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October 5, 2004

U.S. Nuclear Regulatory Commission
ATTN: Mr. Alexander Adams
Office of Nuclear Reactor Regulation
Mail Stop 12 G13
Washington, DC 20555

Reference: Oregon State University TRIGA Reactor (OSTR)
Docket No. 50-243, License No. R-106

Subject: License Renewal of the OSTR

Mr. Adams,

Oregon State University hereby requests a renewal of Operating-License No. R-106 for the OSTR, a Class 104 license utilized for research and education. We are submitting this request early since the existing license does not expire until August 15, 2006. The requested renewal term for License R-106 is for 20 years.

Attached you will find copies of the updated safety analysis report (SAR), technical specifications, and the environmental report. Both the SAR and the technical specifications conform to the format given in NUREG-1537. Financial qualifications are addressed in chapter 15 of the SAR. We do not wish to make any changes to the OSTR Emergency plan, Physical Security Plan, or the Operator Requalification Program as they are current and up-to-date.

If you have any questions pertaining to this matter, please direct them to Dr. Klein at the address given at the top of this letter. We declare under penalty of perjury that the foregoing is true and correct.

Executed on: October 5, 2004.

Sincerely,

George R. (Rich) Holdren
Vice Provost for Research

Andrew C. Klein
Radiation Center Director

A035
A020

Attachments

cc: Document Control Desk, USNRC
Craig Bassett, USNRC
David Stewart-Smith, ODOE
Rich Holdren, OSU (w/o attachments)
John Ringle, OSU (w/o attachments)
Steve Reese, OSU (w/o attachments)

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 #XX
Date of Issuance: XXXX XX, XXXX

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TABLE OF CONTENTS

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.

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1 DEFINITIONS

1.1 Audit: A quantitative examination of records, procedures or other documents.

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

1.3 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 which the channel measures. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall include a Channel Test.

1.4 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.

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

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

1.7 Control Rod: A control rod is a device fabricated from neutron absorbing material or fuel which is used to establish neutron flux changes and to compensate for routine reactivity losses. 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 (Reg 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.

1.8 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.

1.9 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).

1.10 Experiment: Any operation, hardware, or target (excluding devices such as detectors or foils) which is designed to investigate non-routine reactor characteristics or which 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 or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment.

b. **Movable Experiment:** A movable experiment is one where it is intended that the entire experiment may be moved in the core while the reactor is operating.

1.11 Experimental Facilities: Experimental 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 Experiment Safety Systems: Experiment safety systems are those systems, including their associated input channel, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

1.13 Fuel Element: A fuel element is a single TRIGA® fuel rod of either standard or Fuel Lifetime Improvement Program (FLIP) type.

1.14 Instrumented Element: An instrumented element is a special fuel element in which one or more thermocouples have been embedded for the purpose of measuring the fuel temperatures during reactor operation.

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

1.16 Operable: A system or component shall be considered operable when it is capable of performing its intended function.

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

1.18 Operational Core: An operational core may be a standard, FLIP, or mixed core which operates within the licensed power level and satisfies all the requirements of the Technical Specifications. Operational cores shall include:

a. **Standard Core:** A standard core is an arrangement of standard TRIGA® fuel in the reactor grid plate.

b. **Flip Core:** A FLIP core is an arrangement of FLIP fuel elements in the reactor grid plate.

c. **Mixed Core:** A mixed core is an arrangement of standard TRIGA® and FLIP fuel elements with the FLIP fuel elements located in the central region of the core.

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

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

1.21 Reactor Operation: Reactor operation is any condition wherein the reactor is not secured or shut down.

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

1.23 Reactor Secured: The reactor is secured when:

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

b. A combination of the following:

1. The reactor is shut down;

2. No experiments or experimental facilities in the core are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value of one dollar; and

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

1.24 Reactor Shutdown: The reactor is shut down when:

a. if it is subcritical by at least one dollar both in the reference core condition and for all allowed ambient conditions with the reactivity worth of all installed experiments and experimental facilities included; and

b. The console key switch is in the "off" position and the key is removed from the console.

1.25 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 (<0.30 dollars).

1.26 Review: A qualitative examination of records, procedures or other documents.

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

1.28 Secured Shutdown: Secured shutdown is achieved when the reactor is secured and the facility administrative requirements for leaving the facility with no licenced reactor operators present have been met.

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

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

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

1.32 Shutdown Reactivity: Shutdown reactivity is the measured reactivity with all control rods in their least reactive positions (e.g., inserted). The value of shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

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

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

1.35 Substantive Changes: Substantive changes are changes in the original intent of the outcome or safety significance.

1.36 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.

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2 SAFETY LIMITS 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

a. The temperature in a TRIGA®-FLIP fuel element shall not exceed 2,100° F (1,150° C) under any mode of operation.

b. The temperature in a standard TRIGA® fuel element shall not exceed 1,830° F (1,000° C) under any mode of operation.

