiting safety system setting.

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

Specifications: The reactor shall not be operated unless the control rods are operable.

The scram time from the time that a safety system setting is exceeded to the time that the slowest scrammable control rod reaches its fully inserted position shall not exceed 2 seconds. For purposes of this section, this specification shall be considered to be satisfied when the sum of the

response time of the slowest responding safety channel plus the time required for the slowest scrammable control rod to reach its fully inserted position is less than or equal to 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 for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to ensure the safety of the reactor.

3.6.2. Reactor Control System

Applicability: This specification applies to the information that must 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 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

Bases: 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.

3.6.3 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 must 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

		Number operable in specified mode	
Safety Channel	Function	Steady State	Pulse
Fuel temperature	Scram if fuel temperature exceeds 500 °C	1	1
Power level	Scram if power level exceeds 125% of full licensed power	1	
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

Table 3.3 Minimum Interlocks

		Number operable in specified mode	
Interlock	Function	Steady State	Pulse
1 kW pulse interlock	Prevent initiation of a pulse above 1 kW		1
Wide range log power level channel	Prevent control rod withdrawal when neutron count is less than 2 counts per second	1	
Pulse-mode switch	Prevent withdrawal of standard control and regulation rods in pulse mode		1
Pool level	Alarm if pool level falls below 16 ft over the core	1	1
Transient rod control	Prevent application of air pressure unless fully inserted	1	

Bases: 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.

The interlock to prevent startup of the reactor with neutron count rate of less than 2 counts per second 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 or regulating control rods in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The pool level alarm is intended to alert the operator to a significant decrease in the pool level. The transient rod control interlock is to prevent rapid insertion of reactivity when the reactor is in the steady state mode.

3.7 Radiation Monitoring System

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

Objective: The objective is to 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*	Function	Number
Area radiation monitor	Monitor radiation level on the bridge	1
Area radiation monitor	Monitor radiation level in the beam room	1
Continuous air monitor	Monitor the radiation level in the pool room air	1
Exhaust gas monitor	Monitor the argon-41 radioactivity in the exhaust	1

*For periods of time 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.

3.8 Argon-41 Discharge Limit

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

Objective: To ensure that the health and safety of the public are not endangered by the discharge of ^{41}Ar from the WSU TRIGA reactor facility.

Specification: The concentration of ^{41}Ar in the effluent gas from the facility, as diluted by atmospheric air in the lee of the facility as a result of the turbulent wake effect, shall not exceed 1×10^{-8} μCi/mL averaged over one year.

Basis: The maximum allowable concentration of ^{41}Ar in air in unrestricted areas as specified in Appendix B, Table II of 10 CFR 20 is 1×10^{-8} μCi/mL. Section 6.5 of the safety analysis report for conversion of the WSU TRIGA reactor to FLIP fuel substantiates a 3.4×10^{-3} atmospheric dilution factor for a 4.4 mph wind speed. A somewhat more conservative value of 4×10^{-3} has been selected for the calculation of ^{41}Ar dilution.

3.9 Engineered Safety Feature - 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: The reactor shall not be operated unless the facility ventilation system is operable, except for periods of time not to exceed 48 hours to permit repair or testing of the ventilation system. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be operated in the dilution or isolation mode to prevent the release of a significant quantity of airborne radioactivity from the facility.

Basis: During normal operation of the reactor and the ventilation system, the concentration of ^{41}Ar and other airborne radionuclides discharged from the facility is below the applicable maximum air effluent concentration (AEC) values. In the event of a substantial release of airborne radioactivity within the facility, the ventilation system will be operated in the dilution or isolation mode as required by the situation. This action will minimize the concentration of airborne radioactive materials discharged to the environment until the concentration is below the appropriate AEC value. In addition, 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.

3.10 Limitations on Experiments

Applicability: This specification applies to experiments installed in the reactor and its experimental facilities (defined in Section 1.0).

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) Nonsecured experiments shall have reactivity worths less than $1.00.

(2) The reactivity worth of any single experiment shall not exceed $2.00.

(3) Total worth of all experiments shall not exceed $5.00.

(4) Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 mg shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 mg may be irradiated in the reactor or experimental facilities, provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than half the design pressure of the container.

(5) Experiment 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, or (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, the airborne concentration of radioactivity averaged over a year would not exceed the applicable limits of Appendix B of 10 CFR 20.

In calculations pursuant to item 5 above, the following assumptions shall be used:

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

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

c) For materials whose boiling point is above 60 °C and in cases where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors can escape.

d) An atmospheric dilution factor of 4×10^{-3} for gaseous discharges from the facility.

(6) Each fueled experiment shall be controlled so that the total radioactive inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 Ci.

(7) If a capsule fails and releases material that could damage the reactor fuel or structure by corrosion or other means, that material shall be removed and physically inspected to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by a senior reactor operator and must be determined to be satisfactory before operation of the reactor is resumed.

Bases:

(1) This specification is intended to provide assurance that the worth of a single unsecured experiment will be limited to such a value that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted.

(2) The maximum worth of a single experiment is limited so that its addition to or removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. Since an experiment of such worth must be fastened in place, its removal from the reactor operating at full power would result in a relatively slow power increase so that the reactor protective systems would act to prevent power levels from exceeding the safety limits.

(3) The total worth of all experiments is limited to ensure that the reactor will remain subcritical in the event of a simultaneous removal of all of the experiments with one control rod withdrawn.

(4) This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials.

(5) This specification is intended to reduce the likelihood that radioactive airborne particles in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere outside the facility.

(6) The 1.5-Ci limitation on iodine isotopes 131 through 135 ensures 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.

(7) Operation of the reactor with the reactor fuel or structure damaged is prohibited (to avoid release of fission products).

3.11 Limitations on Irradiations

Applicability: This specification applies to irradiations performed in the irradiation facilities contained in the reactor pool as defined in Section 1.0,"Irradiation Facilities." Irradiations are a subclass of experiments that fall within the specifications stated in this section. The surveillance requirements for irradiations are given in Section 4.3.5(2).

Objective: The objective is to prevent damage to the reactor, excessive release of radioactive materials, or excessive personnel radiation exposure during the performance of an irradiation.

Specifications: A device or material shall not be irradiated in an irradiation facility under the classification of an irradiation unless all the following conditions exist:

(1) The irradiation meets all the specifications of Section 3.10 for an experiment.

(2) The expected radiation field produced in air by the device or sample upon removal from the reactor pool is less than 10 rem/hr beta and gamma equivalent at 1 ft; otherwise, it shall be classified as an experiment.

