OREGON STATE UNIVERSITY

DOCKET NO. 50-243

RENEWED FACILITY OPERATING LICENSE

License No. R-106

1. The U.S. Nuclear Regulatory Commission (NRC or the Commission) has found the following:

   A. The application for renewal of Facility Operating License No. R-106 filed by the Oregon State University (the licensee) dated October 5, 2004, as supplemented on August 8, 2005; May 24, 2006; November 10, 2006; November 21, 2006; July 10, 2007; July 27, 2007; July 31, 2007; August 6, 2007; April 14, 2008; August 6, 2008; and August 11, 2008 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Chapter I of the *Code of Federal Regulations* (CFR);

   B. Construction of the Oregon State University TRIGA reactor (the facility) was completed in substantial conformity with Construction Permit No. CPRR-93 and the application, as amended; the provisions of the Act; and the rules and regulations of the Commission;

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

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

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

   F. The licensee is a university which will use the facility for the conduct of educational training and academic research purposes, and has submitted documentation that is required by Subsection 170a of the Act and has been approved by the NRC;

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

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

I. The receipt, possession and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70.

2. Renewed Facility License No. R-106 is hereby amended in its entirety to read as follows:

A. The license applies to the TRIGA nuclear research reactor (hereafter, the facility) owned by the Oregon State University. The facility is located in Corvallis, Oregon, and is described in the licensee's application for renewal of the license dated October 5, 2004, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Oregon State University:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location in Corvallis, Benton County, Oregon in accordance with the procedures and limitations described in the application and set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material,"

a. to receive, possess and use, in connection with operation of the facility, up to 16.30 kilograms of contained uranium-235 enriched to less than 20 percent in the form of TRIGA reactor fuel;

b. to receive, possess and use, in connection with operation of the facility, up to 100 grams of contained uranium-235 of any enrichment in the form of fission chambers and flux foils;

c. to receive, possess, but not use, up to 656 grams of uranium-235 enriched to less than 20 percent in the form of the core from the AGN-201 reactor;

d. to receive, possess, use, but not separate, in connection with operation of the facility, such special nuclear material as may be produced by operation of the facility;

e. to possess, but not use, up to 12.83 kilograms of contained uranium-235 at equal to or greater than 20 percent enrichment in the form of TRIGA fuel until the existing inventory of this fuel is removed from the facility; and

f. to receive, possess, and use, but not separate, in connection with operation of the facility, up to 0.5 kilograms of contained uranium-235 enriched to less than 20 percent in the form of molybdenum-99 production targets.

(3) Pursuant to the Act and 10 CFR Part 30 to receive, possess and use in connection with operation of the facility the following:

a. up to a 7-curie sealed polonium-210 beryllium source which may be used for reactor startup;

b. up to a 3-curie sealed americium-241 beryllium neutron source which may be used for reactor startup;

c. such byproduct material as may be produced by the operation of the facility. Byproduct material cannot be separated except for byproduct material produced in experiments.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state power levels not in excess of 1.1 megawatts (thermal), and in the pulse mode, with reactivity insertions not to exceed $2.30.

(2) Technical Specifications

The technical specifications contained in Appendix A, as revised through Amendment No. 26, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the technical specifications.

(3) Physical Security Plan

The licensee shall maintain and fully implement all provisions of the NRC-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 73.21, entitled “Oregon State University TRIGA Reactor Physical Security Plan,” as revised.

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

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Enclosure:
Appendix A Technical Specifications

Date of Issuance: September 10, 2008

APPENDIX A

TO

FACILITY LICENSE NO. R-106

TECHNICAL SPECIFICATIONS

AND BASIS

FOR THE

OREGON STATE UNIVERSITY

TRIGA® REACTOR

DOCKET NO. 50-243

Current through Amendment #_____

Date of Issuance:__________

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

1 DEFINITIONS

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

1.2 Channel Calibration: A channel calibration is an adjustment of the channel such that its output corresponds 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, and shall be deemed to include a channel test.

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

1.4 Channel Test: A channel test is the introduction of a signal into the channel for verification that it is operable.

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

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

- **a. Regulating Rod:** The regulating rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled-follower section. Its position may be varied manually or by the servo-controller.
- **b. Shim/Safety Rod:** A shim/safety rod is a control rod having an electric motor drive and scram capabilities. It may have a fueled-follower section.
- **c. Transient Rod:** The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. It may have a voided follower.

1.7 Core Lattice Position: The core lattice position is defined by a particular hole in the top grid plate of the core. It is specified by a letter indicating the specific ring in the grid plate and a number indicating a particular position within that ring.

1.8 Excess Reactivity: Excess reactivity is that amount of reactivity that would exist if all control rods were moved to the maximum reactive condition from the point where the reactor is exactly critical ($k_{eff} = 1$) at reference core conditions.

1.9 Experiment: An experiment is any operation, hardware, or target (excluding devices such as detectors or foils) that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within an irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of their design to carry out experiments is not normally considered an experiment. Specific experiments shall include:

a. **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. The restraining forces shall 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.

b. **Unsecured Experiment:** An unsecured experiment is any experiment or component of an experiment that does not meet the definition of a secured experiment.

c. **Movable Experiment:** A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the core or into and out of the core while the reactor is operating.

1.10 Fuel Element: A fuel element is a single TRIGA® fuel rod.

1.11 Irradiation Facilities: Irradiation facilities shall mean beam ports, including extension tubes with shields; thermal columns with shields; vertical tubes; rotating specimen rack; pneumatic transfer system; and any other in-tank irradiation facilities.

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

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

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

1.15 Operational Core: An operational core shall be a fuel element core that operates within the licensed power level and satisfies all the requirements of the Technical Specifications.

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

1.17 Radiation Center Complex: The Radiation Center Complex is the physical area defined by the Radiation Center Building and the fence surrounding the north, west, and east sides of the Reactor Building.

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

1.19 Reactor Safety Systems: Reactor safety systems are those systems, including their associated input channels, that are designed to initiate, automatically or manually, a reactor scram for the primary purpose of protecting the reactor.

1.20 Reactor Secured: The reactor is secured when:

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

b. All of the following exist:

    1. The four (4) neutron absorbing control rods are fully inserted;
    2. The console key switch is in the "off" position and the key is removed from the console;
    3. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods; and
    4. No experiments or irradiation facilities in the core are being moved or serviced that have, on movement or servicing, a reactivity worth exceeding the maximum value of $0.50.

1.21 Reactor Shut Down: The reactor is shut down if it is subcritical by at least one dollar in the reference core condition with the reactivity worth of all installed experiments and irradiation facilities included.

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

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

1.24 Scram Time: Scram time is the elapsed time between reaching a limiting safety system setting and the instant that the slowest scrammable control rod reaches its fully-inserted position.

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

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

1.27 Square-Wave Mode (S.-W. Mode): Square-wave mode shall mean any operation of the reactor with the mode selector switch in the square-wave position.

1.28 Steady-State Mode (S.-S. Mode): Steady-state mode shall mean operation of the reactor with the mode selector switch in the steady-state or automatic position.

1.29 Substantive Changes: Substantive changes are changes in the original intent or safety significance of an action or event.

1.30 Surveillance Intervals: Allowable surveillance intervals shall not exceed the following:

a. Biennial – interval not to exceed 30 months

b. Annual – interval not to exceed 15 months

c. Semi-annual – interval not to exceed 7.5 months

d. Quarterly – interval not to exceed 4 months

e. Monthly – interval not to exceed 6 weeks

f. Weekly – interval not to exceed 10 days

2 SAFETY LIMIT AND LIMITING SAFETY SYSTEM SETTING

2.1 Safety Limit - Fuel Element Temperature

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

Objective. The objective is to define the maximum fuel element temperature that can be permitted with confidence that no damage to the fuel element cladding shall result.