Basis. The important parameter for a TRIGA® reactor is the fuel element temperature. This parameter is well suited as a single specification especially since it can be measured. 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®-FLIP fuel element is based on data which 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 2100° F (1150° C) and the fuel cladding is water cooled (SAR 4.5.3.1).

The safety limit for the standard TRIGA® fuel is based on data including the large mass of experimental evidence obtained during high performance reactor tests on this fuel. These data indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1,830° F (1,000° C) and the fuel cladding is water cooled (SAR 4.5.3.1).

2.2 Limited Safety System Setting

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

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

Specifications The limiting safety system setting shall be 510°C (950°F) as measured in an instrumented fuel element. The instrumented fuel element shall be located in the B-ring.

Basis. The limiting safety system setting is a temperature, which, if exceeded shall cause the reactor safety system to initiate a reactor scram. This setting applies to all modes of operation. In steady-state operation up to 1.1 MW, ample margins exist between this setting and the safety limits of 1150°C and 1000°C for FLIP and Standard fuel, respectively (SAR 4.5.3.1).

The highest fuel temperatures are experienced during pulse transients, initiated from low power. The fuel temperature scram, to which the limiting safety system setting applies, can prevent reaching the safety limit of the fuel by reducing the energy released in the "tail" of the pulse. A setting of 510°C is conservatively estimated to provide the largest permissible pulses. These estimates are obtained from calculations based on an adiabatic reactor kinetics model. This model was applied to existing cores yields characteristics which are in good agreement with measured values (SAR 4.5.3.1).

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3 LIMITING CONDITIONS OF OPERATION

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 assure 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. Thermal and hydraulic calculations indicate that TRIGA® fuel may be safely operated up to power levels of at least 1.9 MW with natural convection cooling (SAR 4.5.3.3).

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 assure that the reactor can be shut down at all times and to assure 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 greater than $0.55 with:

1. experimental facilities and experiments in place and the highest-worth, non-secured experiment in its most reactive state;
2. the most reactive control rod fully-withdrawn; and
3. the reactor in the reference core condition.

Basis. The value of the shutdown margin assures 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 assure that the reactor can be shut down at all times and to assure 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 $10.00.

Basis. The nominal total rod worth of the most reactive core configuration (normal core) is $12.00. Subtracting the shutdown margin ($0.55) and the nominal rod worth of the most reactive rod ($4.00), and adding the allowed total reactivity worth for all experiments ($2.55), the core excess reactivity remaining shall not be allowed to exceed a value of $10.00.

3.1.4 Pulse Mode Operation

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

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

Specifications. The reactivity to be inserted for pulse operation shall be determined and limited by a mechanical block and electrical interlock on the transient rod, such that the reactivity insertion shall not exceed $2.55.

Basis. The fuel temperature rise during a pulse transient has been estimated conservatively by adiabatic models. These models accurately predict pulse characteristics for operation of mixed cores and FLIP cores and should be accepted with confidence, relying also on information concerning prompt neutron lifetime and prompt temperature coefficient of reactivity. These parameters have been established for mixed and full FLIP cores by calculations and have been confirmed in part by measurements at existing facilities. In addition, the calculations rely on flux profiles and corresponding power densities which have been calculated for a variety of operational mixed and full FLIP cores.

In this manner, it is estimated conservatively that reactivity insertions up to $2.55 in operational cores should produce pulse transients with maximum fuel temperatures no

greater than 950°C in FLIP fuel and 800°C in standard fuel; i.e., a safety margin of 200°C with respect to the safety limit of the fuel is maintained in either case, allowing for any uncertainties in measurements and/or calculations (SAR 13.2.2.2.1).

3.1.5 Core Configuration Limitations

Applicability. This specification applies to mixed cores of FLIP and standard types of fuel.

Objective. The objective is to assure that the fuel temperature safety limit shall not be exceeded due to power peaking effects in a mixed core.

Specifications. The FLIP-fueled region in a mixed core shall contain at least 80 FLIP fuel elements in a contiguous block of fuel in the central region of the reactor core. Single element positions may be left vacant or occupied by other items as specified in Sections 5.3.1.c and 5.3.1.d of these Technical Specifications.

Basis. The limitation of power peaking effects ensures that the fuel temperature safety limit shall not be exceeded in an operational core (SAR 4.5.2).

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 must 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.

Specification. 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 assures 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 assure the safety of the reactor (SAR 13.2.2.2.1).

3.2.2 Reactor Measuring Channels

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

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

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

Table 1 - Minimum Measuring Channels

Measuring Channel	Effective Mode		
	S.-S.	Pulse	S.-W.
Fuel Element Temperature	1	1	1
Linear Power Level	1	-	1
Log Power Level	1	-	1
Power Level	1	-	1
Nvt-Circuit	-	1	-

Basis. Fuel temperature displayed at the control console gives continuous information on this parameter which has a specified safety limit (SAR 7.2.3.2). The power level monitors assure that the reactor power level is adequately monitored for both steady-state, square wave and pulse 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 (SAR 7.2.3.1).