(3) The device or material is encapsulated in a suitable container.

(4) The reactivity worth of the device or material is $0.25 or less; otherwise, it shall be classified as an experiment.

(5) The device or material does not remain in the reactor for more than a 15-day period; otherwise, it shall be classified as an experiment.

Basis: This specification is intended to provide assurance that the special class of experiments called irradiations will be performed in a manner that will not permit any safety limit to be exceeded.

3.12 As Low As Reasonably Achievable (ALARA) Radioactive Effluent Releases

Applicability: This specification applies to the measures required to ensure that the radioactive effluents released from the facility are in accordance with ALARA criteria.

Objective: The objective is to limit the annual population radiation exposure to a small percentage of the normal local background exposure due to the operation of the WSU TRIGA reactor.

Specifications:

(1) In addition to the radiation monitoring specified in Section 5.4, an environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the environs of the facility on a quarterly basis.

(2) 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%.

(3) Whenever practicable, the reactor shall be operated 4 in. or more from the thermal column in order to minimize the production of ^{41}Ar.

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

(5) In the event of a significant fission product leak from a fuel rod or a significant airborne radioactive release from a sample being irradiated, as detected by the continuous air monitor, the reactor shall be 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 leakage.

(6) Before discharge, the facility liquid effluents collected in the holdup tanks shall be analyzed for their radioactivity content. The total annual quantity of radioactivity in liquid effluents released (above background) shall not exceed 1 Ci per year.

Basis: The simplest and most reliable method of ensuring that ALARA release limits are accomplishing their objective of minimal facility-caused radiation exposure to the general public is to actually measure the integrated radiation exposure in the environment on and off the site.

3.13 Primary Coolant Conditions

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

Objectives: The objectives are (1) to minimize the possibility for corrosion of the cladding on the fuel rods, and (2) to minimize neutron activation of dissolved materials.

Specifications:

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

(2) The pH of the pool water shall be between 5.0 and 7.5.

Basis: A small rate of corrosion continuously occurs in a water-metal system. In order to limit this rate and thereby extend the longevity and integrity of the fuel cladding, a water purification system is required. Experience with water quality control at many reactor facilities has shown that maintenance of water conductivity and pH within the specified limits provides acceptable control.

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This 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.

3.14 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 source or sources that are stored or used in the pool do not constitute a significant hazard to the operation of the reactor, 2) sealed radioactive source or sources do not create a significant environmental or

personal radiation exposure hazard, and 3) sealed radioactive source or 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 away from the face of an operating reactor core. The total radioactivity of all sealed sources stored in the pool shall not exceed 100,000 curies. All sealed source configurations shall be designed so that a loss of pool water accident will not cause a sealed source encapsulation integrity problem and the sources shall be stored in an appropriate shield so as not to produce a significant radiation hazard in the event of a loss of reactor pool water accident.

(2) All storage of sealed sources greater than 100 curies in the reactor pool shall be considered as an experiment and shall be reviewed and approved by the Reactor Safeguards Committee. A written operating procedure for the storage and use of sealed sources in the reactor pool shall be in effect.

(3) The radionuclide content of the reactor pool water shall be monitored monthly at an interval not to exceed six (6) weeks in order to detect a significant leak in the sources stored in the reactor pool. If the specific radionuclide content of the pool water for radionuclides from a sealed source stored in the reactor pool exceeds one-third (1/3) the 10 CFR 20 Appendix B, Table 3 value, steps shall be taken to isolate the source of the radioactivity and to mitigate the problem.

Basis:

(1) Limiting the proximity of sealed sources to five (5) or more feet away from the surface of the reactor core minimizes the effect of such sources on the reactor and the operation of the reactor upon the sources. The neutron flux at a distance of five (5) feet from the core surface is insignificant and thus could not cause activation of the sources and any associated shielding. The presence of the sources in the pool would have no impact upon the maximum hypothetical accident which is the rupture of the cladding on one fuel rod. However, the presence of sources in the pool could contribute to the radiation hazard associated with a loss of pool water accident. The dose rate 25 feet above an unshielded core in the event of a loss of pool water accident would only 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.

(2) Classifying the storage of sealed sources in the reactor pool as an experiment mandates that such storage be reviewed by the Reactor Safeguards Committee.

(3) The 10 CFR 20 Appendix B, Table 3 limit for ^{60}Co is 3×10^{-5} μCi/mL. At this limit the entire pool could be dumped into the WSU sewage system without taking advantage of the dilution factor associated with the discharge volume of the WSU sewage system. The detection limit for ^{60}Co in the reactor pool water depends upon the system used but in the worst case would be at least 1×10^{-7} μCi/mL, or 100 pCi/mL, or about one-three-hundredth of the 10 CFR 20 limit stated above. Setting a limit of 100 times the detection

limit and one-third the discharge limits provides the facility with ample time to take corrective action in the event the limit is exceeded and does not compromise ALARA considerations.

3.15 Generation of Boron Neutron Capture Facility Beam

Definitions:

(1) For the purpose of this technical specification, the term "BNC facility" shall refer to the boron neutron capture facility which includes the beam, bridge moving system, beam monitoring equipment, beam shielding room, access gate and experimental area viewing equipment. The experimental bench, positioning equipment, and other equipment used for the beam targets are not considered part of the BNC facility for purposes of this provision, except insofar as radiation safety (i.e., activation and/or contamination) is concerned.

(2) The term "BNC experiment" shall refer to a boron neutron capture experiment involving the neutron irradiation of biological cells enriched with boron.

(3) The term "calibration check" refers to the process of checking the beam intensity and quality via one or more of the following: foil activation; use of a fission chamber; use of an ion chamber; or an equivalent process. The purpose of a calibration check is to ensure that the beam has not changed in a significant way (e.g., energy spectrum or intensity) from the beam that was characterized.

(4) The term "functional check of the beam monitors" shall consist of verifying that system output is consistent (± 10%) with previously measured values upon normalization to a common reactor neutronic power level.

(5) The term "characterization" refers to the process of obtaining the dose-versus-depth profile in phantoms. The dose-versus depth profile from the surface of the phantom to a depth at least equivalent to the total thickness of the target volume to be irradiated on a central axis is deemed adequate for a characterization. Fast neutron, thermal neutron, and gamma ray components are determined in a characterization and monitors are normalized by this characterization.

(6) The term "calibration of the beam monitors" refers to the process whereby the beam monitors are calibrated against instruments that measure dose including a tissue equivalent chamber and a graphite or magnesium wall ionization chamber (or the equivalent to any of these three) that have in turn been calibrated by a secondary calibration laboratory.