Specifications. The temperature in a TRIGA® fuel element shall not exceed 1150°C (2100°F) under any mode of operation.

Basis. Fuel element cladding is the barrier that prevents fission products from contaminating the coolant. A loss of the integrity of the fuel element cladding could arise from a build-up of excessive pressure between the fuel-moderator and the cladding if the 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 TRIGA® fuel element is based on data that indicate that the stress in the cladding due to the hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress, provided the temperature of the fuel does not exceed 1150°C (2100°F) and the fuel cladding is water cooled.

2.2 Limiting Safety System Setting

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

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

Specifications The limiting safety system setting (LSSS) shall not exceed 1.1 MW as measured by the safety and percent power level channels.

Basis. During steady state operation, maximum temperatures are predicted to occur in the LEU MOL ICIT core. Chapter 4 shows that, during steady state 1.1 MW operation, maximum temperatures are predicted to occur in the LEU MOL ICIT core, with a maximum hot rod thermal power of 18.52 kW. Analysis shows that the MDNBR in the hot channel (using the Bernath correlation) will reach a value of 2.00 at approximately 19.85 kW hot channel steady state 1.1 MW power. This is 107.2% of the 18.52 kW produced in the hot channel of the LEU MOL ICIT core configuration. The power level channels scram the reactor at 106% of 1 MW power which is far below the power level required to reach a MDNBR of 2.00. The two power level channels provide redundant and independent protection from exceeding the safety limit of 1150°C. Therefore, measurement of the reactor power by two redundant and independent power level channels will ensure that the safety limit is not reached during steady state operations.

Analysis indicates that at 18.52 kW, the maximum temperature anywhere in the hot channel fuel element will be less than 500°C, which is far below the safety limit of 1150°C.

Additional analysis was performed on the very unlikely accident involving the simultaneous withdrawal of all four control rods, with an initial power of 100 W, a scram setpoint of 1.06 MW, and a 0.5 second delay time to scram the reactor. The analysis showed that the maximum hypothetical reactivity insertion would be $1.13, which is far below the maximum allowable pulse of $2.30. Pulsing thermal hydraulic analysis showed that a pulse of $2.30 would not exceed 830°C, which is below the safety limit and prevents potential damage due to exceeding eutectic limits.

3 LIMITING CONDITIONS FOR OPERATIONS

3.1 Reactor Core Parameters

3.1.1 Steady-State Operation

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

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

Specifications. The reactor power level shall not exceed 1.1 MW except for pulsing operations.

Basis. Chapter 4 indicates that, in the most conservative core configuration (LEU MOL ICIT core), at 18.52 kW, the maximum temperature anywhere in the hot channel fuel element will be less than 500°C, which is far below the safety limit of 1150°C.

3.1.2 Shutdown Margin

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

Objective. The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit shall not be exceeded.

Specifications. The reactor shall not be operated unless the following conditions exist:

The shutdown margin provided by control rods shall be at least $0.55 with:

a. Irradiation facilities and experiments in place and the total worth of all unsecured experiments in their most reactive state;

b. The most reactive control rod fully withdrawn; and

c. The reactor in the reference core condition.

Basis. The value of the shutdown margin ensures that the reactor can be shut down from any operating condition even if the most reactive control rod should remain in the fully withdrawn position.

3.1.3 Core Excess Reactivity

Applicability. This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments. It applies for all modes of operation.

Objective. The objective is to ensure that the reactor can be shut down at all times and to ensure that the fuel temperature safety limit shall not be exceeded.

Specifications. The maximum available excess reactivity based on the reference core condition shall not exceed $7.55.

Basis. An excess reactivity limit of $7.55 allows flexibility to operate the reactor in various core configurations without the need to add or remove fuel elements when changing between operating modes. Operating with a fuel element in the B1 position is the most reactive core configuration. If operating in this core configuration with the minimum shutdown margin of $0.55 and typical control rod worths of $2.50 (Safety), $2.60 (Shim), $3.70 (Regulating) and $3.00 (Transient), the calculated NRC core excess would be -$0.55+$2.50+$2.60+$3.00 = $7.55. The shutdown margin calculation assumes a) irradiation facilities and experiments in place and the total worth of all unsecured experiments in their most reactive state, b) the most reactive control rod fully withdrawn and c) the reactor in the reference core condition. Activities such as moving away from the reference state or adding negative worth experiments will decrease core excess and increase shutdown margin. The OSTR normally operates with a cadmium-lined irradiation facility (CLICIT) in B1, which is significantly less reactive than the limiting core configuration. The only activity that could result in requiring fuel movement to meet shutdown margin and core excess limits would be the unusual activity of adding an experiment with large positive reactivity worth.

3.1.4 Pulse and Square-Wave Mode Operation

Applicability. This specification applies to the energy generated in the reactor as a result of a pulse or square-wave insertion of reactivity.

Objective. The objective is to ensure that the fuel temperature safety limit shall not be exceeded.

Specifications. The reactivity to be inserted for pulse mode or square-wave mode operation shall be determined and limited by a mechanical block and electrical interlock on the transient rod, such that the maximum reactivity insertion shall not exceed $2.30.

Basis. GA-C26017, Pulsing Temperature Limit for TRIGA® LEU Fuel, recommends that TRIGA® fuel should not exceed 830°C. The fuel temperature rise during a pulse transient has been estimated by RELAP-5-3D using non-adiabatic models. The core at the middle of life was found to have the highest accumulative peaking factor for any core configuration during the fuel lifetime. These models predict pulse characteristics for operation of operational cores and should be accepted with confidence, relying also on information concerning prompt neutron lifetime and prompt temperature coefficient of reactivity. The reactivity value calculated to produce a temperature of 830°C is $2.33. Therefore, limiting reactivity insertions to a maximum of $2.30 will ensure that fuel temperature will not exceed 830°C.

3.1.5 This section intentionally left blank.

3.1.6 Fuel Parameters

Applicability. This specification applies to all fuel elements.

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

Specifications. The reactor shall not operate with damaged fuel elements, except for the purpose of locating damaged fuel elements. A fuel element shall be considered damaged and shall be removed from the core if:

a. The transverse bend exceeds 0.0625 inches over the length of the cladding;

b. Its length exceeds its original length by 0.125 inches;

c. A cladding defect exists as indicated by release of fission products; or

d. Visual inspection identifies bulges, gross pitting, or corrosion.

Basis. Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation and bend limits are the values found acceptable to the USNRC (NUREG-1537).

3.2 Reactor Control and Safety System

3.2.1 Control Rods

Applicability. This specification applies to the function of the control rods.

Objective. The objective is to determine that the control rods are operable.

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

a. Damage is apparent to the rod or rod drive assemblies; or

b. The scram time exceeds 2 seconds.

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

3.2.2 Reactor Measuring Channels

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

Objective. The objective is to specify the minimum number of measuring channels that shall be available to the operator to ensure safe operation of the reactor.

Specifications. The reactor shall not be operated in the specified mode unless the minimum number of measuring channels listed in Table 1 are operating.

Table 1 - Minimum Measuring Channels

Measuring Channel	Effective Mode		
	S.-S.	Pulse	S.-W.
Linear Power Level	1	-	1
Log Power Level	1	-	1
Safety Power Level	1	-	1
Percent Power Level	1	-	1
Nvt-Circuit	-	1	-

(1) Any single measuring channel (except the Nvt-Circuit) may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration. Note that the Linear Power Level and Log Power Level are connected to the same detector and both become inoperable when performing a channel check, test, or calibration.

