THE UNIVERSITY OF WISCONSIN

DOCKET NO. 50-156

FACILITY LICENSE

License No. R-74

1. The U.S. Nuclear Regulatory Commission ("the Commission") has found that:

   A. The application for renewal of Facility License No. R-74 filed by the University of Wisconsin ("the licensee") dated May 9, 2000, as supplemented on September 7, 2004; October 17, 2008; twice on June 16, July 8, August 11, November 22, and December 8, 2010; and January 28 and February 8, 2011 ("the application"), 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 Title 10, Chapter 1, of the *Code of Federal Regulations (*10 CFR);

   B. Construction of the University of Wisconsin Nuclear Reactor ("the facility") has been substantially completed, in conformity with the construction permit number CPRR-97 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 (i) that the activities authorized by this license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

F. The applicable provisions of 10 CFR Part 140, “Financial Protection Requirements and Indemnity Agreements,” have been satisfied;

G. The issuance of this 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 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; and

I. The receipt, possession, and use of byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material,” and 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.”

2. Accordingly, Facility License No. R-74 is hereby renewed in its entirety to read as follows:

   A. The license applies to the University of Wisconsin’s nuclear reactor with the TRIGA nuclear core and control system (herein “the facility”) owned by the University of Wisconsin, located on the licensee’s campus in Madison, Wisconsin, and described in the licensee's application for license renewal, as supplemented.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the University of Wisconsin as follows:

      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 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, the following activities are included:

         a. to receive, possess, and use, in connection with operation of the facility, up to 15.0 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 150 grams of contained uranium-235 of any enrichment in the form of neutron detectors;

c. to receive, possess, and use, in connection with operation of the facility, up to 16 grams of contained plutonium in the form of plutonium-beryllium neutron source;

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; and

e. to possess, but not use, up to 18.0 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.

3. Pursuant to the Act and 10 CFR Part 30, to receive, possess, and use, in connection with operation of the facility, such byproduct material as may be produced by operation of the reactor, which cannot be separated except for byproduct material produced in non-fueled experiments.

C. This license shall be deemed to contain and is subject to the conditions specified in 10 CFR Parts 20, 30, 50, 51, 55, 70, and 73 of the Commission's regulations; 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 1,000 kilowatts (thermal), and in pulse mode, with reactivity insertions not to exceed 1.4%Δk/k.

(2) Technical Specifications

The Technical Specifications contained in Appendix A 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 fully implement and maintain in effect all provisions of the physical security plan approved by the Commission and all amendments and changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p), respectively. The approved plan, which is exempt from public disclosure pursuant to the provisions of 10 CFR 73.21, is entitled, "University of Wisconsin Nuclear Reactor Security Plan," Revision 4, submitted by letter dated June 17, 1991.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A, Technical Specifications

Date of Issuance: March 25, 2011

APPENDIX A

TO

FACILITY LICENSE NO. R-74

DOCKET NO. 50-156

TECHNICAL SPECIFICATIONS AND BASES FOR

THE UNIVERSITY OF WISCONSIN NUCLEAR REACTOR

MARCH 2011

UWNR TECHNICAL SPECIFICATIONS

TS 1 INTRODUCTION

TS 1.1 Scope

This document constitutes the Technical Specifications for the University of Wisconsin Nuclear Reactor as required by 10 CFR 50.36. This document includes the basis to support the selection and significance of the specifications. Each basis is included for information purposes only, and is not part of the Technical Specifications in that it does not constitute requirements or limitations which the licensee must meet in order to meet the specifications.

These specifications are formatted to NUREG-1537 and ANSI/ANS 15.1-2007. Changes are denoted by redlining (indicated by vertical line in margin).

TS 1.2 Format

Content and section numbering is consistent with section 1.2.2 of ANSI/ANS 15.1-2007.

TS 1.3 Definitions

The terms used herein are explicitly defined to ensure uniform interpretation of the Technical Specifications.

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.

CHANNEL CHECK:
A channel check is a qualitative verification of acceptable performance by observation of channel behavior.

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

COLD CRITICAL:

The reactor is in the cold critical condition when it is critical in the reference core condition.

CONFINEMENT:

Confinement is an enclosure of the facility that is designed to limit the release of effluents between the enclosure and its external environment through controlled pathways. This is room 1215 of the Mechanical Engineering Building.

CONTROL ELEMENT:

A control element is any one or a combination of the following:

1. Shim-safety blade
2. Transient rod
3. Regulating blade

CORE LATTICE POSITION:

A core lattice position is that region in the core (approximately 3" by 3") over a grid hole. It may be occupied by a fuel bundle, an experiment or experimental facility, or a reflector element.

EXCESS REACTIVITY:

Excess reactivity is that amount of reactivity that would exist if all control elements were fully withdrawn from the core from the point where the reactor is exactly critical ($k_{eff} = 1$) in the reference core condition.

EXPERIMENT:

Experiment shall mean:

1. Any apparatus, device or material which is not a normal part of the reactor core or experimental facility, or

2. Any activity external to the biological shield using a beam of radiation emanating from the reactor core, or

3. Any operation designed to measure reactor parameters or characteristics.

Classification of experiments shall be:

1. Routine experiments. Routine experiments are those which have previously been approved and performed at the facility.

2. Modified routine experiments. Modified routine experiments are those which have not been performed previously but are similar to the routine experiments in that the hazards are neither greater nor significantly different than those for the corresponding routine experiments.

3. Special experiments. Special experiments are those which are not routine or modified routine experiments.

EXPERIMENT SAFETY SYSTEMS:

Experiment safety systems are those systems, including their associated input circuits, 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.

EXPERIMENTAL FACILITIES:

Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and any other in-pool irradiation facilities.

FUEL BUNDLE:

A fuel bundle is a cluster of three or four fuel elements secured in a square array by a top handle and a bottom grid plate adaptor.

FUEL ELEMENT:
A fuel element is a single TRIGA fuel rod of LEU 30/20 type.

INSTRUMENTED ELEMENT:
An instrumented element is a special fuel element in which thermocouples are embedded for the purpose of measuring fuel temperatures during reactor operation.

IRRADIATION:
Irradiation shall mean the insertion of any device or material that is not a normal part of the core or experimental facilities into an experimental facility so that the device or material is exposed to a significant amount of the radiation available in that irradiation facility.

LEU 30/20 CORE:
A LEU 30/20 core is an arrangement of TRIGA LEU 30/20 fuel in the reactor grid plate.

LIMITING SAFETY SYSTEM SETTINGS:
Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions.

MEASURED VALUE:
The measured value is the magnitude of that variable as it appears on the output of a measuring channel.

MEASURING CHANNEL:
A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device which are connected for the purpose of measuring the value of a variable.

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 reactor while the reactor is operating.

NON-SECURED EXPERIMENT

Any experiment not meeting the criteria of a secured experiment. A non-secured experiment may also be a movable experiment.

OPERABLE:

A system, device, or component shall be considered operable when it is capable of performing its intended functions in a normal manner.

OPERATING:

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

OPERATIONAL CORE:

An operational core is an LEU 30/20 core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable pulse reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

PROTECTIVE ACTION:

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

PULSE MODE (PU):

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

REACTIVITY WORTH OF AN EXPERIMENT:

The reactivity worth of an experiment is the value of the reactivity change that results from the experiment, being inserted into or removed from its intended position.

REACTOR OPERATION:

Reactor operation is any condition wherein the reactor is not secured.

REACTOR OPERATOR:

An individual who is licensed to manipulate the controls of a reactor.