(7) The term "design modification" as applied to the BNC facility beam refers (a) to a change that is shown to alter the dose-versus-depth profile of the fast neutrons, thermal neutrons, or gamma rays in the beam as sensed by the calibration check and (b) to a change that has the potential to increase significantly the amount of activation products in the BNC facility.

(8) The term "radiation fluence" means the total fluence of neutrons and gamma radiation that is emitted in the BNC facility beam. The determination of the ratios of gamma, fast neutron, and thermal neutron fluences is part of the beam characterization. Knowledge of these ratios allows the total radiation fluence to be monitored by the on-line detectors, which are neutron sensitive. Compliance with the limits specified on radiation fluence by this specification is determined by reference to the fluence monitored by these detectors.

Applicability: This specification applies solely to the generation of the BNC facility beam for BNC experiments. It does not apply to any other use of the BNC facility and/or its beam. Surveillances listed in this specification are required only if BNC experiments are planned for the interval of the surveillance. However, in the event of a hiatus in the scheduled performance of any given surveillance, that surveillance shall be performed prior to the initiation of BNC experiments during the interval in question.

Objective: To acquire testing and operational experience in use of a facility developed specifically for Boron Neutron Capture Technology.

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 the access gate located at its entrance.

(3) The following features and/or interlocks shall be operable:

(a) An interlock shall prevent moving the bridge from the retracted position unless the BNC facility access gate is closed.

(b) The reactor shall scram and the bridge shall move to the retracted position automatically upon opening the treatment room access gate.

(c) The bridge shall be designed to move to the retracted position automatically upon failure of facility electric power or low voltage on the backup batteries that power the bridge motor.

(d) Bridge movement that controls beam delivery shall be designed for manual movement to the retracted position.

(e) It shall be possible to move the bridge to the retracted position from within the BNC facility.

(f) A BNC facility lockdown near the access gate shall inhibit blade withdrawal when the key is not inserted and turned to the locked position.

(4) Bridge shall be equipped with a position readout that indicates the status of the bridge. A bridge position readout shall be visible at the local control panel of the BNC facility. In the event of a bridge position readout malfunction, it shall be acceptable to use an alternate means of verifying position such as a video camera in the pool room providing a signal to a monitor at the local control panel of the BNC facility. Use of this alternate means of bridge position verification is limited to seven consecutive working days.

(5) The BNC facility shall be equipped with a read out display of the reactor log-power and the linear power on the BNC facility control console just outside of the shielding.

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

(a) This radiation monitor shall be equipped with a backup power supply such as the reactor emergency power system or a battery.

(b) This radiation monitor shall be checked for proper operation by means of a check source on the calendar day of and prior to any BNC experimentation.

(c) This radiation monitor shall be calibrated quarterly.

(d) The audible alarm shall be set at or below 50 mR/hr. This monitor and/or its alarm may be disabled once the BNC room has been searched and secured, such as is done immediately prior to initiation of BNC experimentation. If this is done, the monitor and/or its alarm shall be interlocked so that they become functional upon opening of the BNC facility access gate.

(e) In the event that this monitor is inoperable, personnel entering the BNC facility shall use either portable survey instruments or audible alarm personal dosimeters as a temporary means of satisfying this provision. These instruments/dosimeters shall be in calibration as defined by the radiation protection program of the WSU research reactor facility and shall be source-checked daily prior to use on any day that they are used to satisfy this provision. Use of these instruments/dosimeters as a temporary means of satisfying this provision is limited to seven consecutive working days.

(7) An intercom or other means of two-way communication shall be operable both 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 shielding.

(8) It shall be possible for personnel monitoring a BNC experiment to open the BNC facility access gate manually.

(9) It shall be possible to observe the BNC experiment by means of two independent closed-circuit TV cameras. Both cameras providing visualization shall be operable at the outset of any BNC experiment. Should either fail during the irradiation, the experiment may be continued at the discretion of the experimenter. Adequate lighting to permit such viewing shall be assured by the provision of emergency lighting and backup power for one TV camera and monitor.

(10) The following interlocks or channels shall be tested at least monthly and prior to a BNC experiment if the interlock or channel has been repaired or deenergized:

	Interlock or Channel	Surveillance
a)	The reactor scrams and the bridge retracts upon BNC facility scram	Scram test
b)	Bridge will not move from the retracted position unless access gate is closed	Operational test
c)	Upon opening the BNC room access gate the reactor scrams and the bridge moves to the retracted position	Operational test
d)	The bridge moves toward the retracted position on loss of electrical power and low voltage on the bridge motor batteries	Operational test
e)	Manual movement of bridge	Operational test
f)	Bridge can be moved manually by someone standing on the reactor bridge	Operational test
g)	Bridge position indicator and status lights	Operational test
h)	Radiation monitor alarm	Operational test
i)	Radiation monitor and/or alarm enabled upon opening of shield door	Operational test
j)	Intercoms	Operational test
k)	BNC facility TV cameras, monitors and its power backup	Operational test
l)	BNC facility emergency lighting	Operational test
m)	BNC facility lockdown blade inhibit	Operational test

In addition to the above, the BNC facility scram shall be tested prior to reactor startup if the reactor has been shut down for more than sixteen hours.

(11) Manual operation of the access gate of the BNC facility in which the door is opened fully shall be verified semi-annually.

(12) Use of the BNC facility beam shall be subject to the following:

(a) A calibration check of the beam and a functional check of the beam monitors shall be made weekly for any week that the beam will be used for BNC experiments. These checks shall be made prior to any BNC experiment for a given week. In addition, a calibration check shall be performed prior to any BNC experiment in the event that any component of a given beam design has been replaced. Finally, a calibration and a functional check shall be performed prior to any BNC experiment in the event of a design modification.

(b) A characterization of the beam shall be performed every six months for any six-month interval that the beam will be used for BNC experiments. This six-month characterization shall be made prior to any BNC experiment for a given six-month interval. A characterization shall also be performed prior to any BNC experiment in the event of a design modification. As part of the characterization process, the proper response of the beam monitors shall be verified.

(c) A calibration of the beam monitors shall be performed at least once every two years for any two-year interval that the beam will be used for BNC experimentation. The two-year calibration shall be made prior to any BNC experimentation during any given two-year interval.

(d) A scram from full power initiated when the reactor is positioned against the BNC facility filter shall be performed every six months or in the event of a design modification. The BNC room radiation monitor reading shall not exceed 50 mR/hr, 30 seconds after the initiation of the scram and bridge retraction.