3.2.3 Reactor Safety System

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

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

Specifications. The reactor shall not be operated unless the minimum number of safety channels described in Table 2 and interlocks described in Table 3 are operable.

Table 2 - Minimum Reactor Safety Channels

Safety Channel	Function	Effective Mode		
		S.-S.	Pulse	S.-W.
Fuel Element Temperature	SCRAM @ 510°C	1	-	1
Power Level	SCRAM @ 1.1 MW(t) or less	1	-	1
Console Scram Button	SCRAM	1	-	1
High Voltage	SCRAM @ ≤25% of nominal operating voltage	1	1	1

Table 3 - Minimum 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	Prevents pulse insertion of reactivity greater than $2.55	-	1	1

Basis. The fuel temperature, and power level scrams provide protection to assure that the reactor can be shut down before the safety limit on the fuel element temperature will be exceeded. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. The high voltage scram ensures that the power measuring channels operate within their intended range as required for proper functioning of all power level scrams. The interlock to prevent startup of the reactor at count rates less than 2 cps assures that the startup is not initiated unless a reliable indication of the neutron flux level in the reactor core is available. The interlock to prevent the initiation of a pulse above 1 kW is to assure that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing the reactor in the steady-state mode. The interlock to prevent withdrawal of the shim, safety or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period. The interlock to prevent simultaneous withdrawal of two control rods is to limit reactivity insertion rate from the standard control rods. The interlock on the transient rod cylinder position prevents the pulse insertion of more than $2.55 of reactivity during the pulse or square-wave mode (SAR 7.3.3).

3.3 Reactor Primary Tank Water

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

Objective. The objective is to assure that there is an adequate amount of water in the reactor tank for fuel cooling and shielding purposes, and that the bulk 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 greater than 14 feet above the top of the core;

b. the bulk tank water temperature shall be less than 120°F (49°C); or

c. the conductivity of the tank water shall be less than 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 (SAR 4.3, 4.5.3, and 11.1.5.5). The bulk water temperature limit is necessary, according to the reactor manufacturer, to ensure that the aluminum reactor tank maintains its integrity and is not degraded (SAR 4.3). Experience at many research reactor facilities has shown that maintaining the conductivity within the specified limit provides acceptable control (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 assure 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 2 hours to permit repair, maintenance or testing of the ventilation system.

b. In the event of a substantial release of airborne radioactivity, the ventilation system shall be shut down.

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 limit for individual members of the public. This has been shown to be true for scenarios were 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 13.2.1). 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 or radioactive material to the unrestricted area nor should it cause occupational doses that exceed those limits given in 10 CFR 20 (SAR 11.1.1.1.2). 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 11.1.4.2).

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 which must be available to the reactor operator during reactor operation.

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

Specifications. The reactor shall not be operated unless the minimum number of radiation monitoring channels listed in Table 4 are operating except for time periods of up to 2 hours for repair and maintenance provided:

a. the ventilation system is operating; and

b. the continuous air particulate radiation monitor or both the exhaust gas and particulate monitors are operating.

Each channel shall have a readout in the control room and be capable of sounding an audible alarm which can be heard in the reactor control room.

Table 4 - Minimum Radiation Monitoring Channels

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

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 (SAR 11.1.4.1).

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 4×10^{-6} μCi/ml at the point of discharge.

Basis. If ^{41}Ar is continuously discharged at 2.5×10^{-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 TEDE of 5 mrem (SAR 11.1.1.1.1). This is only 50% of the applicable limit of 10 mrem (Regulatory Guide 4.20). Therefore, an emission of 4×10^{-6} μCi/ml would correspond to an annual TEDE of 8 mrem which is still 20% below the applicable limit.

3.8 Limitations on Experiments

3.8.1 Reactivity Limits

Applicability. This specification applies to experiments installed in the reactor and its experimental 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. movable experiments shall have reactivity worths less than \$1;

b. the reactivity worth of any single secured experiment shall be less than \$2.55;

c. total experiment worth of all experiments shall be less than \$2.55;

Basis. The worst event which could possibly arise is the sudden removal of a movable experiment immediately prior to, or following, a pulse transient of the maximum licensed reactivity insertion. Limiting the worth of the movable experiment to less than \$1 will assure that the additional increase of transient power and temperature is slow enough for the fuel temperature scram to be effective (SAR 7.2.3.1 and 13.2.2).

The worst event which may be considered in conjunction with a single secured experiment is its sudden removal while the reactor is operating in a critical condition at a low power level.

This is equivalent to pulse-mode operation of the reactor. Hence, the reactivity limitation for a single secured experiment is the same as that for pulsing (SAR 13.2.2.2.1).