(2) If any required measuring channels becomes inoperable while the reactor is operating for reasons other than that identified in footnote (1), the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shut down.

Basis. The power level monitors ensure that the reactor power level is adequately monitored for both steady-state and square wave modes of operation. The Nvt-Circuit provides indication of peak power and total energy generated during a pulse. The specifications on reactor power level indication are included in this section, since the power level is related to the fuel temperature. For footnote (1), taking these measuring channels off-line for short durations for the purpose of a check, test, or calibration is considered acceptable because in some cases, the reactor needs to be operating in order to perform the check, test, or calibration. Additionally there exist two redundant and independent power level indications operating at any given time while the third single channel is off-line. For footnote (2), events that lead to these circumstances are self-revealing to the operator. Furthermore, recognition of appropriate action on the part of the operator as a result of an instrument failure would make this consistent with TS 6.7.2.

3.2.3 Reactor Safety Systems

Applicability. This specification applies to the reactor safety systems.

Objective. The objective is to specify the minimum number of reactor safety systems that shall be available to the operator to ensure safe operation of the reactor.

Specifications. The reactor shall not be operated in the specified mode unless the minimum number of safety systems described in Table 2 are operable.

Table 2 - Minimum Reactor Safety Systems

Safety System	Function	Effective Mode		
		S.-S.	Pulse	S.-W.
Safety Power Level	SCRAM @ 1.1 MW or less	1	-	1
Percent Power Level	SCRAM @ 1.1 MW or less	1	-	1
Manual Scram	SCRAM (button on console)	1	1	1
Preset Timer	Transient rod SCRAM @ ≤ 15 sec after pulse	-	1	-
High Voltage	SCRAM @ ≥ 25% of nominal operating voltage	1	1	1

(1) The Safety Power channel or the Percent Power channel may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.
(2) If any required safety channel becomes inoperable while the reactor is operating for reasons other than that identified in footnote (1), the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shut down.

Basis.

Safety and Percent Power Level Scrams: The set point for both the safety and percent power channels are normally set to 106% of 1 MW, which is below the licensed power of 1.1 MW. The 6% difference allows for expected and observed instrument fluctuations at the normal full operating power of 1 MW to occur without scramming the reactor unnecessarily. Conversely, SAR Chapter 13 shows that this set point is more than sufficient to prevent the operator from inadvertently exceeding the licensed power.

Manual Scram: The manual scram shall be functional at all times the reactor is in operation. It has no specified value for a scram set point. It is initiated by the reactor operator manually.

Preset Timer Scram: The preset timer ensures that the reactor power level will reduce to a low level after pulsing and preclude an unintentional restart or ramped increase to some equilibrium power.

High Voltage Scram: The high voltage scram shall be set to initiate a scram before the high voltage for any of the three detectors drops below 25% of the nominal operating voltage. The loss of operating voltage down to this level is an indication of detector failure. Many measuring channels and safety systems are fundamentally based upon accurate response of the detectors.

For footnote (1), taking these safety channels off-line for short durations for the purpose of a check, test, or calibration is considered acceptable because in some cases, the reactor needs to be operating in order to perform the check, test, or calibration. Additionally there exist two redundant and independent power level indications operating at any given time while the third single channel is off-line. For footnote (2), events that lead to these circumstances are self-revealing to the operator. Furthermore, recognition of appropriate action on the part of the operator as a result of an instrument failure would make this consistent with TS 6.7.2.

3.2.4 Reactor Interlocks

Applicability. This specification applies to the reactor safety system interlocks.

Objective. The objective is to specify the minimum number of reactor safety interlocks that shall be available to the operator to ensure safe operation of the reactor.

Specifications. The reactor shall not be operated in the specified mode unless the minimum number of interlocks described in Table 3 are operable.

Table 3 – Minimum Reactor Interlocks

Interlock	Function	Effective Mode		
		S.-S.	Pulse	S.-W.
Wide-Range Log Power Level Channel	Prevents control rod withdrawal @ less than 2 cps	1	-	-
Transient Rod Cylinder	Prevents application of air unless fully inserted	1	-	-
1 kW Pulse Interlock	Prevents pulsing above 1 kW	-	1	-
Shim, Safety, and Regulating Rod Drive Circuit	Prevents simultaneous manual withdrawal of two rods	1	-	1
Shim, Safety, and Regulating Rod Drive Circuit	Prevents movement of any rod except transient rod	-	1	-
Transient Rod Cylinder Position	Prevent pulse insertion of reactivity greater than $2.30	-	1	1

(1) Any single interlock may be inoperable while the reactor is operating for the purpose of performing a channel check, test, or calibration.
(2) If any required interlock becomes inoperable while the reactor is operating for reasons other than that identified in footnote (1) above, the interlock shall be restored to operation within 5 minutes or the reactor shall be immediately shut down.

Basis.

Wide-Range Log Power Level Channel Interlock: The rod withdrawal prohibit interlock prevents the operator from adding reactivity when the count rate on the wide-range log power channel falls below 2 cps. When this happens, the count rate is insufficient to produce meaningful instrumentation response. If the operator were to insert reactivity under this condition, the period could quickly become very short and result in an inadvertent power excursion. A neutron source is added to the core to create sufficient instrument response that the operator can recognize and respond to changing conditions.

Transient Rod Cylinder Interlock: This interlock prevents the application of air to the transient rod unless the cylinder is fully inserted. This will prevent the operator from pulsing the reactor in steady-state mode.

1 kW Pulse Interlock: The 1 kW permissive interlock is designed to prevent pulsing when wide range log power is above 1 kW. Analysis of pulsing at full power shows that if the initial temperature was higher, the resulting peak temperature will be lower. However, there has not been an experiment to look at the relationship between heat generated within the fuel at power (i.e., > 1 kW) and heat generated on the surface of the fuel during a pulse. Therefore, this interlock prevents the reactor from pulsing at power levels that produce measurably significant increases in fuel temperature.

Shim, Safety, and Regulating Rod Drive Circuit: The single rod withdrawal interlock prevents the operator from removing multiple control rods simultaneously such that reactivity insertions from control rod manipulation are done in a controlled manner. The analysis in SAR Chapter 13 shows that the reactivity insertion due to the removal rate of the most reactive rod or all the control rods simultaneously is still well below $2.30 of reactivity.

Shim, Safety, and Regulating Rod Drive Circuit: In pulse mode, it is necessary to limit the reactivity insertion to $2.30. This interlock ensures that all pulse reactivity is due to only the transient rod while in pulse mode. Otherwise, any control rod removal in pulse mode would add to the inserted reactivity of the transient rod and create an opportunity for exceeding the reactivity insertion limit.

Transient Rod Cylinder Position Interlock: The transient rod cylinder interlock shall limit reactivity insertions below $2.30. Furthermore, this interlock is designed such that if the electrical (i.e., limit switch) portion fails, a mechanical (i.e., metal bracket) portion will still keep the reactivity insertion below the criterion.

For footnote (1), taking these interlocks off-line for short durations for the purpose of a check, test, or calibration is considered acceptable because in some cases, the reactor needs to be operating in order to perform the check, test, or calibration. For footnote (2), events that lead to these circumstances are self-revealing to the operator. Furthermore, recognition of appropriate action on the part of the operator as a result of an interlock failure would make this consistent with TS 6.7.2.

3.3 Reactor Primary Tank Water

Applicability. This specification applies to the primary water of the reactor tank.