(13) Maintenance, repair, and modification of the BNC facility shall be performed under the supervision of a senior reactor operator who is licensed by the U.S. Nuclear Regulatory Commission to operate the WSU Research Reactor. All modifications will be reviewed pursuant to the requirements of 10 CFR 50.59.

(14) Personnel who are not licensed to operate the WSU Research Reactor but who are responsible for either the BNC or the beam design including construction and/or modification may operate the controls for the BNC facility beam provided that:

(a) Training has been provided and proficiency satisfactorily demonstrated on the design of the facility, its controls, and the use of those controls. Proficiency shall be demonstrated annually.

(b) Instructions are posted at the local control panel of the BNC facility that specifies the procedure to be followed:

(i) to ensure that only the appropriate target is in the irradiation facility before turning the primary beam of radiation on to begin an irradiation;

(ii) if the operator is unable to turn the primary beam of radiation off with controls outside the BNC facility, or if any other abnormal condition occurs. A directive shall be included with these instructions to notify the reactor console operator in the event of any abnormality.

(c) In the event that bridge movement affects reactivity, 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 operator immediately prior to such action. Emergency scrams causing a bridge retraction are an exception and may be made without first requesting permission.

Records of the training provided under subparagraph (a) above shall be retained in accordance with the WSU Research Reactor training program or at least for three years. A list of personnel so qualified shall be maintained in the reactor control room.

Basis: The requirement that it be possible to initiate a scram from a control panel located in the BNC facility area assures the experimenter of the capability to terminate the irradiation immediately should the need arise. The provision that access to the BNC facility be limited to a single gate ensures that there will be no inadvertent entries. The various interlocks for the bridge movement system that controls beam delivery ensure that exposure levels in the BNC facility will be minimal prior to entry by personnel. The bridge position indicator and status lights serve to notify personnel of the beam status. The provision for a radiation monitor ensures that personnel will have information available on radiation levels in the BNC facility prior to entry. The purpose of the monitor audible alarm is to alert personnel to the presence of elevated radiation levels. This monitor and/or its alarm may be disabled once the BNC facility has been searched and secured so that it will not distract attending personnel. The monitor and/or its alarm are interlocked with the access gate so that they are made functional upon opening that gate, and hence prior to any possible entry to the BNC facility. One intercom provides a means for the prompt exchange of information between the experimenter(s) and the reactor operator(s).

The provision for manual operation of the access gate of the BNC facility ensures access to the experimental area in the event of a loss of electrical power. The closed-circuit TV cameras provide the experimenter(s) with the opportunity to monitor the target area visually as well as through the use of various instruments. The emergency lighting and the backup power for a TV camera and monitor will permit visual surveillance of the target area in the event of a power failure.

The surveillance requirements for beam calibration checks and characterizations provide a mechanism for ensuring that the BNC facility and its beam will perform as originally designed. Similarly, the surveillance requirements on the beam monitors ensure that these instruments are calibrated by a means traceable to the National Institute of Standards and Technology. The

chambers specified (tissue-equivalent, and graphite or magnesium-wall) were chosen because they measure dose as opposed to fluence.

The specifications on maintenance and repair of the BNC facility ensures that all such activities are performed under the supervision of personnel cognizant of quality assurance and other requirements such as radiation safety. The provision on the training and proficiency of non-licensed personnel ensures that all such personnel will receive instruction equivalent to that given to licensed reactor operators as regards use of the BNC facility beam. (Note: Licensed reactor operators may, of course, operate the BNC facility beam.) Also, this provision provides for the posting of instructions to be followed in the event of an abnormality.

4.0 SURVEILLANCE REQUIREMENTS

4.1 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 mechanism, or the reactor safety system 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. 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.

4.2 Safety Limit - Fuel Rod Temperature

Applicability: This specification applies to the surveillance requirements of the fuel rod temperature measuring channel.

Objective: The objective is to ensure that the fuel rod temperatures are properly monitored.

Specifications:

(1) Whenever a reactor scram caused by high fuel rod temperature occurs, the peak indicated fuel temperature shall be examined to determine whether the fuel rod temperature safety limit was exceeded.

(2) The fuel rod temperature measuring channel shall be calibrated semiannually or at an interval not to exceed 8 months by the substitution of a thermocouple simulator in place of the instrumented fuel rod thermocouple.

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

Basis: Operational experience with the TRIGA system gives assurance that the thermocouple measurements of fuel rod temperature have been sufficiently reliable to ensure accurate indication of this parameter.

4.3 Limiting Conditions for Operation

4.3.1 Reactivity Requirements

Applicability: These specifications apply to the surveillance requirements for reactivity control of experiments and systems.

Objective: The objective is to measure and verify the worth, performance, and operability of those systems affecting the reactivity of the reactor.

Specifications:

(1) The reactivity worth of each control rod and the shutdown margin shall be determined annually but at intervals not to exceed 15 months.

(2) The reactivity worth of each control rod and the shutdown margin shall be determined after significant reactor core changes, significant control rod changes, or after a significant change to the transient rod and transient rod drive system.

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

(4) The control rods shall be visually inspected for deterioration at biennial intervals.

(5) The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated semiannually at intervals not to exceed 7.5 months.

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

Basis: The reactivity worth of the control rods is 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. Past experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to ensure no significant changes in the shutdown margin. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation in the reactor. The reactor is pulsed at suitable intervals and a comparison is made with previous similar pulses to determine if changes in fuel or core characteristics are taking place.

4.3.2 Control and Safety System

Applicability: These specifications apply to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

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

Specifications:

(1) The scram time shall be measured annually but at intervals not to exceed 15 months.

(2) A channel check of each of the reactor safety system channels for the intended mode of operation shall be performed before each day's operation or before each operation extending more than 1 day, except for the pool level channel which shall be tested monthly.

(3) A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually, but at intervals not to exceed 15 months.

(4) A channel test of each item in Table 3.2 and Table 3.3, other than measuring channels, shall be performed semiannually, but at intervals not to exceed 7.5 months.

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

Basis: Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly. The channel tests will ensure that the safety system channels are operable on a daily basis or before extended operation of the reactor. The power level channel calibration will ensure that the reactor will be operated at the proper power levels. Transient control rod checks and semiannual maintenance ensure proper operation of this control rod.

4.3.3 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: All radiation monitoring systems shall be verified to be operable at least monthly at an interval not to exceed 45 days. In addition, the following surveillance activities shall be performed on an annual basis at intervals not to exceed 15 months: 1) the area radiation monitoring system shall be calibrated using a certified source; 2) a calibration of the Ar-41 system shall be done using at least two different calibrated gamma-ray sources; 3) a calibration shall be performed on the CAM in terms of counts per unit time per unit of radioactivity using

calibrated beta-particle emitting sources.