It is conservatively assumed that simultaneous removal of all experiments in the reactor at any given time shall not exceed the maximum reactivity insertion limit of $2.55 (SAR 13.2.2.2.1).

3.8.2 Materials

Applicability. This specification applies to experiments installed in the reactor and its experimental 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 experimental facilities. Explosive materials in quantities less than 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 the design pressure of the container. EXCEPTION: Explosive materials not exceeding 0.014 lbs-equivalent of TNT may be irradiated in the laboratory area adjacent to the end of the OSTR tangential beamport for the purpose of neutron radiography; and

b. each fueled experiment shall be controlled such that the total inventory of ^{131}I - ^{135}I in the experiment is no greater than 1.5 curies.

Basis. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials. The 1.5-curie limitation on ^{131}I - ^{135}I assures that in the event of a failure of a fueled-experiment involving total release of the iodine, the dose in the reactor bay and in the unrestricted area will be considerably less than that allowed by 10 CFR 20 (SAR 13.2.6).

3.8.3 Failures and Malfunctions

Applicability. This specification applies to experiments installed in the reactor and its experimental 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. Where the possibility exists that the failure of an experiment (except fueled experiments) 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. if the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape;

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

d. 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.

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4 SURVEILLANCE REQUIREMENTS

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 maximum power level of the reactor does not exceed the authorized limits for power, shutdown margin, and core excess reactivity.

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 in core or control rod configuration.

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 in core or control rod configuration.

d. The core excess reactivity shall be determined annually or following any significant change in core or control rod configuration.

e. All fuel elements shall be inspected visually for damage or deterioration and measured for length and bend at intervals not to exceed the sum of $3,500 in pulse reactivity.

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

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 which are directly related to reactor safety.

Specifications.

a. The control rods 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 I and II in section 3.2.3 other than measuring channels, shall be performed semi-annually.

f. A channel calibration of the fuel temperature measuring channel shall be performed 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 assure that the reactor tank water level and the bulk water temperature monitoring systems are operating, and to verify appropriate alarm settings.

Specifications.

a. A channel check of the reactor tank water level shall be performed semi-annually.

b. A channel check of the reactor tank water temperature system shall be performed prior to each day's operation or prior to each operation extending more than one day.

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

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

Basis. Experience has shown that the frequencies of checks on systems which monitor reactor primary water can adequately keep 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 assure 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 semi-annual basis are sufficient to assure proper operation of the system and its control over releases of radioactive material.

4.6 This section intentionally left blank.

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 assure 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.

4.8 Experimental Limits

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

Objective. The objective is to prevent the conduct of experiments which 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 experimental 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 of these Technical Specifications, and the procedures which are established for this purpose.

Basis. Experience has shown that experiments which 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 and 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 reactor building and the fence surrounding the reactor building.

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

c. The reactor building 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 control room.

Basis. The Radiation Center, reactor building and site description are strictly defined (SAR 2.0). The facility is designed such that the ventilation system will normally maintain a negative pressure 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 control room. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the control room with a minimum of exposure to operating personnel (SAR 9.1 and 13.2.1).

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 assure 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 level alarm shall be provided to indicate loss of coolant if the tank level drops approximately 6 inches below normal level.

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

Basis.

a. This specification is based on thermal and hydraulic calculations which show that the TRIGA®-FLIP core can operate in a safe manner at power levels up to 1.9 MW with natural convection flow of the coolant water (SAR 4.5.3.3).

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 (SAR 5.2).

c. Loss-of-coolant alarm caused by a 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 (SAR 5.2).

d. The bulk 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 (SAR 7.2.3.2).

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 assure 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 shall be an arrangement of TRIGA® uranium-zirconium hydride fuel-moderator elements positioned in the reactor grid plate.

b. The TRIGA® core assembly may consist of standard fuel elements, FLIP fuel elements, or a combination thereof (mixed core). Any operational mixed core assembly shall have no less than 80 FLIP fuel elements, located in a contiguous, central region.

c. The fuel shall be arranged in a close-packed configuration except for single element positions occupied by in-core experiments, experimental facilities, graphite dummies, aluminum dummies, stainless steel dummies, control rods, and startup sources.

d. 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.

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

Basis.

a. Standard TRIGA® cores have been in use for years and their characteristics are well documented. The Puerto Rico Nuclear Center, the Gulf Mark III all-FLIP cores and the Texas A&M core are operational and characteristics are available. Gulf has also performed a series of experiments using standard and FLIP fuel in mixed cores. In addition, analytic studies performed at OSTR for a variety of mixed core arrangements indicate that such cores with mixed loadings would safely satisfy all operational requirements (SAR 4.2).

b. In mixed cores, it is necessary to arrange FLIP elements in a contiguous, central region of the core to control flux peaking and power generation peak values in individual elements (SAR 4.2).

c. 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 experimental positions. Various non-hydrogenous experiments positioned in element positions have been demonstrated to be safe in standard and FLIP cores up to 2-MW operation (SAR 4.2).

d. 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 (SAR 4.2).

e. 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 experimental facility radiation requirements (SAR 4.2).