Objective. The objective is to ensure that there is an adequate amount of water in the reactor tank for fuel cooling and shielding purposes, and that the temperature of the reactor tank water remains sufficiently low to guarantee reactor tank integrity.

Specifications. The reactor primary water shall exhibit the following parameters:

a. The tank water level shall be at least 14 feet above the top of the core;

b. The tank water temperature shall not exceed 49°C (120°F); and

c. The conductivity of the tank water shall not exceed 5 μmhos/cm.

Basis. The minimum height of 14 feet of water above the top of the core guarantees that there is sufficient water for effective cooling of the fuel and that the radiation levels at the top of the reactor are within acceptable levels. The water temperature limit is necessary, according to the reactor manufacturer, to ensure that the aluminum reactor tank maintains its integrity and is not degraded. Experience at many research reactor facilities has shown that maintaining the conductivity within the specified limit provides acceptable control of corrosion (NUREG-1537).

3.4 This section intentionally left blank.

3.5 Ventilation System

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

Objective. The objective is to ensure that the ventilation system shall be in operation to mitigate the consequences of possible releases of radioactive materials resulting from reactor operation.

Specifications.

a. The reactor shall not be operated unless the facility ventilation system is operating and the reactor bay pressure is maintained negative with respect to surrounding areas, except for periods of time not to exceed two (2) hours to permit repair, maintenance, or testing of the ventilation system.

b. The ventilation system shall be shut down upon a high activity alarm from the Exhaust Particulate Radiation Monitor.

c. Experiments with significant potential for airborne radioactivity release shall not be transferred from an irradiation facility unless the facility ventilation system is operating and the reactor bay pressure is maintained negative with respect to surrounding areas.

Basis. During normal operation of the ventilation system, the annual average ground concentration of ^{41}Ar in unrestricted areas is well below the applicable effluent concentration limit in 10 CFR 20. In addition, the worst-case maximum total effective dose equivalent is well below the applicable annual limit for individual members of the public. This has been shown to be true for scenarios where the ventilation system continues to operate during the maximum hypothetical accident (MHA), where the ventilation system is secured during the MHA, and where the ventilation system and the confinement building are not present during the MHA (SAR Chapter 13). Therefore, operation of the reactor for short periods while the ventilation system is shut down for repair or testing does not compromise the control over the release of radioactive material to the unrestricted area, nor should it cause occupational doses that exceed those limits given in 10 CFR 20 (SAR Chapter 11). The two hour exception to permit repair, maintenance, or testing should not diminish the effectiveness of the reactor top area radiation monitor or the continuous air particulate radiation monitor. The sampling locations for both of these monitors are located directly above the core. Any fission product release should be detected in the same manner regardless of the status of the ventilation system because of the close proximity of the sampling point to the source term. Moreover, radiation monitors in the building, independent of the ventilation system, will give warning of high levels of radiation that might occur during operation of the reactor while the ventilation system is secured (SAR Chapter 11). The exhaust gas and particulate radiation monitors will be affected by the status of the ventilation system as they are designed to monitor the ventilation exhaust directly and are not in close proximity to the source term (i.e., reactor core). However, control of the release into the unrestricted area will be minimally compromised because the ventilation will be by definition off and the leak rate is negligible compared to the ventilation rate.

Furthermore, this situation is bounded by the MHA scenario A (i.e., without the reactor building) and C (i.e., ventilation off) in the SAR (SAR Chapter 13). Significant potential for airborne radioactivity is the radioactivity that when released into the reactor bay results in the calculated occupational or general public dose equivalent or committed dose equivalent greater than half that of the corresponding calculated value for the maximum hypothetical accident as described in the Safety Analysis Report (as amended).

3.6 This section intentionally left blank.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Radiation Monitoring Systems

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

Objective. The objective is to specify the minimum radiation monitoring channels that shall be available to the operator to ensure safe operation of the reactor.

Specifications. The reactor shall not be operated nor shall experiments with significant potential for airborne radioactivity be transferred from an irradiation facility unless the minimum number of radiation monitoring channels listed in Table 4 are operating.

Table 4 - Minimum Radiation Monitoring Channels

Radiation Monitoring Channels	Number
Reactor Top Area Radiation Monitor	1
Continuous Air Particulate Radiation Monitor	1
Exhaust Gas Radiation Monitor	1
Exhaust Particulate Radiation Monitor	1

Exception: When a single required radiation monitoring channel becomes inoperable, operations may continue only if portable instruments, surveys, or analyses may be substituted for the normally installed monitor within one (1) hour of discovery for periods not to exceed one (1) month.

Basis. The radiation monitors provide information to operating personnel regarding routine releases of radioactivity and any impending or existing danger from radiation. Their operation will provide sufficient time to evacuate the facility or take the necessary steps to prevent the spread of radioactivity to the surroundings. Furthermore, calculations show that for both routine operations and under the three accident scenarios identified in SAR Chapter 13, predicted occupational and general public doses are below the applicable annual limits specified in 10 CFR 20 (SAR Chapters 11 and 13). That being the case, we have reasonable assurance that the applicable regulatory limits are being satisfied for the one hour period. Significant potential for airborne radioactivity is the radioactivity that when released into the reactor bay results in the calculated occupational or general public dose equivalent or committed dose equivalent greater than half that of the corresponding calculated value for the maximum hypothetical accident as described in the Safety Analysis Report (as amended).

3.7.2 Effluents

Applicability. This specification applies to the release rate of ^{41}Ar.

Objective. The objective is to ensure that the concentration of the ^{41}Ar in the unrestricted areas shall be below the applicable effluent concentration value in 10 CFR 20.

Specifications. The annual average concentration of ^{41}Ar discharged into the unrestricted area shall not exceed 4 x 10^{-6} µCi/ml at the point of discharge.

Basis. If ^{41}Ar is continuously discharged at 2.5 x 10^{-6} µCi/ml (i.e., the concentration produced when the nitrogen purge of the rotating rack is off, all valves on the argon manifold are open, and all beam port valves are open), measurements and calculations show that ^{41}Ar released to the unrestricted areas under the worst-case weather conditions would result in an annual TEDE of 5 mrem (SAR Chapter 11). This is only 50% of the applicable limit of 10 mrem (Regulatory Guide 4.20). Therefore, an emission of 4 x 10^{-6} µCi/ml would correspond to an annual TEDE of 8 mrem, which is still 20% below the applicable limit.

3.8 Experiments

3.8.1 Reactivity Limits

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

Objective. The objective is to prevent damage to the reactor due to excessive reactivity insertion in the event of an experiment failure.

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

a. The absolute value of the reactivity worth of any single unsecured experiment shall not exceed $0.50; and

b. The sum of the absolute values of the reactivity worths of all experiments shall not exceed $2.30.

Basis. The reactivity limit of $0.50 for unsecured experiments is designed to prevent an inadvertent pulse from occurring and maintain a value below the shutdown margin. Unsecured experiments are located in such a way that it is possible for them to be removed from or inserted into an irradiation facility, either accidentally or deliberately, while the reactor is critical. That being said, the value is clearly less than the limit on pulsing.

The reactivity worth limit for all experiments is designed to prevent an inadvertent pulse from exceeding the recommended limit on pulsing. This limit applies to movable, unsecured, and secured experiments. A maximum reactivity insertion of $2.30 will ensure that fuel temperature will not exceed 830°C.

3.8.2 Materials

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

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:

a. Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or irradiation facilities. Explosive materials in quantities up to 25 milligrams may be irradiated 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; and

b. Experiments containing corrosive materials shall be doubly encapsulated. The failure of an encapsulation of material that could damage the reactor shall result in removal of the sample and physical inspection of potentially damaged components.