Basis: Experience has shown that monthly verification of radiation monitoring systems operability in conjunction with an annual surveillance is adequate to correct for variations in the systems caused by a change of operating characteristics over time.

4.3.4 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; the isolation and dilute modes would be used to control the release of radioactive material to the environment in the event of an emergency.

Specifications: The operation of the pool room ventilation system shall be checked monthly (at intervals not to exceed 6 weeks) 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. The pressure drop across the absolute filter in the pool ventilation system shall be measured at least twice a year. The absolute filter shall be changed whenever the pressure drop across the filter increases by 1 in. of water.

The air flow rates in the ventilation system shall be measured biennially, at intervals not to exceed 30 months.

Basis: The most reliable method of testing the ventilation system is to cycle the system into the various operational modes and visually check each portion of the system for proper operation.

4.3.5 Experiment and Irradiation Limits

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

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

Specifications:

(1) A new experiment shall not be installed in the reactor or its experimental facilities until a hazards analysis has been performed and reviewed for compliance with "Limitations on Experiments," Section 3.10, by the Reactor Safeguards Committee. Minor modifications to a reviewed and approved experiment may be made at the discretion of a senior reactor operator, 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 original approved experiment.

(2) An irradiation of a new type of device or material shall not be performed until an analysis of the irradiation has been performed and reviewed for compliance with "Limitations on Irradiations," Section 3.11, by a licensed senior reactor operator qualified in health physics, or a licensed senior reactor operator and a person qualified in health physics.

Basis: It has been demonstrated over a number of years that experiments and irradiations reviewed by the reactor staff and the Reactor Safeguards Committee, as appropriate, can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

4.4 Reactor Fuel Rods

Applicability: This specification applies to the surveillance requirements for the fuel rods.

Objective: The objective is to verify the continuing integrity of the fuel rod cladding.

Specifications: All fuel rods shall be inspected visually for damage or deterioration and measured for length and bend at intervals not to exceed the sum of $3,500.00 in pulse reactivity. The reactor shall not be operated with damaged fuel. A fuel rod shall be considered damaged and must be removed from the core if:

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

(2) the elongation exceeds the original length of the fuel rod by 0.125 in.

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

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

The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel rods shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel rod bowing that could result in touching has occurred without deleterious effects. 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.

4.5 Primary Coolant Conditions

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

Objective: The objective is to ensure that water quality does not deteriorate over extended periods of time if the reactor is not operated.

Specification: The conductivity and pH of the primary coolant water shall be measured at least once every 2 weeks, and shall be as follows:

(1) conductivity $\leq 5 \times 10^{-6}$ mhos/cm

(2) pH between 5.0 and 7.5

The radionuclide content of the reactor pool water shall be monitored monthly at an interval not to exceed six (6) weeks.

Basis: Section 3.13 ensures that the water quality is adequate during reactor operation. Section 4.5 ensures that water quality is not permitted to deteriorate over extended periods of time even if the reactor does not operate.

5.0 DESIGN FEATURES

5.1 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 of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications:

(1) 30/20 LEU; the individual unirradiated 30/20 LEU fuel rods shall have the following characteristics:

- uranium content: maximum of 30% by weight uranium, enriched to a maximum of 19.95% ^{235}U with a nominal enrichment of 19.75% ^{235}U.
- hydrogen-to-zirconium ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65
- natural erbium content (homogeneously distributed): nominal 0.90% by weight
- cladding: 304 stainless steel, nominal thickness 0.020 inches

(2) Standard TRIGA fuel; the individual unirradiated standard TRIGA fuel rods shall have the following characteristics:

- uranium content: maximum of 9.0% by weight enriched to less than 20% ^{235}U
- hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.5 and 1.8

- cladding: 304 stainless steel, nominal thickness 0.020 inches

Basis: The fuel specification permits a maximum uranium enrichment of 19.95% 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 must not deviate from the design value by more than -3.3%). This variation for a single fuel element would result in the 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.

5.2 Reactor Core

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

Objective: The objective is to ensure that provisions are made to restrict the arrangement of fuel rods and experiments so as 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 standard fuel, 30/20 LEU fuel, or a combination of standard and 30/20 LEU (mixed cores) provided that the 30/20 LEU fuel region contains at least 51 30/20 fuel rods located in a contiguous block in the central region of the core.

(3) The reactor fueled with a mixture of fuel types shall not be operated with a core lattice position vacant in the 30/20 LEU fuel region. Water holes in the 30/20 LEU region shall be limited to single-rod holes. Vacant lattice positions in the core fuel region shall be occupied with fixtures that will prevent the installation of a fuel assembly.

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

Bases: Standard TRIGA cores have been used for years and their characteristics are well-documented. Mixed cores of 30/20 LEU 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 may be safely operated.

In mixed cores, it is necessary to arrange 30/20 LEU fuel rods in a contiguous, central region of the core to control flux peaking and power generation peak values in individual fuel rods.

Vacant core lattice positions in the standard fuel region will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. Vacant core positions are not permitted in the 30/20 LEU fuel region.

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.

5.3 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 are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specifications:

(1) The standard control rod shall have scram capability and contain borated graphite, B_4C powder, or boron and its compounds in solid form as a poison in aluminum or stainless steel cladding.

(2) The regulation control rod need not have scram capability and shall be stainless steel or contain the materials as specified for standard control rods.

(3) The transient control rod shall have scram capability and contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient rod shall have an adjustable upper limit to allow a variation of reactivity insertions. This control rod may incorporate a nonfueled follower.

Basis: The poison requirements for the control rods are satisfied by using neutron-absorbing borated graphite, B_4C powder, or boron and its compounds. Since the regulating rod normally is a low worth control rod, its function could be satisfied by using solid stainless steel. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water. Scram capabilities are provided for rapid insertion of the control rod which is the primary safety

feature of the reactor. The transient control rod is used to produce a reactor pulse. The nuclear behavior of the nonfueled follower which may be incorporated into the transient control rod is similar to a void.

5.4 Radiation Monitoring System

Applicability: This specification describes the functions and essential components of the area radiation monitoring equipment and the system for continuously monitoring airborne radioactivity.

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

Specifications:

(1) Function of Area Radiation Monitor (gamma radiation sensitive instruments): Monitor radiation fields in key locations, alarm and readout at the reactor control console.