5.3.2 Control Rods

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

Objective. The objective is to assure 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 and its compounds in solid form as a poison in aluminum or stainless steel cladding. These rods may incorporate fueled followers which have the same characteristics as the fuel region in which they are used.

b. The transient control rod shall have scram capability and contain borated graphite 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 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 and its compounds. 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 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 FLIP 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 assure 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.

a. TRIGA®-FLIP Fuel

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

1. uranium content: maximum of 9 wt% enriched to nominal 70% ^{235}U;
2. hydrogen-to-zirconium atom ratio (in the ZrH_x): between 1.5 and 1.65;
3. natural erbium content (homogeneously distributed): between 1.1 and 1.6 wt%;
4. cladding: 304 stainless steel, nominal 0.020 inches thick; and
5. identification: top pieces of FLIP elements will have characteristic markings to allow visual identification of FLIP elements employed in mixed cores.

b. Standard TRIGA® fuel

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

1. uranium content: maximum of 9.0 wt% enriched to a nominal 20% ^{235}U;
2. hydrogen-to zirconium atom ratio (in the ZrH_x): between 1.5 and 1.8; and
3. cladding: 304 stainless steel, nominal 0.020 inches thick.

Basis.

a. A maximum uranium content of 9 wt% in a TRIGA®-FLIP element is about 6% greater than the design value of 8.5 wt%. Such an increase in loading would result in an increase in power density of about 2%. Similarly, a minimum erbium content of 1.1% in an element is about 30% less than the design value. This variation would result in an increase in power density of only about 6%. An increase in local power density of 6% reduces the safety margin by, at most, 10%. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad 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 clad stress during an accident would not exceed the rupture strength of the clad. When standard and FLIP fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings.

b. A maximum uranium content of 9 wt% in a standard TRIGA® element is about 6% greater than the design value of 8.5 wt%. Such an increase in loading would result in an increase in power density of less than 6%. An increase in local power density of 6% reduces the safety margin by, at most, 10%. The maximum hydrogen-to-zirconium ratio of 1.8 could result in a maximum stress under accident conditions to the fuel element clad 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 clad stress during an accident would not exceed the rupture strength of the clad. When standard and FLIP fuel elements are used in mixed cores, visual identification of types of elements is necessary to verify correct fuel loadings.

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 assure that fuel which 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 is less than 0.9 for all conditions of moderation.

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

Basis. The limits imposed are conservative and assure 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. The minimum qualification for all members of the reactor operating staff shall be in accordance with ANSI/ANS 15.4, "Standard for the Selection and Training of Personnel for Research Reactors."

6.1.1 Structure

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

6.1.2 Responsibility

The following specific organizational levels, and responsibilities shall exist:

a. Radiation Center Director (Level 1): The Radiation Center Director is accountable for ensuring all licensing requirements, including implementation and enforcement, are in accordance with all license and USNRC requirements.

b. Reactor Administrator (Level 2): The Reactor Administrator is responsible to the Radiation Center Director for guidance, oversight, and technical support of reactor operations.