Basis. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive or corrosive materials. Operation of the reactor with the reactor fuel or structure potentially damaged is prohibited to avoid potential release of fission products.

3.8.3 Failures and Malfunctions

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

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

Specifications. Where the possibility exists that the failure of an experiment under normal operating conditions of the experiment or reactor, credible accident conditions in the reactor, or possible accident conditions in the experiment could release radioactive gases or aerosols to the reactor bay or the unrestricted area, the quantity and type of material in the experiment shall be limited such that the airborne radioactivity in the reactor bay or the unrestricted area will not result in exceeding the applicable dose limits in 10 CFR 20, assuming that:

a. 100% of the gases or aerosols escape from the experiment;

b. Each experiment containing fissile material shall be controlled such that the total inventory of ^{131}I escaping from the experiment shall not exceed 0.0141 curies;

c. If the effluent from an irradiation facility exhausts through a holdup tank that closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape;

d. If the effluent from an irradiation facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these aerosols can escape; and

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

Basis. This specification is intended to meet the purpose of 10 CFR 20 by reducing the likelihood that released airborne radioactivity to the reactor bay or unrestricted area surrounding the OSTR will result in exceeding the total dose limits to an individual as specified in 10 CFR 20. The limit on fissile material is based on the amount of ^{131}I estimated to be made available to the reactor bay air during the Maximum Hypothetical Accident found in the Safety Analysis Report.

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3.10 Targets

3.10.1 Permissible In-Core Target Lattice Positions

Applicability. This specification applies to the core at any time when targets are located in any core lattice position.

Objective. The objective is to ensure assumptions made for the neutronic and thermal hydraulic analyses are not compromised.

Specifications. Permissible target locations are core positions G32, G33, and G34. Targets shall not be placed in any other core lattice positions.

Basis. Analyzed target locations were G32, G33, and G34. Location G34 was found to produce the highest integrated power in a target. Thermal hydraulic analysis was based on power distribution in this hot target.

3.10.2 Pulse or Square Wave Mode Operation with Targets Located in Any Core Lattice Position

Applicability. This specification applies to the reactor at any time when targets are located in any core lattice position.

Objective. The objective is to prevent all pulse and square wave activity while targets are present in any core lattice position.

Specifications. The reactor shall not be operated in pulse mode or square wave mode while targets are present in any core lattice position.

Basis. Target performance has not been analyzed under rapid transient pulse conditions; therefore, pulsing shall not be allowed when targets are present in the core. Pulse mode operation is prohibited. Square wave mode operation is also prohibited because it is possible to add more than $1.00 of reactivity in square wave mode. A rod withdrawal accident will not introduce sufficient reactivity to pulse the reactor.

3.10.3 Allowed Target Storage Locations

Applicability. This specification applies to targets at any time targets are located in the reactor tank and not in transit or in an in-core lattice position.

Objective. The objective is to maintain k-effective of stored targets less than 0.9 under all conditions of moderation.

Specifications. The targets shall be stored in the standard in-tank TRIGA® storage racks. No other items shall be present in any storage rack containing targets.

Basis. Storage racks are sufficiently far from the core such that the presence of targets in the core will not affect the criticality condition of targets in the storage racks. Criticality analysis assumes no other objects are present in the vicinity of the stored targets. The criticality analysis for the storage of the fuel assumed no other objects (i.e., fuel elements) were stored with the targets. The k-effective was calculated to be less than 0.9 when stored in the storage racks.

3.10.4 Target Fabrication Requirements

Applicability. This specification applies to any target that will be placed in the reactor tank.

Objective. The objective is to ensure that targets placed in the core may be used with a high degree of reliability with respect to their physical and nuclear properties.

Specifications.

a. The maximum enrichment of uranium in each target shall not exceed 19.75%.

b. The maximum mass of uranium in a target shall not result in a dose to a member of the general public in excess of 100 mrem from an accident involving a single target.

c. Cladding: aluminum, nominal thickness 0.32 cm.

Basis.

a. Targets shall be fabricated with LEU (i.e., less than or equal to 19.75% enriched in ^{235}U). An enrichment of 20% was assumed for the neutronic and the thermal hydraulic analysis for the purpose of bounding the calculations.

b. The dose to the general public from the target cladding failure accident is a function of many variables. Provided all other variables remain constant, the predicted dose should be directly proportional to the mass of uranium in the target. Analysis has shown that the maximum dose to a member of the general public will not exceed 100 mrem given the assumptions made in the calculation of the target cladding failure accident. Therefore, the mass of uranium in each target is limited by the parameters of the analysis and the dose performance criteria.

c. Cladding of this type provides adequate structural integrity while minimizing parasitic neutron absorption.

3.10.5 ^{99}Mo Target Irradiation

Applicability. This specification applies to the irradiation of ^{99}Mo demonstration targets.

Objective. The objective is to ensure that the time that the ^{99}Mo demonstration targets are irradiated is limited by reactor power history.

Specifications. The ^{99}Mo demonstration targets shall be irradiated in a core lattice position for no more than 7.15-MW days (MWD).

Basis. The predicted radionuclide inventory was based upon a 6.5-day irradiation while the reactor is at full power (i.e., 1.1 MW). The multiple of these two numbers represents the effective full power days for the core while the targets are in the core lattice positions analyzed. This power history creates the source term inventory that was predicted for the accident analysis that could potentially be released from the uranium bearing material within the targets. Limiting the irradiation time to 7.15 MWD will ensure that the potential accident consequences are less than the dose limit for individual members of the general public identified in 10 CFR 20.1301.

4 SURVEILLANCE REQUIREMENTS

4.0 General

Applicability. This specification applies to all surveillance requirements for required reactor systems.

Objective. The objective is to ensure that all required reactor systems are operable when the reactor is operating.

Specifications.

a. Surveillance requirements may be deferred during reactor shutdown (except Technical Specifications 4.3.a and 4.3.e); however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practicable after reactor startup. Scheduled surveillance that cannot be performed with the reactor operating may be deferred until a planned reactor shutdown.

b. Any 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 rod drive mechanism, or the reactor safety systems shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications reviewed by the Reactor Operations Committee. A system shall not be considered operable until after it is successfully tested.

Basis. This specification relates to changes in reactor systems that are required by these Technical Specifications. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria.

4.1 Reactor Core Parameters

Applicability. This specification applies to the surveillance requirements for reactor core parameters.

Objective. The objective is to verify that the reactor does not exceed the authorized limits for power, shutdown margin, and core excess reactivity; to verify that the specifications for fuel element condition are met; and to verify the total reactivity worth of each control rod.

Specifications.

a. A channel calibration shall be made of the power level monitoring channels by the calorimetric method annually.

b. The total reactivity worth of each control rod shall be measured annually or following any significant change (>$0.25) from a reference core.

c. The shutdown margin shall be determined prior to each day's operation, prior to each operation extending more than one day, or following any significant change (>$0.25) from a reference core.

d. The core excess reactivity shall be determined annually or following any significant change (>$0.25) from a reference core.

e. Twenty percent of the fuel elements comprising the core shall be inspected visually for damage or deterioration and measured for concentric or other swelling annually such that the entire core is inspected over a five year period. Annual inspections shall be of non-repeating representative samples of fuel elements from each ring.

Basis. Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components. The value of a significant change in reactivity (>$0.25) is measurable and will ensure adequate coverage of the shutdown margin after taking into account the accumulation of poisons. For inspection, looking at fuel elements from each ring annually will identify any developing fuel integrity issues throughout the core. Furthermore, the observed mechanism for non-conforming fuel at the OSTR has been exclusively concentric swell. Looking for swell will both provide early indication of fuel nonconformance and significantly reduce the amount of fuel movements needed.