REACTOR SAFETY SYSTEMS:

Reactor safety systems are those systems, including their associated input circuits, which are designed to initiate a reactor scram for the primary purpose of protecting the reactor or to provide information which requires manual protective action to be initiated.

REACTOR SECURED:

The reactor is secured when:

1. 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 upon optimum available conditions of moderation and reflection, or

2. The following conditions exist:

   a. All control elements are fully inserted, with the exception of the regulating blade in the event of an emergency,

   b. The reactor is shut down,

   c. The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area,

   d. No work is in progress involving core fuel, core structure, control elements, or control element drives unless the work on the drive can not move the control element, and

   e. No experiments are being moved or serviced that have, on movement, a positive reactivity worth exceeding 0.7% $\Delta k/k$.

REACTOR SHUTDOWN:

The reactor is shut down when the reactor is subcritical by at least 0.7% $\Delta k/k$ of reactivity.

REFERENCE CORE CONDITION:

The reactor is in the reference core condition when the fuel and bulk water temperatures are at ambient conditions and the reactor is xenon free.

REGULATING BLADE:
The regulating blade is a low worth control blade that need not have scram capability. Its position may be varied manually or by the servo-controller.

REPORTABLE OCCURRENCE:
A reportable occurrence is any of the following:

1. Operation with any safety system setting less conservative than specified in the technical specifications;

2. Operation in violation of a Limiting Condition for Operation listed in Section 3;

3. 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;

4. Any unanticipated or uncontrolled change in reactivity greater than 0.7% $\Delta k/k$, excluding reactor trips from a known cause;

5. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy causes or could have caused the existence or development of a condition which could result in a violation of technical specifications or applicable regulations;

6. Abnormal and significant degradation in reactor fuel or cladding, confinement, or coolant boundary (excluding minor leaks).

SAFETY CHANNEL:
A safety channel is a measuring channel in the reactor safety system.

SAFETY LIMITS:
Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

SCRAM TIME:
The time from the initiation of a scram signal to the time that the slowest scrammable control element reaches its fully inserted position.

SECURED EXPERIMENT:

A secured experiment shall mean any experiment that is held firmly in place by a mechanical device or by gravity, that is not readily removable from the reactor, and that requires one of the following actions to permit removal:

1. Removal of mechanical fasteners

2. Use of underwater handling tools

3. Moving of shield blocks or beam port containers.

SENIOR REACTOR OPERATOR:

An individual who is licensed to direct the activities of reactor operators. Such an individual is also a reactor operator.

SHALL, SHOULD, AND MAY:

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

SHIM-SAFETY BLADE:

A shim-safety blade is a control blade having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.

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 (assuming the most reactive scrammable control element and any non-scrammable control elements remain full out), and the reactor will remain subcritical without further operator action.

SQUARE WAVE MODE (SW):

Square wave mode operation shall mean any operation of the reactor with the mode selector switch in the square wave position.

STEADY STATE MODE (SS):

Steady state mode operation shall mean operation of the reactor with the mode selector switch in the manual or automatic positions.

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. Its position may be varied manually or by the servo-controller. It may have a voided or solid aluminum follower.

TS 2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

TS 2.1 Safety Limits

Applicability

This specification applies to fuel element temperature and steady-state reactor power level.

Objective

The objective is to define the maximum fuel element temperature and reactor power level that can be permitted with confidence that no fuel element cladding failure will result.

Specification

1. The temperature in a TRIGA LEU 30/20 fuel element shall not exceed 1150°C under any conditions of operation.

2. The reactor steady-state power level shall not exceed 1500 kW under any conditions of operation.

Basis

1. A loss of integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by air, fission product gases, and hydrogen from dissociation of 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 LEU 30/20 fuel element is based on data which indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided the temperature does not exceed 1150°C and the fuel cladding is water cooled[2].

2. It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500kW. The LEU Conversion SAR[1] section 4.7.8 shows by analysis that a power level of 1500 kW corresponds to a peak fuel temperature of 665°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

TS 2.2 Limiting Safety System Settings

Applicability

This specification applies to the scram setting which prevents the safety limit from being reached.

Objective

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

Specification

1. The limiting safety system setting for fuel temperature shall be 400°C as measured in an instrumented fuel element with a pin power peaking factor between 0.87 and 1.16, or 500°C as measured in an instrumented fuel element with a pin power peaking factor of at least 1.16.

2. The limiting safety system setting for reactor power level shall be 1.25 MW.

Basis

1. The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. Analyses performed in section 4.7.6 of the LEU Conversion Analysis[1] show that with the IFE in a core location with a pin power peaking factor of at least 0.87, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 400°C providing a margin of 472°C to the safety limit. The same analyses also show that with the IFE in a core location with a pin power peaking factor of at least 1.16, the maximum fuel temperature would be no greater than 678°C if the IFE thermocouple reaches 500°C providing a margin of 472°C to the safety limit.

   In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). In this mode, however, the temperature trip will act to reduce the amount of energy generated in the entire pulse transient by cutting off the "tail" of the energy transient in the event the pulse rod remains stuck in the fully withdrawn position.

2. Analysis in section 4.7 of the Conversion Analysis SAR[1] shows that at 1.3 MW, the peak fuel temperature in the core will be approximately 604°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

TS 3 LIMITING CONDITIONS FOR OPERATION

TS 3.1 Reactor Core Parameters

TS 3.1.1 Excess Reactivity

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control elements and applies for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times.

Specification

The excess reactivity shall not exceed 5.6% $\Delta k/k$.

Basis

As shown in chapter 4 of the SAR, this amount of excess reactivity will provide the capability to operate the reactor at full power with experiments in place. The primary limitation providing reactivity safety, however, is the shutdown margin requirement discussed in the next specification.

TS 3.1.2 Shutdown Margin

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control elements and applies for all modes of operation.

Objective

The objective is to assure that the reactor can be shut down at all times.

Specification

The reactor shall not be operated unless the shutdown margin provided by control elements shall be greater than 0.2% $\Delta k/k$ with:

1. the highest worth non-secured experiment in its most reactive state,

2. the highest worth control element and the regulating blade (if not scrammable) 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 highest worth control element should remain in the fully withdrawn position. If the regulating blade is not scrammable, its worth is not used in determining the shutdown reactivity.

TS 3.1.3 Pulse Limits

Applicability

This specification applies to the reactivity worth of the transient rod and pulse interlocks based on power level. It applies to pulse mode operation.

Objective

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

Specification

1. The reactivity to be inserted for pulse operation shall be determined and mechanically limited such that the reactivity insertion will not exceed 1.4% $\Delta k/k$.

2. Pulses shall not be initiated at power levels exceeding 1 kilowatt.

Basis

1. The LEU Conversion SAR[1] section 4.7.10 shows by analysis that a 1.4 %$\Delta k/k$ limitation on pulse reactivity will result in a maximum fuel temperature of 790°C. This leaves a margin to the 1150°C Safety Limit of 360°C, and a margin of 40°C to the 830°C operational limit recommended by General Atomics, "Pulsing Temperature Limit for TRIGA LEU Fuel," GA-C26017 (December, 2007).

2. The temperature rise from pulse initiation is in addition to the temperature in the fuel at the time the pulse is initiated. Limiting the initial power level to 1 kW assures that excessive temperatures will not be reached.

TS 3.1.4 Core Configurations

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 will not be produced.

Specification

1. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

2. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

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

4. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel assembly.

5. Control elements shall not be manually removed from the core unless the core has been shown to be subcritical with all control elements in the full out position.