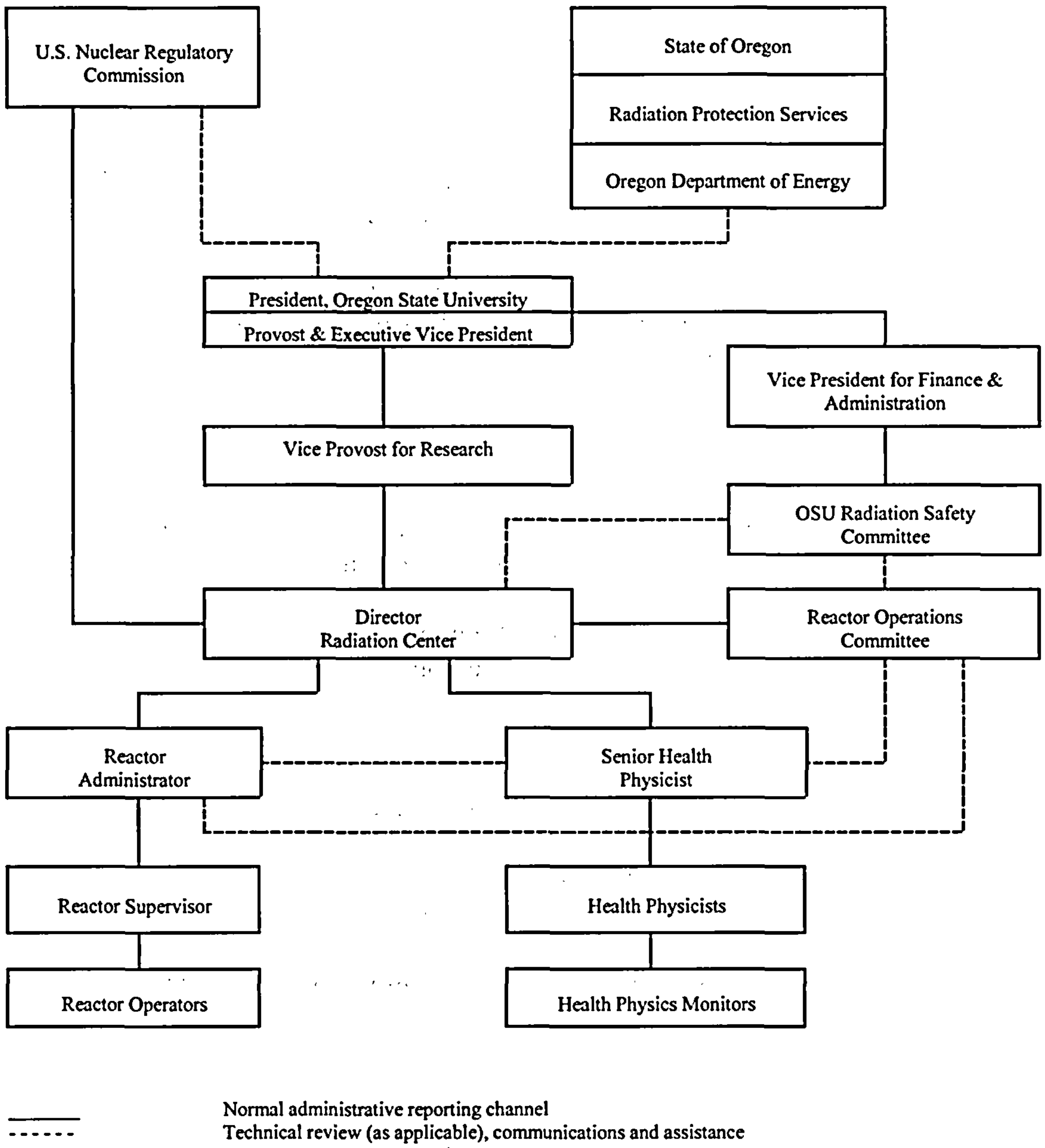
c. Senior Health Physicist (Level 2): The Senior Health Physicist is responsible to the Radiation Center Director for directing the activities of health physics personnel including implementation of the radiation safety program.

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

e. 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 operations and maintenance of reactor related equipment.

Rev. 0 7/01/2004

Figure 1 - Administrative Structure



6.1.3 Staffing

a. The minimum staffing when the reactor is operating shall be:

1. a reactor operator or a senior reactor operator in the control room;

2. a second person present in the Radiation Center Complex able to carry out prescribed written instructions; and

3. if neither of these two individuals is a senior reactor operator, a senior reactor operator 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. keeps the operator on duty informed of where they may be rapidly contacted and the phone number; 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. Events requiring the direction of a senior reactor operator

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

2. relocation of any in-core experiment or experimental facility with a reactivity worth greater than one dollar;

3. recovery from unplanned or unscheduled shutdown; and

4. initial startup of each day's operation or prior to each operation extending more than one day.

6.1.4 Selection and Training of Personnel

The selection, training and requalification of operations personnel shall be in accordance with ANSI/ANS 15.4, "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.

6.2.1 Composition and Qualifications

An ROC of at least three members knowledgeable in fields which relate to reactor engineering and nuclear safety shall review, evaluate, and approve safety standards associated with the operation and use of the facility. The jurisdiction of the ROC shall include all nuclear operations in the facility and associated general safety standards.

6.2.2 Charter and Rules

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

a. meeting frequency (at least annually);

b. voting rules;

c. quorums;

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 Review Function

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

1. review and approval of experiments utilizing the reactor facilities;

2. review and approval of all changes to the safety analysis report or technical specifications;

3. review and approval of all new procedures and substantive changes to existing procedures;

4. review all changes to the facility or safety evaluations under 10 CFR 50.59;

5. review of the operation and operational records of the facility;

6. review of abnormal performance of plant equipment and operating anomalies; and

7. review of all events which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours.

6.2.4 Audit Function

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

1. reactor operating records;

2. inspection of the reactor operating areas; and

3. reportable occurrences.

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, "Radiation Protection at Research Reactor Facilities".

6.4 Procedures

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

a. performing experiments and maintenance;

b. startup, operation and shutdown of the reactor;

c. emergency situations;

d. core changes and fuel movement;

e. control element removal and replacement;

f. performing preventive maintenance and calibration tests on the reactor and associated equipment;

g. administrative controls;

h. power calibration; and

i. radiation protection.