4.2 Reactor Control and Safety Systems

Applicability. This specification applies to the surveillance requirements of reactor control and safety systems.

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

Specifications.

a. The control rods and drives shall be visually inspected for damage or deterioration biennially.

b. The scram time shall be measured annually.

c. The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary, semi-annually.

d. A channel check of each of the reactor safety system channels for the intended mode of operation shall be performed prior to each day's operation or prior to each operation extending more than one day.

e. A channel test of each item in Table 2 in section 3.2.3 and Table 3 in section 3.2.4 shall be performed semi-annually.

Basis. Experience has shown that the identified frequencies will ensure performance and operability for each of these systems or components.

4.3 Reactor Primary Tank Water

Applicability. This specification applies to the surveillance requirements for the reactor tank water.

Objective. The objective is to ensure that the reactor tank water level and reactor tank water temperature monitoring systems are operating, to verify appropriate alarm settings, and to verify that the reactor tank water conductivity is within the limit.

Specifications.

a. A channel check of the reactor tank water level monitor shall be performed monthly.

b. A channel check of the reactor tank water temperature system, including a verification of the alarm set point, shall be performed prior to each day's operation or prior to each operation extending more than one day.

c. An operability check of the reactor tank temperature alarm shall be performed monthly.

d. A channel calibration of the reactor tank water temperature system shall be performed annually.

e. The reactor tank water conductivity shall be measured monthly.

Basis. Experience has shown that the frequencies of checks on systems that monitor reactor primary water level, temperature, and conductivity adequately keep the tank water at the proper level and temperature and maintain water quality at such a level to minimize corrosion and maintain safety.

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4.5 Ventilation System

Applicability. This specification applies to the reactor bay confinement ventilation system.

Objective. The objective is to ensure the proper operation of the ventilation system in controlling releases of radioactive material to the unrestricted area.

Specifications.

a. A channel check of the reactor bay confinement ventilation system's ability to maintain a negative pressure in the reactor bay with respect to surrounding areas shall be performed prior to each day's operation or prior to each operation extending more than one day.

b. A channel test of the reactor bay confinement ventilation system's ability to be secured shall be performed annually.

Basis. Experience has demonstrated that tests of the ventilation system on the prescribed daily and annual basis are sufficient to ensure proper operation of the system and its control over releases of radioactive material.

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4.7 Radiation Monitoring System

Applicability. This specification applies to the surveillance requirements for the area radiation monitoring equipment and the air monitoring systems.

Objective. The objective is to ensure that the radiation monitoring equipment is operating properly and to verify the appropriate alarm settings.

Specifications.

a. A channel check of the radiation monitoring systems in section 3.7.1 shall be performed prior to each day's operation or prior to each operation extending more than one day.

b. A channel test of the continuous air particulate, exhaust gas, and exhaust particulate radiation monitors shall be performed monthly.

c. A channel calibration of the radiation monitoring systems in section 3.7.1 shall be performed annually.

Basis. Experience has shown that an annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. A daily channel check and monthly channel test are adequate to ensure that the radiation monitoring equipment is operating correctly.

4.8 Experimental Limits

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

Objective. The objective is to prevent the performance of experiments that may damage the reactor or release excessive amounts of radioactive materials as a result of experiment failure.

Specifications.

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

b. An experiment shall not be installed in the reactor or its irradiation facilities unless a safety analysis has been performed and reviewed for compliance with Section 3.8 by the Reactor Operations Committee in full accord with Section 6.2.3, and the procedures that are established for this purpose.

Basis. Experience has shown that experiments that are reviewed by the staff of the OSTR and the Reactor Operations Committee can be conducted without endangering the safety of the reactor or exceeding the limits in the Technical Specifications.

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5 DESIGN FEATURES

5.1 Site and Facility Description

Applicability. This specification applies to the Oregon State TRIGA® Reactor site location and specific facility design features.

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

Specifications.

a. The restricted area is that area inside the fence surrounding the reactor building and the reactor building itself. The unrestricted area is that area outside the fence surrounding the reactor building.

b. The reactor building houses the TRIGA® reactor and is abutted to the Oregon State University Radiation Center Building.

c. The reactor bay shall be equipped with ventilation systems designed to exhaust air or other gases from the reactor building and release them from a stack at a minimum of 65 feet from ground level.

d. Emergency shutdown controls for the ventilation systems shall be located in the reactor control room.

Basis. The Radiation Center, reactor building, and site description are strictly defined. The facility is designed such that the ventilation system will normally maintain a negative pressure in the Reactor Building with respect to the outside atmosphere so that there will be no uncontrolled leakage to the unrestricted environment. Controls for startup and normal operation of the ventilation system are located in the reactor control room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the reactor control room with a minimum of exposure to operating personnel.

5.2 Reactor Coolant System

Applicability. This specification applies to the tank containing the reactor and to the cooling of the core by the tank 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.

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

b. The tank water inlet and outlet pipes to the heat exchanger and to the demineralizer shall be equipped with siphon breaks not less than 14 feet above the top of the core.

c. A tank water level alarm shall be provided to indicate loss of coolant if the tank level drops more than 6 inches below normal level.

d. A tank water temperature alarm shall be provided to indicate high tank water temperature if the temperature exceeds 49°C (120°F).

Basis.

a. This specification is based on thermal and hydraulic calculations that show that the TRIGA® core can operate in a safe manner at power levels up to 1.9 MW with natural convection flow of the coolant water.

b. In the event of accidental siphoning of tank water through inlet and outlet pipes of the heat exchanger or demineralizer system, the tank water level will drop to a level no less than 14 feet from the top of the core.

c. Loss-of-coolant alarm caused by a water level drop of no more than 6 inches provides a timely warning so that corrective action can be initiated. This alarm is located in the control room.

d. The water temperature alarm provides warning so that corrective action can be initiated in a timely manner to protect the quality of the reactor tank. The alarm is located in the control room.

5.3 Reactor Core and Fuel

5.3.1 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 elements and experiments so as to provide assurance that excessive power densities shall not be produced.

Specifications.

a. The core assembly shall consist of TRIGA® fuel elements.

b. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, irradiation facilities, graphite reflector elements, aluminum dummies, control rods, and startup sources.

c. The reactor shall not be operated at power levels exceeding 1 kW with a core lattice position water filled, except for positions on the periphery of the core assembly.

d. The reflector, excluding experiments and irradiation facilities, shall be water or a combination of graphite and water.

Basis.

a. Only TRIGA® fuel is anticipated to ever be used.

b. In-core water-filled experiment positions have been demonstrated to be safe in the Gulf Mark III reactor. The largest values of flux peaking will be experienced in hydrogenous in-core irradiation positions. Various non-hydrogenous experiments positioned in element positions have been demonstrated to be safe in TRIGA® fuel element cores up to 2 MW operation.

c. For cases where one in-core position is water filled, except in the core periphery, the maximum reactor power level is reduced to 1 kW to ensure safe peak power generation levels in adjacent element positions.

d. The core will be assembled in the reactor grid plate, which is located in a tank of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of irradiation facility radiation requirements.

5.3.2 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.

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

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

Basis. The poison requirements for the control rods are satisfied by using neutron absorbing borated graphite, B_4C powder, or boron. These materials shall 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 tank water environment. Control rods that are fuel-followed provide additional reactivity to the core and increase the worth of the control rod. The use of fueled-followers in the fueled region has the additional advantage of reducing flux peaking in the water-filled regions vacated by the withdrawal of the control rods. Scram capabilities are provided for rapid insertion of the control rods, which is the primary safety feature of the reactor. The transient control rod is designed for rapid withdrawal from the reactor core, which results in a reactor pulse.