Basis

1. TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and Texas A&M and their successful operational characteristics are available. In addition, the analysis performed at Wisconsin indicates that the LEU 30/20 core will safely satisfy all operational requirements. See chapters 4 and 13 of the LEU Conversion Analysis SAR[1].

2. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

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

4-5. Manual manipulation of core components will be allowed only when a single manipulation can not result in inadvertent criticality.

TS 3.1.5 Reactivity Coefficients

Does not apply to TRIGA reactors.

TS 3.1.6 Fuel Parameters

Applicability

This specification applies to the dimensional and structural integrity of the fuel elements.

Objective

The objective is to assure that the reactor will not be operated with defective fuel elements installed.

Specification

The reactor shall not be operated with damaged fuel except for purposes of identifying the damaged fuel. A fuel element shall be considered damaged and shall be removed from the core if:

1. In measuring the transverse bend, its sagitta[3] exceeds 0.125 inch over the length of the cladding;

2. In measuring the elongation, the length of the cladding exceeds its original length by 0.125 inch;

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

4. The fuel has not been visually inspected within the previous 15 months.

5. The burnup of uranium-235 in the UZrH fuel matrix exceeds 50 percent of the initial concentration.[4,5]

Basis

The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow through the top grid plate.

TS 3.2 Reactor Control and Safety Systems

TS 3.2.1 Operable Control Elements

Applicability

This specification applies to the number of operable control elements that must exist in order to operate the reactor.

Objective

The objective of this requirement is to insure that the reactor may be shut down from any condition of operation.

Specification

The reactor shall not be operated unless at least three control elements are operable and scrammable in accordance with TS 3.2.2.

Basis

The specification for shutdown margin assumes the regulating blade (non-scrammable control element) and the highest worth scrammable control element is fully withdrawn. Furthermore, analysis shows that with the regulating blade and any one scrammable control element in any position, the reactor core thermal-hydraulics are within the design basis.

TS 3.2.2 Reactivity Insertion Rates (Scram time)

Applicability

This specification applies to the time required for the scrammable control elements to be fully inserted from the instant that a safety channel variable reaches the Safety System Setting.

Objective

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

Specification

The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control element reaches its fully inserted position shall not exceed 2 seconds.

Basis

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

TS 3.2.3 Other Pulsed Operation Limitations

Limitations other than those on core configuration and pulsed reactivity insertion limits are not required on this reactor.

TS 3.2.4 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 channels that must be operable for safe operation.

Specification

The reactor shall not be operated unless the safety channels described in Table 3.2.4 are operable.

Table 3.2.4 Reactor Safety System Channels

	Safety Channel	Setpoint and Function	Number operable in specified mode		
			SS	SW	PU
1	Fuel Temperature	Scram if fuel temperature exceeds 400°C in the fuel temperature safety channel for an instrumented fuel element pin power peaking factor of 0.87-1.16, or 500°C for an instrumented fuel element pin power peaking factor greater than 1.16.	1	1	1
2	Linear Power Level	Scram if power > 125% full power	2	2	-
3	Manual Scram	Manually initiated scram	1	1	1
4	Preset Timer	Transient rod scram 15 seconds or less after pulse	-	-	1
5	Reactor water level	Scram if < 19 feet above top of core	1	1	1
6	High Voltage Monitor	Scram on loss of high voltage to neutron and gamma ray power level instrument detectors	1	1	1
7	Reactor water temperature	Scram if > 130°F	1	1	1

Basis

1-2. The fuel temperature and power scrams provide protection to ensure that the reactor is shut down before the safety limit is reached.

3. The manual scram allows the operator a means of rapid shutdown in the event of unsafe or abnormal conditions.

4. The preset timer assures reduction of reactor power to a low level after a pulse.

5. The reactor pool water level scram assures shutdown of the reactor in the event of a serious leak in the primary system or pool.

6. The high voltage monitor prevents operation of the reactor with other systems inoperable due to failure of the detector high voltage supplies.

7. The reactor pool water temperature scram prevents operation of the reactor in an un-analyzed condition.

TS 3.2.5 Interlocks

Applicability

This section applies to the interlocks which inhibit or prevent control element withdrawal or reactor startup.

Objective

The objective of these interlocks is to prevent operation under unanalyzed or imprudent conditions.

Specification

The reactor shall not be operated in the indicated modes unless the interlocks in Table 3.2.5 are operable.

Table 3.2.5 Interlocks

	Channel	Setpoint and Function	Number operable in specified mode		
			SS	SW	PU
1	Log Count Rate	Prevent control element withdrawal when neutron count rate < 2 per second	1	1	1
2	Transient Rod Control	Prevent application of air to fire transient rod unless drive is at IN limit.	1	0	0
3	Log N Power Level	Prevent application of air to fire transient rod when power level is above 1 kW and transient rod is not full in.	1	1	1
4	Pulse Mode Control	Prevents withdrawal of control blades while in pulse mode.	0	0	1

Basis

1. The Log count rate interlock does not allow control element withdrawal unless the neutron count rate is high enough to assure proper instrument response during reactor startup.

2. The Transient Rod Control interlock prevents inadvertent addition of excessive amounts or reactivity in steady-state modes.

3. The Log N interlock prevents firing of the transient rod at power levels above 1.0 kW if the transient rod drive is not in the full down position. This effectively prevents inadvertent pulses which might cause fuel temperature to exceed the safety limit on fuel temperature.

4. The pulse mode control blade withdrawal interlock prevents reactivity addition in pulse mode other than by firing the transient rod.

TS 3.2.6 Backup Shutdown Mechanisms

Backup shutdown mechanisms are not required for this reactor.

TS 3.2.7 Bypassing Channels

Applicability

This specification applies to the interlocks in Table 3.2.5.

Objective

The objective is to indicate the conditions in which an interlock may be bypassed.

Specification

The Log Count Rate interlock in Table 3.2.5 may be bypassed:

1. During fuel loading in order to allow control element withdrawal necessary for the fuel loading procedure or

2. When Log Power Level and Linear Power Level channels are on-scale.

Basis

1. During early stages of fuel loading the count-rate on the source range channel will be below the interlock setpoint. The bypass allows control element movements necessary for loading fuel with control elements partially withdrawn and for performing inverse multiplication determinations of control element worth and core reactivity status.

2. Once the other power indications are available the startup count rate channel is no longer required, so the interlock no longer serves any purpose.

TS 3.2.8 Control Systems and Instrumentation Required for Operation

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless measuring channels listed in Table 3.2.8 are operable.

Table 3.2.8 Instrumentation and Controls Required for Operation

	Channel	Function	Number operable in specified mode		
			SS	SW	PU
1	Fuel Temperature	Input for fuel temperature scram.	1	1	1
2	Linear Power Level	Input for safety system power level scram	2	2	0
3	Log Power Level	Wide range power indication, permissive for initiation of Pulse Mode	1	1	0
4	Startup Log Count Rate	Wide range power indication, permissive for control element withdrawal	1*	1*	0
5	Pulsing Power Level	Pulse power level indication	0	0	1

* Required during startup only until the Log Power Level and Linear Power Level channels are on-scale. See TS 3.2.7.

Basis

1. Fuel temperature indicated at the control console gives continuous information on the process variable which has a specified safety limit.

2-5. The power level monitors assure that reactor power level is adequately monitored for all modes of operation.

TS 3.3 Reactor Pool Water Systems

Applicability

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

Objective

The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding and to prevent damage to in-pool components by corrosion.

Specification

1. A pool level alarm shall indicate loss of coolant if the pool level drops one foot or more below normal level.

2. A pool water temperature alarm shall indicate if water temperature reaches 130°F.

3. The reactor shall not be operated if the conductivity of the pool water exceeds 5 µS/cm (<0.2 MΩ-cm) when averaged over a period of one week.