(2) Function of Continuous Air Radiation Monitor (beta radiation, gamma radiation sensitive detector with particulate collection capability): Monitor radioactive particulate radioactivity in the pool room air, alarm and readout at the reactor control console.

(3) Function of argon-41 Air Exhaust Monitor (gamma radiation sensitive detector): Monitor ^{41}Ar content in ventilation system exhaust air, alarm and readout at the reactor control console.

Basis: The radiation monitoring system is intended to provide information to operating personnel of 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.

5.5 Fuel Storage

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

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

Specifications:

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

(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 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 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 minimum of 20 ft from ground level.

(3) Emergency shutdown controls for the ventilation system shall be located outside the pool and control room areas and the system shall be designed to shut down in the event of a substantial release of airborne radioactivity within the facility.

(4) The pool room ventilation system shall have a dilution mode of operation in which air from the pool room is mixed and diluted with outside air before being discharged from the facility when the ventilation system is operated in the dilution mode.

Basis: The facility is designed so that the ventilation system will normally maintain a negative pressure with respect to the atmosphere to minimize uncontrolled leakage to the environment. The free air volume within the reactor building is confined when the ventilation system is operated in the isolation mode. Emergency controls for startup, isolation, dilution, and normal operation of the ventilation system are located external to the control and pool rooms. Proper handling of airborne radioactive materials (in emergency situations) can be managed with a minimum of exposure to personnel.

5.7 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 a loss of coolant if the pool level drops more than 2 feet below the normal level.

(4) The reactor shall not be operated with less than 15 feet of water above the top of the core.

Basis: This specification is based on thermal hydraulic calculations which show that the TRIGA-30/20 LEU core can operate in a safe manner at power levels up to 2000 kW with natural convection flow of the coolant water. A comparison between operation of the TRIGA-30/20 LEU and standard TRIGA MARK III has shown them to be safe for the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as those for TRIGA-30/20 LEU and standard cores.

In the event of accidental siphoning of pool water through system pipes, the pool water level will drop no more than 5 ft from the top of the pool.

Loss of coolant requires corrective action. The loss of coolant alarm activates in the reactor control room and at an off-site alarm monitoring station.

5.8 Physical Security

The Licensee shall maintain in effect and fully implement all provisions of the NRC staff-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.70, collectively titled,"Washington State University, Pullman, Washington TRIGA Reactor Security Plan."

5.9 Site and Facility Description

Applicability: This specification applies to the Washington State University TRIGA reactor site location and specific facility design features.

Objective: The objective is to specify the location of facility design features.

Specifications:

(1) The Washington State University research reactor shall be operated by the Washington State University Nuclear Radiation Center.

(2) The Dodgen Research Facility is the building that houses the Washington State University Nuclear Radiation Center. The Dodgen Research Facility is divided into two areas:

(a) Controlled Access Areas
(b) laboratories and office space that are not Controlled Access Areas

(1) Controlled Access Areas shall be building locations to which unescorted access is restricted. An individual must undergo a fingerprint-based criminal history check and trustworthiness and reliability determination before being allowed unescorted access to Controlled Access Areas. Controlled Access Areas include the following:

(a) the reactor pool room i.e. the room in which the reactor is located (also known as Room 201)
(b) all rooms that adjoin Room 201 which have unrestricted access to Room 201
(c) the beam room (also known as Room 2)
(d) the unirradiated fuel storage area

(3) The section of the building that houses the reactor shall be equipped with a ventilation system that is designed to exhaust air and other gases from the reactor pool room and the beam room and release them through an exhaust stack that is a least 20 feet above ground level.

(4) Emergency shutdown controls for the ventilation system shall be located in the reactor room; additional emergency shutdown controls may also be located elsewhere in the Dodgen Research Facility.

Basis: The Nuclear Radiation Center, reactor building, and site description are described in detail in the Washington State University Safety Analysis Report. The ventilation system is designed such that it will under normal operating conditions maintain a negative pressure relative to outside atmospheric pressure in the reactor pool room (Room 201) so there will be no uncontrolled or unmonitored leakage of the internal atmosphere to the external environment. All exhaust from the reactor pool room will exit through a monitored exhaust stack. The Controlled Access Area (CAA) ventilation system is a separate, stand-alone system that services only the CAA and does not receive input from the non-CAA sections of the Dodgen Research Facility. Controls for operation of the ventilation system are located in the reactor control room. A second set of emergency controls are located in reception area of the front office (Room 50) of the Dodgen Research Facility. Management of the ventilation system is normally done within the reactor control room; the second set of controls provides redundancy in case of in case of emergency.

6.0 ADMINISTRATIVE CONTROL

6.1 Responsibility and Organization

Applicability: This specification applies to the organizational structure as it applies to organizational levels and responsibilities of the licensee.

Objective: The objective is to define the organizational levels and responsibilities for persons who are involved in operation of the Washington State University research reactor.

Specifications: The following specific organizational levels and responsibilities shall exist:

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 each member of the operating staff shall be defined in writing. The following specific 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 reports to the Vice President for Research. The WSU Nuclear Radiation Center shall be under the control of the Director. The Director is responsible for ensuring that regulatory requirements and implementation are in accordance with requirements of the U.S. Nuclear Regulatory Commission and the Code of Federal Regulations and U.S. Federal, State of Washington, and Washington State University regulations.

(c) Reactor Supervisor (Level 3): The Reactor Supervisor reports to the Director of the Nuclear Radiation Center and is responsible for guidance, oversight, and technical support of reactor operations.

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

(e) Radiation Safety Committee: The WSU Radiation Safety Committee supports the University's teaching, research and outreach mission by administering a program that ensures the safe use of radioactive materials and radiation machines on the Pullman campus and other WSU sites around the state. The WSU Radiation Safety Committee reports to the WSU Vice President for Research. The Reactor Supervisor and Reactor Operating staff get communications from the Radiation Safety Committee either directly or through the Director of the Nuclear Radiation Center.

(f) The WSU 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.

Position vacancy: In all instances, responsibilities of one level may be assumed by higher levels or by alternates designated by a higher level, conditional upon meeting all qualification and licensing requirements for the position.