The nuclear behavior of the air or aluminum follower, which may be incorporated into the transient rod, is similar to a void. A voided follower may be required in certain core loadings to reduce flux peaking values.

5.3.3 Reactor Fuel

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

Objective. The objective is to ensure that the fuel elements 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. TRIGA® Fuel Elements

The individual unirradiated fuel elements shall have the following characteristics:

a. Uranium content: nominal 30 wt% enriched to less than 20% in ^{235}U;

b. Hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.5 and 1.65;

c. Natural erbium content (homogeneously distributed): nominal 1.1 wt%;

d. Cladding: 304 stainless steel, nominal 0.020 inches thick; and

e. Identification: top pieces of fuel elements will have characteristic markings to allow visual identification of elements.

Basis. Material analysis of OSTR 30/20 fuel shows that the maximum weight percent of uranium in any fuel element is less than 30.5 percent, and the maximum enrichment of any fuel element is less than 20.0 percent. The minimum erbium content of any fuel element is greater than 1.0 percent. The hydrogen to zirconium ratio for all fuel elements is between 1.55 and 1.65. An element loaded with the maximum ^{235}U content and minimum erbium content would result in an increase in power density of no more than 2.4% over an element with nominal uranium and erbium loading. An increase in the local power density of 2.4% reduces the safety margin by, at most, 4%. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element cladding of about a factor of two greater than the value resulting from a hydrogen-to-zirconium ratio of 1.60. However, this increase in the cladding stress during an accident would not exceed the rupture strength of the cladding.

5.4 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 fuel that is being stored shall not become critical and shall not reach an unsafe temperature.

Specifications.

a. All fuel elements shall be stored in a geometrical array where the k-effective does not exceed 0.90 for all conditions of moderation and reflection.

b. Irradiated fuel elements and fuel devices shall be stored in an array that will permit sufficient natural convection cooling by water or air such that the temperature of the fuel element or fuel device will not exceed the safety limit.

Basis. The limits imposed are conservative and ensure safe storage (NUREG-1537).

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6 ADMINISTRATIVE CONTROLS

6.1 Organization

Individuals at the various management levels, in addition to being responsible for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, technical specifications, and federal regulations.

6.1.1 Structure

The reactor administration shall be related to the University as shown in Figure 1.

6.1.2 Responsibility

The following specific organizational levels, and responsibilities shall exist:

a. Vice President for Research (Level 1): The Vice President for Research is responsible for university center and institute organizations representing Oregon State University.

b. Radiation Center Director (Level 2): The Radiation Center Director reports to the Vice President for Research and is accountable for ensuring that all regulatory requirements, including implementation, are in accordance with all requirements of the USNRC and the Code of Federal Regulations.

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

d. Senior Health Physicist (Level 3): The Senior Health Physicist reports to the Radiation Center Director and is responsible for directing the activities of health physics personnel, including implementation of the radiation safety program.

e. Reactor Supervisor (Level 3): The Reactor Supervisor reports to the Reactor Administrator and is responsible for directing the activities of the reactor operators and senior reactor operators and for the day-to-day operation and maintenance of the reactor.

f. Reactor Operator and Senior Reactor Operator (Level 4): The Reactor Operator and Senior Reactor Operator report to the Reactor Supervisor and are primarily involved in the manipulation of reactor controls, monitoring of instrumentation, and operation and maintenance of reactor related equipment.

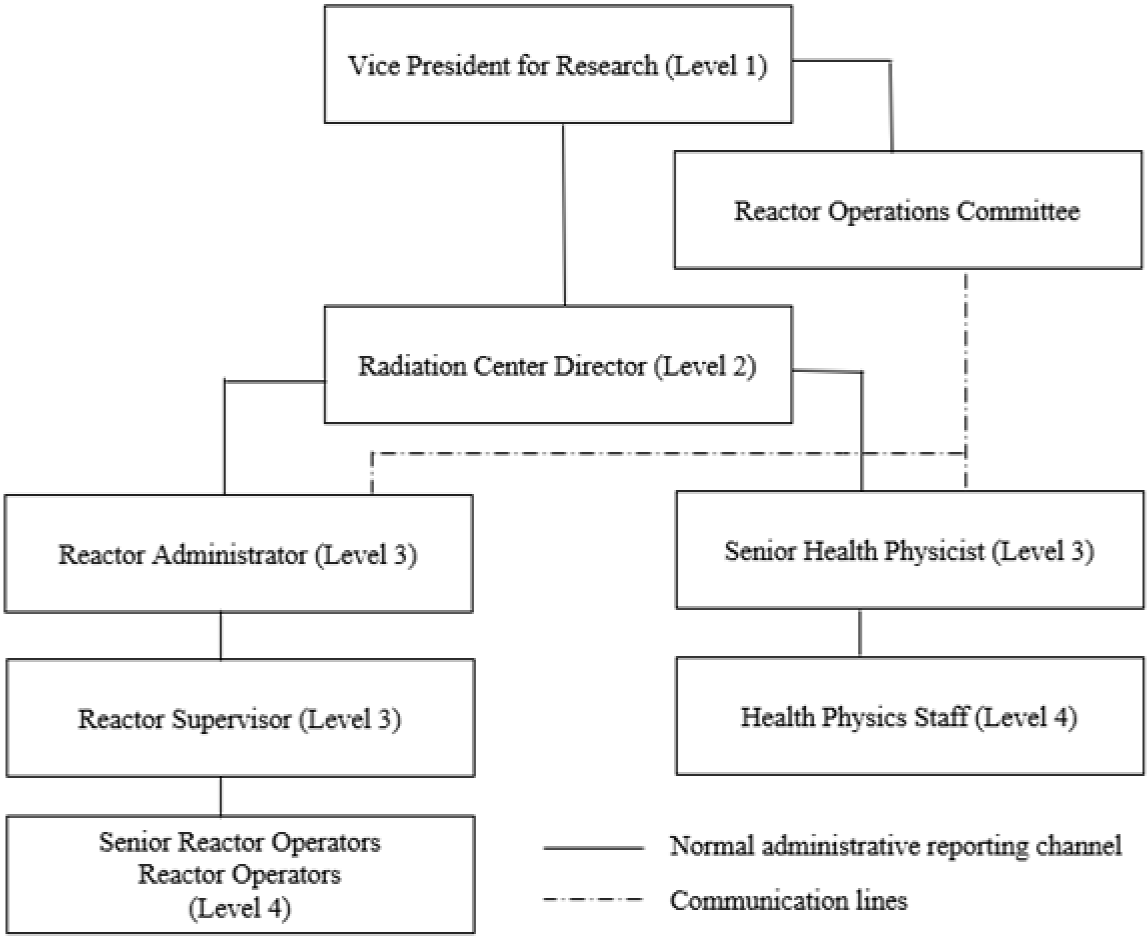

Figure 1 - Administrative Structure

6.1.3 Staffing

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

1. A reactor operator or the senior reactor operator on duty (Duty SRO) in the control room;
2. A second person present in the Radiation Center Complex able to carry out prescribed instructions; and
3. If neither of these two individuals is the Duty SRO, the Duty SRO shall be readily available on call. Readily available on call means an individual who:
    i. Has been specifically designated and the designation is known to the operator on duty;
    ii. Can be rapidly contacted by phone by the operator on duty; and

iii. Is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15 mile radius).