4. The reactor shall not be operated if the radioactivity of pool water exceeds the limits of 10 CFR Part 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours.

5. The reactor shall not be operated if the pH of the pool water is greater than 7.5 or less than 5.5.

Basis

1. Loss of coolant alarm, after one foot of loss, requires corrective action. This alarm is observed in the reactor control room and outside the reactor building.

2. The thermal-hydraulic analysis in the SAR assumes a pool water temperature of 130°F. If the temperature exceeds 130°F then the alarm will prevent continued operation in an un-analyzed condition.

3. The conductivity limit assures that materials within the pool will not be degraded and that the radioactivity of the pool water will be minimized.

4. Analyses in section 12.2.9 of the Safety Analysis Report show that limiting the activity to this level will not result in any person being exposed to concentrations greater than those permitted by 10 CFR Part 20.

5. The pH limit assures that materials within the pool will not be degraded.

TS 3.4 Confinement

Applicability

These specifications apply to the room housing the reactor and the ventilation system controlling that room.

Objective

The objective is to provide restrictions on release of airborne radioactive materials to the environs.

TS 3.4.1 Operations That Require Confinement

Specification

Confinement is required for reactor operation or any movement of irradiated fuel or fueled experiments.

Basis

During reactor operation or movement of irradiated fuel there is the potential for a release of radioactivity from the fuel clad. Confinement will limit the consequences to the public from such a release.

TS 3.4.2 Equipment to Achieve Confinement

Specification

To achieve confinement, the ventilation system shall be operating in accordance with TS 3.5.

Basis

With the ventilation system operating any potential fission product release will be swept out of the lab and exhausted from a monitored and elevated release point to limit the consequences to the public from such a release.

TS 3.5 Ventilation Systems

Applicability

This specification applies to the operation of the reactor laboratory ventilation system.

Objective

The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation.

Specification

The reactor shall not be operated unless the ventilation system is operating. The ventilation system is considered operating when the following conditions exist:

1. One stack exhaust fan is operating,

2. Exhaust flow-rate is at least 9600 scfm, and

3. Exhaust filter total pressure drop is less than 2.5 inches of water column.

Basis

It is shown in the SAR Chapter 11 that Argon-41 release at zero stack height results in concentrations less than the concentrations permitted for non-restricted areas. However, the calculations indicate that operation of the ventilation system significantly reduces the concentration to which the public would be exposed. Exposures in the event of a fuel element cladding leak are also calculated based on non-operation of the ventilation system, but are significantly reduced with the ventilation system running. Therefore, operation of the reactor with the ventilation system running will minimize exposure to the public from routine operation and hypothetical accidents.

TS 3.6 Emergency Power

Emergency power systems are not required for this facility.

TS 3.7 Radiation Monitoring Systems and Effluents

TS 3.7.1 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.

Specification

The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.7.1 are operable.

Table 3.7.1 Radiation Monitoring Systems

	Radiation Monitoring Channels*	Function	Number
1	Area Radiation Monitor	Monitor radiation levels within the reactor room	3
2	Exhaust Gas Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
3	Exhaust Particulate Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
4	Environmental Radiation Monitors	TLD dosimeters in area surrounding the stack	4

* For periods of time, not to exceed 1 week, for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

Basis

1. The area radiation monitor alarms at the reactor control console to preclude the possibility of unknowingly generating high radiation levels by operating the reactor. Upon alarm, the reactor operator will initiate corrective actions to limit exposure to members of the public to less than 10 CFR Part 20 limits. Furthermore, chapter 13 of the SAR shows that the area radiation monitor will initiate a building evacuation in the event of an accident at the facility thereby limiting exposure to members of the public to less than 10 CFR Part 20 limits.

2. The exhaust gas radiation monitor alarms at the reactor control console to preclude the possibility of discharging gas effluent from the facility, as diluted by atmospheric air in the lee of the facility as a result of the turbulent wake effect, in excess of 10 CFR Part 20 Appendix B, Table II limits when averaged over one year. Upon alarm, the reactor operator will initiate corrective actions to limit exposure to members of the public to less than 10 CFR Part 20 limits.

3. The exhaust particulate radiation monitor samples downstream of the main exhaust HEPA filters and alarms at the reactor control console to preclude the possibility of unknowingly discharging particulate effluent from the facility. Upon alarm, the reactor operator will initiate corrective actions to limit exposure to members of the public to less than 10 CFR Part 20 limits.

4. The environment monitors are placed in areas immediately surrounding the reactor laboratory to record integrated dose that would have been delivered to a person continually present in the area.

TS 3.7.2 Effluent (Argon-41) Discharge Limit

Applicability

This specification applies to the concentration of Ar-41 which may be discharged from the facility.

Objective

The objective is to assure that the health and safety of the public are not endangered by the discharge of Ar-41.

Specification

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

Basis

10 CFR Part 20 Appendix B, Table II specifies a limit of 1×10^{-8} μCi/ml for Ar-41. Chapter 13 of the LEU Conversion SAR calculates that the maximum ground-level concentration from operation of the ventilation system is 3.6×10^{-5} μCi/ml per Ci/sec discharged. A ground-level concentration of 1×10^{-8} μCi/ml would result from a discharge rate of 278 μCi/sec; the resulting stack exhaust concentration would be 6.14×10^{-5} μCi/ml. Chapter 11 of the SAR calculates that the maximum hypothetical Ar-41 release rate is only 13.3 μCi/s.

TS 3.8 Experiments

Applicability

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

Objective

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

TS 3.8.1 Reactivity Limits

Specification

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

1. The sum of the absolute values of the reactivity worths of all non-secured experiments does not exceed 0.7 %Δk/k.

2. The reactivity worth of any single secured experiment does not exceed 1.4 %Δk/k.

3. The sum of the absolute values of the reactivity worths of all experiments, both secured and non-secured, does not exceed the maximum excess reactivity specified in TS 3.1.1.

Basis

1. This specification is intended to provide assurance that the worth of non-secured experiments will be limited to a value such that the safety limit will not be exceeded if the positive worth of all experiments were to be suddenly inserted (SAR Chapter 13).

2. The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. Since experiments of such worth must be fastened in place, its removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained. SAR accident analysis includes a sudden addition of 1.4 %Δk/k from firing the transient control rod while operating at the power level scram point, a more severe transient than that which could result from removal of a fixed experiment with the same reactivity worth.

3. This specification provides assurance that by removing all installed experiments the maximum excess reactivity specified in TS 3.1.1 would not be exceeded.

TS 3.8.2 Materials

Specification

1. 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 half the design pressure of the container.

2. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10 CFR Part 20.

3. In calculations pursuant to 2 above, the following assumptions shall be used:

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

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

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

   d. An atmospheric dilution factor of 3.6×10^{-5} µCi/ml per Ci/s for gaseous discharges from the facility.

4. Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies.

5. Experiment materials that are corrosive to reactor components or highly reactive with coolants shall be doubly encapsulated.

<u>Basis</u>

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

2-3. These specifications are intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary of the UWNR. The dilution factor is based on computations reported in Chapter 11 and Appendix A of the Safety Analysis Report.

4. The 1.5 curie limitation on iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR Part 20 for an unrestricted area.

5. This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving corrosive or highly reactive materials.

TS 3.8.3 Experiment Failure and Malfunctions

Specification

If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection of the capsule shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Director or designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Basis

Operation of the reactor with a failed capsule is prohibited to prevent damage to the reactor fuel or structure. Failure of a capsule must be investigated to assure no damage has or will occur.