Substantive changes to the above procedure shall be made only with the approval of the ROC. Except for radiation protection procedures, unsubstantive changes shall be approved and documented by the Reactor Administrator within 30 days of implementation. Unsubstantive changes to radiation protection procedures shall be approved and documented by the Senior Health Physicist within 30 days of implementation.

6.5 Required Actions

6.5.1 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 Chairperson, ROC; and

c. reports shall be made to the USNRC in accordance with Section 6.6.2 of these Technical Specifications. The written report (required within 14 days) 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 ROC for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

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

For all events which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours under Section 6.6.2, except a safety limit violation, the following actions shall be taken:

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

b. operations shall not resume unless authorized by the Reactor Administrator;

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

d. where appropriate, a report shall be submitted to the NRC in accordance with Section 6.6.2 of these Technical Specifications.

6.6 Reports

6.6.1 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:

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

2. the number of unplanned shutdowns, including reasons therefore;

3. a tabulation of major preventative and corrective maintenance operations having safety significance;

4. 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;

5. 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;

6. a summarized result of environmental surveys performed outside the facility; and

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

6.6.2 Special Reports

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

a. a report within 24 hours by telephone or fax to the NRC Operations Center followed by a written report within 14 days that describes the circumstances of the event of any of the following:

   1. any accidental release of radioactivity above applicable limits in unrestricted areas, whether or not the release resulted in property damage, personal injury, or exposure;

   2. any violation of a safety limit;

   3. operation with a safety system setting less conservative than specified in the Technical Specifications.;

   4. operation in violation of a Limiting Condition for Operation;

   5. failure of a required reactor or experiment safety system component which could render the system incapable of performing its intended safety function unless the failure is discovered during maintenance tests or periods of reactor shutdown;

   6. an unanticipated or uncontrolled change in reactivity greater than $1.00;

   7. an observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits; or

   8. a measurable release of fission products from a fuel element;

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;

   2. significant changes in the transient or accident analyses as described in the Safety Analysis Report;

6.7 Records

6.7.1 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.

6.7.2 Records to be Retained for at Least One Training Cycle

a. Retraining and requalification of reactor operators and senior reactor operators shall be retained for at least one training cycle.

b. Records of the most recently completed cycle shall be maintained at all times the individual is employed.

6.7.3 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; and

d. drawings of the reactor facility.

Environmental Report

In Support of Application for Renewal of Operating License

Oregon State University
TRIGA Reactor
Docket No. 50-243, License No. R-106

July 2004

Table of Contents

1.0 General

This document summarizes the environmental effects that are imposed by operation of the Oregon State University TRIGA Reactor (OSTR) that is owned and operated by Oregon State University (OSU). The maximum steady-state power of the OSTR is 1.1 MW.

2.0 Site Description

The OSTR is located on the far west end of the OSU campus in Corvallis, OR. Metropolitan Corvallis has a population of approximately 50,000 and resides primarily to the north-northeast and south of the OSTR. OSU itself has a population (student and employees) of approximately 23,000.

There are no refineries, airports, chemical plants, mining facilities, manufacturing facilities water transportation routs, fuel storage facilities, military facilities, or rail yards located near (i.e., within 1 mile) the OSTR. A single rail line does run approximately 1000 ft due south. Most of the buildings in the vicinity of the OSTR are typical of that found on any major land-grant university. The only notable facility, the OSU Environmental Safety and Health Annex, is located 100 feet to the north of the OSTR. The facility processes, stores and packages all hazardous waste that is produced at OSU for transport to either a permanent storage location or a destruction facility. However, this facility does not pose a significant concern to the OSTR because of its distance from the reactor, the small quantity of hazardous materials stored, and the design and security of the building.

The OSU Radiation Center complex is an approximately 47,000-square foot facility and is comprised of three buildings. The OSTR is located in a four-story building, which is called the Reactor Building, on the north side of the Radiation Center. The Reactor Building contains primarily the main Reactor Bay, the Reactor Control Room, space for reactor mechanical equipment, two research laboratories, office space, and a small conference room. The Advanced Thermal Hydraulics Research Laboratory (ATHRL) is a high-bay facility attached to the east side of the Reactor Building; however, there is no access between these two buildings. The ATHRL houses experimental test loops. The Radiation Center Building houses classrooms, offices, a wide variety of radioisotope laboratories, a ^{60}Co irradiation facility, and a number of supporting facilities.

The site boundary of the OSTR consists of the rectangular area bounded by Jefferson Way on the south, 35th Street on the west, the Reactor Building fence on the north, and the east edge of the Radiation Center complex parking lot on the east.

3.0 Environmental Effects of OSTR Operations

3.1 Thermal Impact

The fission energy generated in the OSTR core is transferred to a closed primary coolant system, and to a secondary coolant system through a heat exchanger. The heat is then dissipated to the environment by means of a cooling tower. City water is used to replenish the secondary coolant loss mainly through evaporation. The rate of heat dissipation is comparable to that associated with local factories and other OSU laboratories.