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

  1. Radiation Center Director
  2. Reactor Administrator
  3. Senior Health Physicist
  4. Any Licensed Reactor or Senior Reactor Operator

c. Events requiring the direction of the Duty SRO:

  1. Initial startup and approach to power of the day;
  2. All fuel or control rod relocations within the reactor core region;
  3. Relocation of any in-core experiment or irradiation facility with a reactivity worth greater than one dollar; and
  4. Recovery from unplanned or unscheduled shutdown or significant power reduction.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel should be in accordance with ANSI/ANS 15.4 – 2016, “Standard for the Selection and Training of Personnel for Research Reactors.”

6.2 Review and Audit

The Reactor Operations Committee (ROC) shall have primary responsibility for review and audit of the safety aspects of reactor facility operations. Minutes, findings, or reports of the ROC shall be presented to Level 1 and Level 2 management within ninety (90) days of completion.

6.2.1 ROC Composition and Qualifications

An ROC of at least five (5) members knowledgeable in fields that relate to reactor engineering and nuclear safety shall review and evaluate the safety aspects associated with the operation and use of the facility. The ROC shall be appointed by Level 1 management.

6.2.2 ROC Rules

The operations of the ROC shall be in accordance with written procedures including provisions for:

  a. Meeting frequency (at least annually);

b. Voting rules;

c. Quorums (5 members, no more than two voting members may be of the operating staff at any time);

d. Method of submission and content of presentation to the committee;

e. Use of subcommittees; and

f. Review, approval, and dissemination of minutes.

6.2.3 ROC Review Function

The responsibilities of the ROC, or designated Subcommittee thereof, include, but are not limited to, the following:

a. Review all changes made under 10 CFR 50.59;

b. Review of all new procedures and substantive changes to existing procedures;

c. Review of proposed changes to the technical specifications, license, or charter;

d. Review of violations of technical specifications, license, or violations of internal procedures or instructions having safety significance;

e. Review of operating abnormalities having safety significance;

f. Review of all events from reports required in sections 6.6.1 and 6.7.2 of these Technical Specifications;

g. Review of audit reports.

6.2.4 ROC Audit Function

The ROC or a Subcommittee thereof shall audit reactor operations at least annually. The annual audit shall include at least the following:

a. Facility operations for conformance to the technical specifications and applicable license or charter conditions;

b. The retraining and requalification program for the operating staff;

c. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety; and

d. The Emergency Response Plan and implementing procedures.

6.3 Radiation Safety

The Senior Health Physicist shall be responsible for implementation of the radiation safety program. The requirements of the radiation safety program are established in 10 CFR 20. The program should use the guidelines of the ANSI/ANS 15.11 – 2016, “Radiation Protection at Research Reactor Facilities”.

6.4 Procedures

Written procedures shall be adequate to ensure the safety of operation of the reactor, but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be in effect for the following items:

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

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

c. Maintenance of major components of systems that could have an effect on reactor safety;

d. Surveillance checks, calibrations, and inspections required by the technical specifications or those that have an effect on reactor safety;

e. Radiation protection;

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

g. Shipping of radioactive materials; and

h. Implementation of the Emergency Response Plan.

Substantive changes to the above procedures shall be made only after review by the ROC. Except for radiation protection procedures, unsubstantive changes shall be approved prior to implementation by the Reactor Administrator and documented by the Reactor Administrator within 120 days of implementation. Unsubstantive changes to radiation protection procedures shall be approved prior to implementation by the Senior Health Physicist and documented by the Senior Health Physicist within 120 days of implementation.

Temporary deviations from the procedures may be made by the responsible Senior Reactor Operator in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported by the next working day to the Reactor Administrator.

6.5 Experiments Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures. Procedures related to experiment review and approval shall include:

a. All new experiments or class of experiments shall be reviewed by the ROC and approved in writing by the Level 2 or designated alternates prior to initiation; and

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

6.6 Required Actions

6.6.1 Actions to Be Taken in Case of Safety Limit Violation

In the event a safety limit (fuel temperature) is exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the USNRC;

b. An immediate notification of the occurrence shall be made to the Reactor Administrator, Radiation Center Director, and Chair of the ROC; and

c. A report, and any applicable follow-up report, shall be prepared and reviewed by the ROC. The report shall describe the following:

    1. Applicable circumstances leading to the violation including, when known, the cause and contributing factors;

    2. Effects of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public; and

    3. Corrective action to be taken to prevent recurrence.

6.6.2 Actions to Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2 Other than a Safety Limit Violation

For all events that are required by regulations or Technical Specifications to be reported to the NRC by the following working day under Section 6.7.2, except a safety limit violation, the following actions shall be taken:

a. The reactor shall be secured and the Reactor Administrator and Director notified;

b. Operations shall not resume unless authorized by the Reactor Administrator and Director;

c. The Reactor Operations Committee shall review the occurrence at their next scheduled meeting; and

d. A report shall be submitted to the NRC in accordance with Section 6.7.2 of these Technical Specifications.

6.7 Reports

6.7.1 Annual Operating Report

An annual report shall be created and submitted by the Radiation Center Director to the USNRC by November 1 of each year consisting of:

a. A brief summary of operating experience including the energy produced by the reactor and the hours the reactor was critical;

b. The number of unplanned shutdowns, including reasons therefor;

c. A tabulation of major preventative and corrective maintenance operations having safety significance;

d. 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;

e. 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 prior to the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed or recommended, a statement to this effect is sufficient;

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

g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed.

6.7.2 Special Reports

In addition to the requirements of applicable regulations, and in no way substituting therefor, reports shall be made by the Radiation Center Director to the NRC as follows:

a. A report not later than the following working day by telephone and confirmed in writing by facsimile or email to the NRC Operations Center, to be followed by a written report that describes the circumstances of the event within 14 days to the NRC Document Control Desk of any of the following:

    1. Violation of the safety limit;

    2. Release of radioactivity from the site above allowed limits;

    3. Operation with actual safety system settings from required systems less conservative than the limiting safety system setting;

4. Operation in violation of limiting conditions for operation unless prompt remedial action is taken as permitted in Section 3;
5. A reactor safety system component malfunction that renders or could render the reactor safety system incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required;
6. An unanticipated or uncontrolled change in reactivity greater than one dollar. Reactor trips resulting from a known cause are excluded;
7. Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable; or
8. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

b. A report within 30 days in writing to the NRC, Document Control Desk, Washington, D.C. of:

1. Permanent changes in the facility organization involving Level 1-2 personnel; and
2. Significant changes in the transient or accident analyses as described in the Safety Analysis Report.

6.8 Records

6.8.1 Records to Be Retained for a Period of at Least Five Years or for the Life of the Component Involved if Less than Five Years

a. Normal reactor operation (but not including supporting documents such as checklists, log sheets, etc., which shall be maintained for a period of at least one year);

b. Principal maintenance activities;

c. Reportable occurrences;

d. Surveillance activities required by the Technical Specifications;

e. Reactor facility radiation and contamination surveys;

f. Experiments performed with the reactor;

g. Fuel inventories, receipts, and shipments;

h. Approved changes to the operating procedures; and
i. Reactor Operations Committee meetings and audit reports.

6.8.2 Records to Be Retained for at Least One Certification Cycle

Records of retraining and requalification of certified reactor operators and senior reactor operators shall be retained at all times the individual is employed or until the certification is renewed.

6.8.3 Records to Be Retained for the Lifetime of the Reactor Facility

a. Gaseous and liquid radioactive effluents released to the environs;
b. Offsite environmental monitoring surveys;
c. Radiation exposures for all personnel monitored;
d. Drawings of the reactor facility; and
e. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.