TS 3.9 Facility Specific LCOs

There are no facility specific LCOs at this facility.

TS 4 SURVEILLANCE REQUIREMENTS

General Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety after maintenance or modification of the system.

Specification

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 system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safety Committee. A system shall not be considered operable until after it is successfully tested.

The following terms for average surveillance intervals shall allow, for operational flexibility only, maximum times between surveillance intervals as indicated below unless otherwise specified within the specification.

- Five-year interval not to exceed six years.
- Biennial interval not to exceed two and one-half years.
- Annual interval not to exceed 15 months.
- Semiannual interval not to exceed seven and one-half months.
- Quarterly interval not to exceed four months.
- Monthly interval not to exceed six weeks.
- Weekly interval not to exceed ten days
- Daily interval shall be done within the calendar day.

Scheduled surveillances, except those specifically required when the reactor is shut down, may be deferred during shutdown periods, but be completed prior to subsequent reactor startup unless operation is required for the performance of the surveillance. Scheduled surveillances which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown. If the reactor is not operational in a particular mode, surveillances required specifically for that mode may be deferred until the reactor becomes operational in that mode.

Basis

This specification relates to changes in reactor systems which could directly affect the safety of the reactor. 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.

TS 4.1 Reactor Core Parameters

Applicability

This specification applies to the surveillance requirements for measurements, tests, and calibrations of reactor core parameters.

Objective

The objective is to verify the core parameters which are directly related to reactor safety.

Specification

1. Excess reactivity
   Excess reactivity shall be determined at least annually and after changes in either the core, in-core experiments, or control elements for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

2. Shutdown margin
   The shutdown margin shall be determined at least annually and after changes in either the core, in-core experiments, or control elements.

3. Pulse limits
   The reactor shall be pulsed semiannually to compare fuel temperature measurements and peak power levels with those of previous pulses of the same reactivity value.

4. Core configuration
   Each planned change in core configuration shall be determined to meet the requirements of Sections 3.1(4) and 5.3 of these specifications before the core is loaded.

5. Fuel Parameters

   a. All fuel elements shall be inspected visually for damage or deterioration annually.

   b. Uninstrumented fuel elements which have been resident in the core during the previous year shall be measured for length and sagitta annually. Fuel elements shall not be added to a core unless a measurement of length and sagitta has been completed within the previous fifteen months.

   c. Fuel elements in the hottest assumed location, as well as representative elements in each of the rows, shall be measured for possible damage in the event there is indication that the Limiting Safety System Setting may have been exceeded.

Basis

1-2. Annual measurements, coupled with measurements made after changes that can affect reactivity values provide adequate assurance that core behavior will be as analyzed. The reactivity values in TRIGA LEU 30/20 fuel change very slowly with fuel burnup.

3. Semiannual verifications assure no changes in behavior are resulting from fuel characteristic changes.

4. Checking contemplated core configurations against requirements will prevent inadvertent loading of cores which do not meet power peaking restraints imposed by composition restrictions.

5. Annual inspection of the TRIGA fuel has been shown adequate to assure fuel element integrity through a long history of standard operation.

TS 4.2 Reactor Control and Safety Systems

Applicability

This specification applies to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective

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

Specification

1. Reactivity worth of control elements
   The reactivity worth of control elements shall be determined annually and following significant changes in core composition or arrangement that increase reactivity by a value greater than or equal to the shutdown margin.

2. Transient Rod and Associated Mechanism
   The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary annually.

3. Scram times of control and safety elements
   The scram time for all scrammable control elements shall be measured annually and following maintenance to the control elements or their drives.

4. Scram and Power Measuring Channels

   a. A channel test of each Reactor Safety System measuring channel in Table 3.2.4 items (1) through (4) and the interlocks in Table 3.2.5 required for the intended modes of operation shall be performed daily before each day's operation or prior to each operation extending more than one day.

   b. A channel test of items (5), (6), and (7) in Table 3.2.4 shall be performed semi-annually.

   c. A channel calibration of items (1) and (2) in Table 3.2.4 shall be performed annually.

5. Operability Tests
   This concern is covered by the General Surveillance criterion at the beginning of this section.

6. Thermal Power Calibration-Natural Convection
   A Channel Calibration shall be made of the power level monitoring channels by the calorimetric method upon changes in core composition or arrangement that could change flux distributions in the region of the nuclear instrumentation and annually thereafter.

7. Control Element Inspection
   The control elements shall be visually inspected for deterioration biennially.

Basis

1. Control element worths change slowly unless the core arrangement is changed, so annual measurement is sufficient to assure safety.

2. Transient rod drive and air supply includes filtration and lubrication, so an annual check coupled with pre-startup checks is sufficient to assure operability.

3. Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control elements to perform properly.

4. The items 1 through 4 in the table are essential safety equipment and thus should be checked frequently, even though no failures have been observed by checkout in nearly 50 years of operation. Frequent testing is unnecessary for item 5, a simple float switch which is very unlikely to fail, and has performed for nearly 50 years without a failure. Testing item 6, the high voltage monitor scram, results in changing the voltage to the neutron detectors. This introduces step changes into the signal circuits of the measuring channels which can lead to long recovery times and a significant increase in failures of the measuring channels. Further, since the checkout of the linear safety channels is a source check, if high voltage were lost that check would not be possible if the voltage had been lost.

5. The general requirement for checks of equipment operability after maintenance or modification of systems will reveal any loss of safety functions due to the maintenance or modification.

6. The power level channel calibration will assure that the reactor will be operated at the proper power levels.

7. Annual checks in other TRIGA reactors and for nearly 50 years in this reactor have been sufficient to insure no failures due to deterioration.

TS 4.3 Coolant Systems

Applicability

This specification applies to the reactor pool water.

Objective

The objective is to assure the water quality and radioactivity is within the defined limits

Specification

1. The pool water conductivity and radioactivity shall be measured quarterly. This specification shall not be deferred during extended periods without operation.

2. Pool water pH shall be measured quarterly during periods of time when the conductivity is greater than 0.1 μS/cm (resistivity <10MΩ-cm). This specification shall not be deferred during extended periods without operation.

Basis

1. Pool water conductivity is continuously monitored, but would be manually monitored on a quarterly basis if the instruments failed. Radioactivity is indirectly monitored by an area radiation monitor near the demineralizer bed, so gross activity increases would be detected immediately. Experience with TRIGA reactors indicates the earliest detection of fuel clad leaks is usually from airborne activity, rather than pool water activity. The quarterly measurement can identify specific radionuclides.

2. Analysis has shown that as long as pool water conductivity is less than 0.1 μS/cm (resistivity >10MΩ-cm), pool water pH is between 7.5 and 6.5. During periods of time when pool water conductivity is greater than 0.1 μS/cm (resistivity <10MΩ-cm), pool water pH must be measured to ensure compliance with TS 3.3.

TS 4.4 Confinement

Applicability

This specification applies to the reactor confinement.

Objective

The objective is to assure that air is swept out of confinement and exhausted through a monitored release point.

Specification

The ventilation system shall be verified operable in accordance with TS 4.5 quarterly.

Basis

Because the ventilation system is the only equipment required to achieve confinement, operability checks of the ventilation system meet the functional testing requirements for confinement.

TS 4.5 Ventilation Systems

Applicability

This specification applies to the building confinement ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the uncontrolled environment.

Specification

It shall be verified quarterly and following repair or maintenance that the ventilation system is operable in accordance with TS 3.5.