3.2 Radiological Impact During Normal Operations

3.2.1 Environmental Monitoring

Environmental monitoring is performed by dosimetry devices, direct dose rate measurements, and sampling. The dosimetry devices monitor ambient radiation levels at locations that vary in direction and distance from the OSTR. Additionally, samples of vegetation, soil, and surface water are taken at locations that also vary in direction and distance from the OSTR. The doses recorded by the off-site dosimetry and direct dose rate measurements consistently give values of approximately 80 mrem per year and can be attributed to natural background radiation, which is about 110 mrem per year for this part of Oregon. The levels of radioactivity detected each year are consistent with naturally occurring radioactivity and comparable to values previously measured.

3.2.2 Personnel Exposure Monitoring

Each person who may use or handle radioactive materials must receive radiation safety training. Radiation exposures to reactor personnel are administratively controlled to meet ALARA (as low as reasonablely achievable) criteria. Experience from OSTR operations shows that all personnel exposures are well below the whole body dose limit of 5000 mrem per year, as specified in 10 CFR 20.

3.2.3 Solid Wastes

Solid wastes generated at the OSTR are low level wastes such as ion-exchange resins, filters, laboratory supplies and cleaning materials. Solid wastes are transferred to OSU Radiation Safety for compaction and sent to an appropriate waste disposal facility.

3.2.4 Liquid Wastes

Liquid radioactive wastes generated at the facility are discharged only to the sanitary sewer

serving the facility. Discharge of radioactive liquid effluents is controlled by sampling a waste tank before discharge. All of the liquid waste volumes and their activities are recorded. However, discharge to the sanitary sewer rarely occurs. The last time this was done was over six years ago.

3.2.5 Radioactive Gas Effluent

Radioactive gas effluent is discharged from the 65-foot confinement building exhaust stack to the environment. All nuclides are well below the regulatory effluent concentration limits given in the OSTR Technical Specifications, 10 CFR 20 Appendix B, Table 2, and USNRC Regulatory Guide 4.20.

3.2.6 Spent Fuel

Because the OSTR operates with Fuel Lifetime Improvement Program fuel, no spent fuel is generated on an annual basis. The end of life for the current fuel is approximately the year 2060 at our current operational tempo.

3.3 Radiological Impacts During Abnormal Operation

Chapter 13 of the OSTR Safety Analysis Report provides accident analysis for the OSTR. The Maximum Hypothetical Accident (MHA) for the OSTR, as it is with virtually all TRIGA type reactors, is postulated to be an instantaneous loss of coolant water, followed an instantaneous movement of volatile fission products from the fuel uniformly distributed into the reactor room air, and an instantaneous disappearance of the north wall of the confinement building. Results of these calculations for this scenario predict a maximum total effective dose equivalent to the maximally exposed member of the general public to be 19 mrem. Other scenarios were looked at. However, in all cases, doses for the general public and occupational workers were all well below the annual dose limits specified in 10 CFR 20.

4.0 Benefits of Facility Operations

The most important use of the OSTR is to provide support to academic and research programs. Instructional use of the OSTR is twofold. First, it is used extensively for classes in nuclear engineering, radiation health physics, and chemistry at both the graduate and undergraduate levels to demonstrate numerous principles that have been presented in the classroom and to produce radionuclides for experiments. Basic neutron behavior is the same in a small reactor as it is in large power reactors. Many demonstrations and instructional experiments can be performed using the OSTR which cannot be carried out with a commercial

power reactor. Second, shorter-term demonstration experiments are also performed for many undergraduate students in physics, chemistry, and biology classes, as well as for visitors from other universities, colleges, high school, and civic groups. In the past year, 13 courses from 4 different university departments directly involved the OSTR within their curriculum. In addition, the OSTR was utilized for research associated with 30 PhD and 7 MS students.

The OSTR is a unique and valuable tool for a wide variety of research applications and serves as an excellent source of neutrons and/or gamma rays. The OSTR has a number of irradiation facilities providing a wide range of neutron flux levels and neutron flux qualities, which are sufficient to meet the needs of most researchers. A total of 2,000 samples were irradiated in various OSTR facilities last year from 215 formal requests for irradiation. During the past year the OSTR accommodate 109 funded research projects utilizing 2,827 hours of reactor time, 685 hours of which involved multiple users.

5.0 Conclusions

5.1 Non-radiological

There is no non-radiological impact from continued operations of the OSTR.

5.2 Radiological

During routine operation, radiation exposure to reactor personnel are administratively controlled to meet the ALARA criteria. Routine exposure to occupational workers and the general public are well below the limits found in 10 CFR 20. Even in the event that the MHA does occur, which is extremely unlikely, the exposures to the general public would still be well below the limits specified in 10 CFR 20.