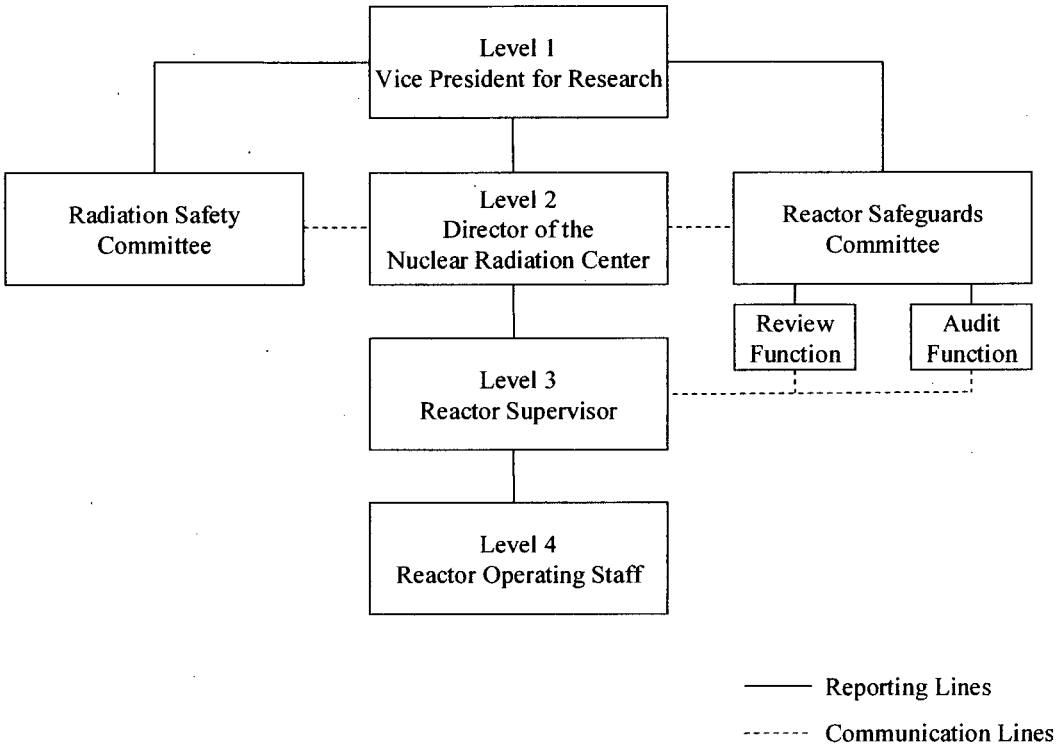

Figure 6.1 Facility organization

Bases: The President of Washington State University is appointed by the WSU Board of Regents, and is the chief officer of Washington State University. The WSU Nuclear Radiation Center is a department in the WSU Office of Research. The WSU Office of Research is headed by the WSU Vice President for Research, who reports to the President of WSU. The Director of the WSU Nuclear Radiation Center reports to the Vice President for Research.

The Nuclear Radiation Center shall be under the control of the Director of the WSU Nuclear Radiation Center. A licensed senior reactor operator may be designated as an alternate by the Director of the Nuclear Radiation Center or the Vice President for Research if the Director is absent or incapacitated. The Director shall be a licensed senior reactor operator, or in the case of absence, incapacitation, or vacancy of the position of Director, the Vice President for Research may designate the reactor supervisor or a senior reactor operator who is responsible for the safe operation and maintenance of the reactor and associated equipment. The Director shall also be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the license, Federal and State regulations, and requirements of the Reactor Safeguards Committee. This standard of responsibility applies at all times, including times when the Director is not on site, irrespective of the operational state of the reactor.

The Reactor Supervisor is responsible for oversight of reactor operations. The Reactor Supervisor shall be a licensed senior reactor operator, or in the event of a newly hired Reactor Supervisor, must obtain a senior reactor operator license at the earliest practicable opportunity.

A senior reactor operator is a reactor operator who is licensed to direct the activities of reactor operators. A senior reactor operator need not be present at the facility at all times.

6.2 Staffing

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

(a) a licensed reactor operator in the control room

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

(c) A designated senior reactor operator shall be readily available on call.

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 must be present as described in section 6.2(1)(b).

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 operator on duty informed of where he/she may be rapidly contacted and the telephone number, and

(c) is capable of getting to the reactor facility within a reasonable time under normal conditions (less than 30 minutes) and this person must remain within a 15 mile radius of the facility.

(2) Facility Staff Qualifications

Each member of the facility staff shall meet or exceed the minimum qualifications of ANS 15.4, "Standard for the Selection and Training of Personnel for Research Reactors," for comparable positions.

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

(a) management personnel

(b) radiation safety personnel

(c) other operations personnel

(4) Events requiring the presence at the facility of a senior reactor operator are:

(a) initial startup and approach to power

(b) all fuel or control rod relocations within the reactor core region

(c) relocation of an experiment with reactivity worth greater than $1.00

(d) recovery from unplanned or unscheduled shutdown or unplanned or unscheduled significant power reduction

6.3 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of American National Standard "Selection and Training of Personnel for Research Reactors" ANSI/ANS-15.4-1988 (R1999), Sections 4 through 7.

6.4 Reactor Safeguards Committee (RSC)

6.4.1 Function

The RSC shall function to provide an independent review and audit of the facility 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

The RSC is a WSU Presidential Committee. Members are appointed by the President of WSU. The President of WSU has delegated supervisory authority of the Reactor Safeguards Committee to the WSU Vice President for Research.

The RSC shall be composed of at least five members knowledgeable in fields that relate to nuclear reactor safety. The members of the Committee shall include one facility Senior Reactor

Operator who may be the Director of the Nuclear Radiation Center and WSU faculty and staff members designated to serve on the Committee in accordance with the procedures specified by the WSU committee manual. The University's Radiation Safety Director shall be an ex officio member of the Committee.

6.4.3 Operation

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

(1) meeting frequency: the full committee shall meet at least semiannually and a subcommittee thereof shall meet at least semiannually

(2) voting rules

(3) quorums: chairman or his designate and two 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 RSC or designated subcommittee thereof shall include, but are not limited to, the following:

(1) review and approval of all new experiments utilizing the reactor facility

(2) review and approval of all proposed changes to the facility license by amendment, and to the Technical Specifications

(3) review of the operation and operational records of the facility

(4) review of significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety

(5) review whether a proposed change, test, or experiment would constitute a change in the Technical Specifications or an unreviewed safety question

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

(7) review and approval of all standard operating procedures and changes thereto

(8) biennial review of all standard operating procedures, the facility emergency plan, and the facility security plan

(9) annual review of the radiation protection program

(10) review audit reports

6.4.5 Audits

The RSC or a subcommittee thereof shall audit reactor operations semiannually, but at intervals not to exceed 8 months. The semiannual audit shall include at least the following:

(1) review of the reactor operating records

(2) inspection of the reactor operating areas

(3) review of unusual or abnormal occurrences

(4) radiation exposures at the facility and adjacent environs

(5) facility operations

The RSC or a subcommittee shall audit the following at intervals of two years, not to exceed 30 months:

(1) Emergency Plan and implementing procedures

(2) Retraining and requalification program

(3) 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 within 30 days to the Vice President for Research and Director of the WSU Nuclear Radiation Center.