Basis

Over 30 years of experience with the previous ventilation system has demonstrated that testing the system quarterly is sufficient to assure the proper operation of the system and control of the release of radioactive material. The new ventilation system is expected to exceed the reliability of the previous system so quarterly testing is still appropriate.

TS 4.6 Emergency Electrical Power Systems

Not Applicable.

TS 4.7 Radiation Monitoring Systems and Effluents

TS 4.7.1 Radiation Monitoring Systems

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the stack air monitoring system.

Objective

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

Specification

1. The radiation monitoring and stack monitoring systems shall be calibrated annually and shall be verified to be operable by monthly source checks or channel tests.

2. The environmental dosimeters shall be evaluated on a quarterly basis.

Basis

1. Experience has shown that monthly verification of area radiation monitor operability and setpoints in conjunction with the downscale-failure feature of the instrument is adequate to assure operability. Annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. Annual calibrations and monthly source or channel checks of the stack particulate and gaseous monitors, along with the high or low flow alarms associated with the monitor assure operability and accuracy.

2. Quarterly evaluation of environmental dosimeters is adequate to assure the annual limit on public exposure is not reached.

TS 4.7.2 Effluents

Applicability

This specification applies to gaseous and liquid discharges from the reactor laboratory.

Objective

The objective is to assure that ALARA and 10 CFR Part 20 limits are observed.

Specification

1. Liquid radioactive waste discharged to the sewer system shall be sampled for radioactivity to assure levels are below applicable limits before discharge. Results of the measurements shall be recorded and reported in the Annual Report.

2. The total annual release of gaseous radioactivity to the environment shall be reviewed quarterly.

Basis

1. Liquid waste releases are batch releases, so the liquid can be sampled before release.

2. Air activity discharged is continuously recorded and the integrated release is reported.

TS 4.8 Experiments

Applicability

This specification applies to experiments to be installed in the reactor and its experimental facilities.

Objective

The objective is to assure that all experiments to be installed in the reactor and its experimental facilities are evaluated.

Specification

Prior to inserting an experiment in the reactor and its experimental facilities, the criteria of TS 3.8 shall be evaluated.

Basis

Evaluating an experiment prior to inserting in the reactor and its experimental facilities will provide assurance that no damage to the reactor fuel or structure will occur.

TS 4.9 Facility-Specific Surveillance

Not applicable. There is no facility-specific surveillance.

TS 5 DESIGN FEATURES

TS 5.1 Site and Facility Description

Applicability

This specification applies to the room housing the reactor and the ventilation system controlling that room.

Objective

The objective is to provide restrictions on release of airborne radioactive materials to the environs.

Specification

1. The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume shall be 2,000 cubic meters.

2. All air or other gas exhausted from the reactor room and the Beam Port and Thermal Column Ventilation System shall be released to the environment a minimum of 30.5 meters above ground level.

3. The operations boundary shall be the Reactor Laboratory, room 1215 of the Mechanical Engineering Building. The operations boundary shall be a restricted area.

4. The site boundary shall be that portion of the center and east wings of the Mechanical Engineering Building south of the north lobby, plus the portion of Engineering Drive south of the designated areas of the building. The site boundary may be a non-restricted area.

Basis

1. Calculations in Chapter 13 of the SAR demonstrate that the occupational doses in the event of the maximum hypothetical accident do not exceed limits if the lab volume is at least 2000 cubic meters.

2. Calculations in Chapter 13 that assume operation of the ventilation system assume a stack height of 30.5m.

3. The Reactor Director has direct authority over all activities within room 1215 of the Mechanical Engineering Building.

4. The Reactor Director may directly initiate emergency activities within the site boundary. The site boundary may be frequented by people unacquainted with reactor operations.

TS 5.2 Reactor Coolant System

Applicability

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

Objective

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

Specification

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

2. The pool water inlet pipe to the demineralizer shall not extend more than 15 feet into the top of the reactor pool when fuel is in the core. The outlet pipe from the demineralizer shall be equipped with a check valve and siphon breaker to prevent inadvertent draining of the pool.

3. Diffuser and other auxiliary systems pumps shall be located no more than 15 feet below the top of the reactor pool.

4. All other piping and pneumatic tube systems entering the pool shall have siphon breakers and valves or blind flanges which will prevent draining more than 15 feet of water from the pool.

5. A pool level alarm shall indicate loss of coolant if the pool level drops approximately one foot below normal level.

6. A pool water temperature alarm shall indicate if water temperature reaches 130°F.

Basis

1. The LEU Conversion SAR Section 4.7.8 shows by analysis that the natural convective cooling of the reactor core is sufficient to maintain the fuel in a safe condition up to at least a power level of 1500 kW (the power level Safety Limit).

2. The inlet pipe to the demineralizer is positioned so that a siphon action will drain less than 15 feet of water. The outlet pipe from the demineralizer discharges into a pipe entering the bottom of the pool through a check valve which prevents leakage from the pool by reverse flow from pipe ruptures or improper operation of the demineralizer valve manifold. In addition, the pipe has a loop equipped with a siphon breaker which prevents loss of pool water.

3. In the event of pipe failure and siphoning of pool water, the pool water level will drop no more than 15 feet from the top of the pool.

4. Other pipes which enter the pool have siphon breakers which prevent pool drainage. Valves are provided for pneumatic tube system lines and primary cooling system pipes. Other piping installed in the pool has blind flanges permanently installed.

5. Loss of coolant alarm, after one foot of loss, requires corrective action. This alarm is observed in the reactor control room and outside the reactor building.

6. The thermal-hydraulic analysis in the SAR assumes a pool water temperature of 130°F. If the temperature exceeds 130°F then the alarm will prevent continued operation in an un-analyzed condition.

TS 5.3 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.

Specification

The individual unirradiated TRIGA LEU 30/20 fuel elements shall have the following characteristics:

1. Uranium content: maximum of 30 Wt-% enriched to maximum of 19.95 Wt-% with nominal enrichment of 19.75 Wt-% Uranium 235.

2. Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms with a maximum H to Zr ratio of 1.65.

3. Natural erbium content (homogeneously distributed): nominal 0.9 Wt-%.

4. Cladding: 304 stainless steel, nominal 0.020 inch thick.

Basis

1. The fuel specification permits a maximum uranium enrichment of 19.95%. This is about 1% greater than the design value for 19.75% enrichment. Such an increase in loading would result in an increase in power density of less than 1%. An increase in local power density of 1% reduces the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

2. The maximum hydrogen-to-zirconium ratio of 1.65 could result in a maximum stress under accident conditions in the fuel element clad 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 (M.T. Simnad, "The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel," General Atomics Report E-117-833, February, 1980).

3. The fuel specification for a single fuel element permits a minimum erbium content of about 5.6% less than the design value of 0.90 Wt-%. (However, the quantity of erbium in the full core must not deviate from the design value by more than -3.3%). This variation for a single fuel element would result in an increase in fuel element power density of about 1-2%. Such a small increase in local power density would reduce the safety margin by less than 2% (Texas A&M LEU Conversion SAR, December 2005).

4. Stainless steel clad has been shown through decades of operation to provide a sufficient barrier against fission product release with minimal corrosion.

TS 5.4 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 will not be produced.

Specification

1. The core shall be an arrangement of TRIGA LEU 30/20 uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

2. The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

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

Basis

1. TRIGA cores have been in use for years and their characteristics are well documented. LEU cores including 30/20 fuel have also been operated at General Atomics and Texas A&M and their successful operational characteristics are available. In addition, the analysis performed at Wisconsin indicates that the LEU 30/20 core will safely satisfy all operational requirements. See chapters 4 and 13 of the LEU Conversion Analysis SAR.

2. Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

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

TS 5.5 Control Elements

Applicability

This specification applies to the control blades and transient control rod.

Objective

The objective is to assure that control elements are fabricated to reliably perform their intended control and safety function.

Specification

1. The safety blades shall be constructed of boral plate and shall have scram capability.

2. The regulating blade shall be constructed of stainless steel.

3. The transient rod shall contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient control rod shall have scram capability and may incorporate an aluminum or air follower.

Basis

1-2. The boral safety blades and stainless steel regulating blade used in the reactor have been shown to provide adequate reactivity worth, structural rigidity, and reliability to assure reliable operation and long life under operating conditions.

3. The transient control rod materials and fabrication techniques have been used in many TRIGA reactors and have demonstrated reliable operation and long life.

TS 5.6 Fissionable Material 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 will not become critical and will not reach an unsafe temperature.

Specification

1. All fuel elements and fueled devices shall be stored in a geometrical array where the value of k-effective is less than 0.8 for all conditions of moderation and reflection.

2. Irradiated fuel elements and fueled 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 or safety limits.

Basis

1-2. The limits imposed by specifications 5.6.1 and 5.6.2 are conservative and assure safe storage.

TS 6. ADMINISTRATIVE CONTROLS

TS 6.1 Organization

TS 6.1.1 Structure

The reactor facility shall be an integral part of the Engineering Physics Department of the College of Engineering of the University of Wisconsin-Madison. The reactor shall be related to the University structure as shown in **Figure TS-1**.

The Radiation Safety office performs audit functions for both the Radiation Safety Committee and the Reactor Safety Committee and reports to both committees as well as to the Reactor Director.

TS 6.1.2 Responsibility

The Reactor Director is responsible for all activities at the facility, including licensing, security, emergency preparedness, and maintaining radiation exposures as low as reasonably achievable.

The reactor facility shall be under the direct control of a Reactor Supervisor designated by the Reactor Director. The Reactor Supervisor shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures, and the requirements of the Radiation Safety Committee and the Reactor Safety Committee.

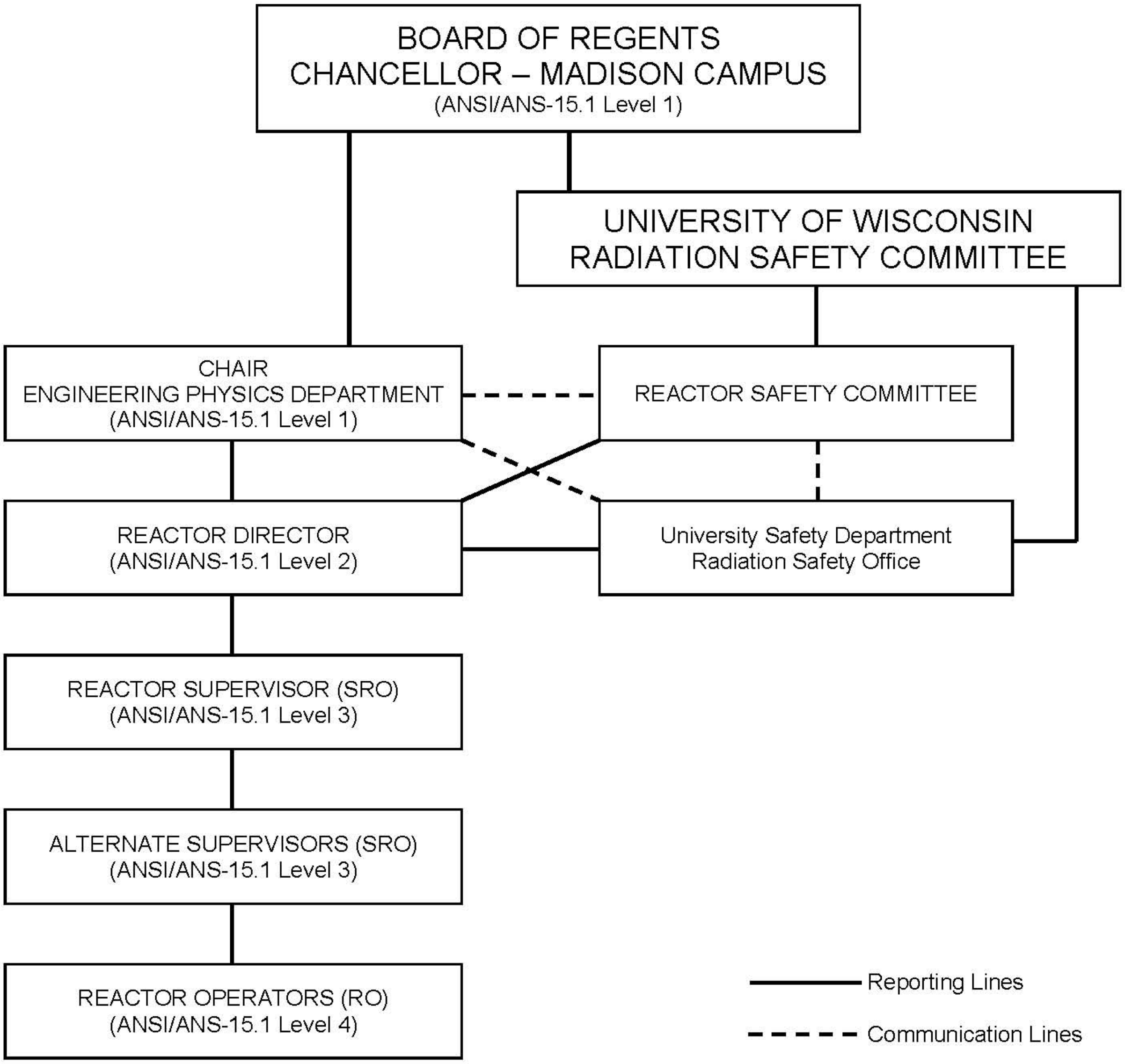

Figure TS-1, Organization Chart

TS 6.1.3 Staffing

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

   a. A licensed reactor operator in the control room (if licensed senior reactor operator, may also be the person required in c).

   b. A second designated person present at the facility or readily available by phone or radio and within 1000 feet capable of carrying out prescribed written instructions.

   c. A designated senior reactor operator shall be readily available at the facility or on call. On call means the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility.

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

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

   a. Initial startup and approach to power.

   b. All fuel handling or control-element relocations.

   c. Relocation of any in-core experiment with a reactivity worth greater than 0.7% $\Delta k/k$.

   d. Recovery from unplanned or unscheduled shutdown or significant power reduction.

TS 6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of ANSI/ANS-15.4-2007 Sections 4-6.

TS 6.2 Review and Audit

There shall be a Reactor Safety Committee which shall review and audit reactor operations to assure that the facility is operated in a manner consistent with public safety and within the conditions of the facility license.

TS 6.2.1 Composition and Qualifications

The Committee shall be composed of a least six members, one of whom shall be a Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office. The Committee shall collectively possess expertise in the following disciplines:

1. Reactor Physics;
2. Heat transfer and fluid mechanics;
3. Metallurgy
4. Instruments and Control Systems;
5. Chemistry and Radio-chemistry;
6. Radiation Safety.

Reactor staff shall not be members of the committee. This does not preclude reactor staff from participating on subcommittees.

TS 6.2.2 Charter and Rules

The Committee shall meet at least annually.

The Committee shall formulate written standards regarding the activities of the full committee; minutes, quorum, telephone polls for approvals not requiring a formal meeting, and subcommittees.

A quorum shall be at least half of the members.