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

(3) The secretary of the RSC shall maintain a file of the minutes of all meetings.

6.5 Radiation Safety

The licensee, WSU, shall be responsible for implementation of a radiation safety program. The requirements for the radiation safety program are established in 10 CFR 20. The radiation safety program should use the guidelines of American National Standard "Radiation Protection at Research Reactor Facilities" ANSI/ANS-15.11-1993 (R2004).

6.6 Action To Be Taken in the Event a Safety Limit Is Exceeded

In the event a safety limit is exceeded:

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

(2) An immediate report of the occurrence shall be made by the Director of the Nuclear Radiation Center to the Chairman of the Reactor Safeguards Committee and to the Vice President for Research.

(3) A report shall be prepared that shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safeguards Committee for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

(4) A report shall be made to the U.S. NRC in accordance with Section 6.10 of these specifications.

6.7 Actions To Be Taken In The Event of a Reportable Occurrence Other Than A Safety Limit Violation

(1) The reactor shall be shut down and reactor operation shall not resume until authorized by the Director of the WSU Nuclear Radiation Center and the Chair of the WSU Reactor Safeguards Committee.

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

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

(4) Report(s) shall be made to the U.S. NRC in accordance with Section 6.10 of these technical specifications.

6.8 Operating Procedures

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 by the situation. Operating procedures shall be in effect for the following:

(1) performing irradiations and experiments

(2) startup, operation, and shutdown of the reactor

(3) emergency situations including provisions for building evacuation, earthquake, radiation emergencies, fire or explosion, personal injury, civil disorder, and bomb threat

(4) core changes and fuel movement

(5) control rod removal and replacement

(6) performing preventive maintenance and calibration tests on the reactor and associated equipment

(7) power calibration

(8) radiation protection

(9) implementing procedures for the Emergency Plan

(10) shipping radioactive materials

Substantive changes to the above procedures shall be made only with the approval of the Director of the Nuclear Radiation Center and after review and approval by the Reactor Safeguards Committee. Temporary minor changes to the procedures (e.g. wording changes to improve clarity of the procedure) that do not change the original intent may be made by a licensed senior reactor operator. All such temporary changes shall be documented and subsequently reviewed by the Director of the Nuclear Radiation Center and the Reactor Safeguards Committee.

6.9 Facility Operating Records

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, 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 At least 5 years for items (1) through (7)

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

(2) principal maintenance operations

(3) reportable occurrences

(4) surveillance activities required by the technical specifications

(5) experiments performed with the reactor

(6) reactor facility radiation and contamination surveys where required by applicable regulations

(7) approved changes in operating procedures

6.9.2 For the life of the facility for items (8) through (14):

(8) gaseous and liquid radioactive effluents released to the environs

(9) off-site inventories and transfers

(10) fuel inventories, receipts, and shipments

(11) facility radiation and contamination surveys

(12) radiation exposures for all personnel monitored

(13) updated, corrected, and as-built drawings of the facility

(14) off-site environmental monitoring surveys required by the technical specifications

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 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting for those requirements, reports shall be made to the Nuclear Regulatory Commission as follows:

(1) A report within 24 hours by telephone to the NRC Operations Center, of

(a) An accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;

(b) A violation of the safety limit

(c) A reportable occurrence as defined in Section 1.0,"Reportable Occurrence;"of these specifications.

(2) A report within 10 days in writing to USNRC Document Control Desk, Washington, D.C. 20555, of

(a) An accidental release or 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 or telegraph report) shall describe, analyze, and evaluate safety implications, and outline the corrective measures taken or planned to prevent recurrence of the event;

(b) A violation of a safety limit;

(c) A reportable occurrence as defined in Section 1.0,"Reportable Occurrence;"of these specifications.

(3) A report within 30 days in writing to the USNRC Document Control Desk, Washington, D.C. 20555, of

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

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

(c) A significant changes in facility organization;

(d) Observed inadequacies in the implementation of administrative or procedural controls.

(4) A report within 60 days after completion of startup testing of the reactor (in writing to the USNRC Document Control Desk, Washington, D.C. 20555) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level describing the measured values of the operating conditions including:

(a) An evaluation of facility performance to date in comparison with design predictions and specifications;

(b) A reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

(5) An annual report within 60 days following the 30th of June of each year in writing to the USNRC Document Control Desk, Washington, D.C. 20555, providing the following information:

(a) A brief narrative summary of (i) operating experience (including experiments performed), (ii) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (iii) results of surveillance tests and inspections;

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

(c) The number of emergency shutdowns and inadvertent scrams, including reasons for them;

(d) 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;

(e) 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;

(f) 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:

Liquid Waste (summarized on a monthly basis)

(i) monthly radioactivity discharged

- total estimated quantity of radioactivity released (in curies),
- an estimation of the specific quantity for each detectable radionuclide in the monthly release,
- 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,
- sum of the fractions for each radionuclide reported above.

(ii) total quantity of radioactive material released by the facility into the sewage system during the year period of the report

Gaseous Waste (summarized on a monthly basis)

(i) radioactivity discharged during the reporting period (in curies)

- total estimated quantity of radioactivity released (in curies) determined by an appropriate sampling and counting method,
- total estimated quantity of ^{41}Ar released (in curies) during the reporting period based on data from an appropriate monitoring system,
- 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,
- 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,
- average concentration of radioactive particulates with half-lives greater than 8 days released in μCi/mL during the reporting period, and
- 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.

Solid Waste (summarized on an annual basis)

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

(ii) total radioactivity in solid waste (in curies),

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

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

(h) An annual summary of the radiation levels of contamination observed during routine surveys performed at the facility in terms of the average and highest levels;

(i) An annual summary of any environmental surveys performed outside the facility.

6.11 Written Communications

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

References

6.5-1 MITR Staff,"Safety Analysis Report for the MIT Research Reactor (MITR-II)," Report No. MITNE-115, 22 Oct. 1970, Section 10.1.3.

6.5-2 Choi, R.J.,"Development and Characterization of an Epithermal Beam for Boron Neutron Capture Therapy at the MITR-II Research Reactor," Ph.D. Thesis, Nuclear Engineering Department, Massachusetts Institute of Technology, April 1991.