TS 6.2.3 Review Function

The responsibilities of the Reactor Safety Committee shall include, but are not limited to, the following:

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

2. Review and approval of all proposed changes to the facility, procedures, license, and technical specifications;

3. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures are allowed in accordance with 10 CFR 50.59 without prior authorization by the NRC;

4. Review of abnormal performance of plant equipment and operating anomalies having safety significance; and

5. Review of unusual or reportable occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50.

6. Review of audit reports.

7. Review of violations of technical specifications, license, or procedures and orders having safety significance.

TS 6.2.4 Audit Function

A Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office shall represent the University Radiation Safety Committee and shall conduct an inspection of the facility at least once every calendar month to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the Health Physics Office shall also be available to the Reactor Safety Committee, and will extend the scope of the audit to cover license, technical specification, and procedure adherence.

The committee shall annually audit operation and operational records of the facility, correction of deficiencies, requalification program, security plan, and emergency plan and their implementing procedures. If the committee chooses to use the staff of the Health Physics organization for the audit function, the reports of audit results will be distributed to the committee and included as an agenda item for committee meetings.

TS 6.3 Radiation Safety

The Reactor Laboratory shall meet the requirements of the University Radiation Safety Regulations as submitted for the University Broad License, License Number 25-1323-01 and is subject to the authority of the state license.

The Reactor Director shall have responsibility for maintaining radiation exposures as low as reasonably achievable and for implementation of laboratory procedure for insuring compliance with 10 CFR Part 20 regulations.

TS 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 judgment and action should the situation require such. Operating procedures shall be in effect for the following items:

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

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

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

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

5. Personnel radiation protection, consistent with the University of Wisconsin Radiation Protection Program and applicable regulations;

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

7. Implementation of required plans such as emergency or security plans;

8. Use, receipt, and transfer of by-product material held under the reactor license.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Safety Committee. Temporary changes to the procedures that do not change their original intent may be made by the Senior Operator in control or designated alternate. All such temporary changes shall be documented and subsequently reviewed by the Reactor Safety Committee.

TS 6.5 Experiment Review and Approval

1. Routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval.

2. Prior to performing any experiment which is not a routine experiment, the proposed experiment shall be evaluated by the senior operator responsible for operation. The senior operator shall consider the experiment in terms of its effect on reactor operation and the possibility and consequences of its failure, including where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, reactivity effects, and interactions with reactor instrumentation. The experiment shall only be performed if the evaluation concludes that the requirements of 10 CFR 50.59(c) are met.

3. Modified routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval by the Reactor Safety Committee provided that the evaluation performed in accordance with Section 6.5(2) results in a determination that the hazards associated with the modified routine experiment are neither greater nor different than those involved with the corresponding routine experiment which shall be referenced.

4. No special experiment shall be performed until the proposed experiment has been reviewed and approved by the Reactor Safety Committee.

5. Favorable evaluation of an experiment shall conclude that failure of the experiment will not lead directly to damage of reactor fuel or interference with movement of a control element.

TS 6.6 Required Actions

TS 6.6.1 Action to be Taken in Case of Safety Limit Violation

In the event a safety limit is exceeded:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.
2. An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Committee, and Reactor Director, and reports shall be made to the NRC in accordance with Section 6.7 of these specifications, and
3. A report shall be prepared which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Committee (RSC) for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

TS 6.6.2 Action to be Taken in the Event of an Occurrence of the Type Identified in 6.7.2(1)b., and 6.7.2(1)c.

In the event of a reportable occurrence (see TS 1.3) the following actions shall be taken:

1. The reactor shall be shut down until operation is authorized by the Reactor Director.
2. The Director or designated alternate shall be notified and corrective action taken with respect to the operations involved,
3. The Director or designated alternate shall notify the Chairman of the Reactor Safety Committee,
4. A report shall be made to the Reactor Safety Committee which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, and
5. A report shall be made to the NRC in accordance with Section 6.7.2 of these specifications.

TS 6.7 Reports

TS 6.7.1 Operating Reports

1. An annual report covering the activities of the reactor facility during the previous fiscal year shall be submitted (in writing to U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555) within six months following the end of each fiscal year, providing the following information:

    a. A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

    b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

    c. The number of emergency shutdowns and inadvertent scrams, including reasons therefore;

    d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

    e. A brief description, including a summary of the safety evaluations of changes in the facility or in the procedures and of tests and experiments carried pursuant to Section 50.59 of 10 CFR Part 50;

    f. A summary of radiation exposures received by facility personnel and visitors, including dates and time of significant exposures and a summary of the results of radiation and contamination surveys performed within the facility; and

    g. A description of any environmental surveys performed outside the facility.

h. 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;

(1) Liquid Effluents (summarized on a monthly basis)

Liquid radioactivity discharged during the reporting period tabulated as follows:

(a) Total estimated radioactivity released (in curies).

(b) The isotopic composition if greater than 1×10^{-7} microcuries/cc for fission and activation products.

(c) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

(d) Average concentration at point of release (in microcuries/cc) during the reporting period and the fraction of the applicable limit in 10 CFR Part 20.

(e) Total volume (in gallons) of effluent water (including diluent) during periods of release.

(2) Exhaust Effluents (summarized on a monthly basis)

Radioactivity discharged during the reporting period (in curies) for:

(a) Gases.

(b) Particulates with half lives greater than eight days.

(c) The estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis and the fraction of the applicable 10 CFR Part 20 limits for these values.

(3) Solid Waste

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

(b) The total activity involved (in curies).

(c) The dates of shipment and disposition (if shipped off site).

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

   a. An evaluation of facility performance to date in comparison with design predictions and specifications, and

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

TS 6.7.2 Special Reports

1. There shall be a report of any of the following not later than the following day by telephone or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report describing the circumstances of the event and sent within 14 days to U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555:

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

   b. Any violation of a safety limit;

   c. Any reportable occurrences as defined in TS 1.3 of these specifications.

2. A written report within 30 days in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555 of:

   a. Permanent changes in facility organization at Reactor Director or Department Chair level.

   b. Any significant change in the transient or accident analysis as described in the Safety Analysis Report;

TS 6.8 Records

TS 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

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

2. Principal maintenance activities,

3. Reportable occurrences,

4. Surveillance activities required by the Technical Specifications,

5. Reactor facility radiation and contamination surveys where required by applicable regulations,

6. Experiments performed with the reactor,

7. Fuel inventories, receipts, and shipments,

8. Approved changes in operating procedures,

9. Records of meeting and audit reports of the review and audit group.

TS 6.8.2 Records to be Retained for at Least One Certification Cycle

Record of retraining and requalification of certified operations personnel shall be maintained at all times the individual is employed or until the certification is renewed. For the purposes of this technical specification, a certification is an NRC issued operator license.

TS 6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

Annual reports which contain the information in items 1 and 2 may be used as records for those items.

1. Gaseous and liquid radioactive effluents released to the environs,

2. Offsite environmental monitoring surveys required by technical specifications,

3. Radiation exposures for all personnel monitored,

4. Updated, corrected, and as-built drawings of the facility.

5. Notification that safety limit was exceeded.

6. Notification that automatic safety system did not function as required.

7. Notification of failure to meet limiting conditions for operation.

TS 7 REFERENCES

1. LEU Conversion SAR, 2008, as amended.

2. GA-9064, pages 3-1 to 3-23.

3. "Sagitta" refers to the bow in the fuel element and means the maximum excursion of the clad surface from a chord connecting the two ends of the clad surface over the length of the fuel.

4. Simnad and West, 1986.

5. NUREG-1282.