TEES | TEXAS A&M ENGINEERING EXPERIMENT STATION

NUCLEAR SCIENCE CENTER

June 5, 2015 2015-0036

Document Control Desk
ATTN: Geoffrey Wertz
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to U.S. NRC Request for Additional Information Regarding the Renewal of Facility Operating License No. R-83 for the Nuclear Science Center TRIGA Reactor (TAC No. ME 1584), from the Texas A&M University System, Texas Engineering Experiment Station, Nuclear Science Center Reactor (NSCR, License No. R-83, Docket 50-128)

To Whom It May Concern:

The Texas A&M University System, Texas Engineering Experiment Station (TEES), Nuclear Science Center (NSC, License No. R-83) operates a LEU, 1MW, TRIGA reactor under timely renewal. In December, 2003 the NSC submitted a Safety Analysis Report (SAR) as part of the license renewal process. In December, 2005 a conversion SAR (Chapter 18) was submitted resulting in an order to convert from the U.S. NRC. In July 2009, the NSC submitted an updated SAR, dated June 2009, to the U.S. NRC. This updated 2009 version of our SAR incorporated the information from the conversion SAR and the startup of the new LEU reactor core. On May 6, 2015 the U.S. NRC submitted a Request for Additional Information as a part of the review process. Attached is our reply to this request.

If you have any questions, please contact Dr. Sean McDeavitt or Mr. Jerry Newhouse at 979-845-7551.

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I declare under penalty of perjury that the foregoing is true and correct. Executed on June 5, 2015.

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Appendix A

TO

FACILITY LICENSE NO. R-83
DOCKET NO. 50-128

TECHNICAL SPECIFICATIONS AND BASES

TEXAS ENGINEERING EXPERIMENT STATION
NUCLEAR SCIENCE CENTER (NSC)

JUNE 2015

TECHNICAL SPECIFICATIONS

1 Introduction

1.1 Scope

This document constitutes the Technical Specifications for the Facility License No. R-83 as required by 10 CFR 50.36 and supersedes all prior Technical Specifications. This document includes the "bases" to support the selection and significance of the specifications. Each basis is included for information purposes only. They are not part of the Technical Specifications, and they do not constitute limitations or requirements to which the licensee must adhere.

1.2 Format

These specifications are formatted to NUREG-1537 and ANSI/ANS 15.1-2007.

1.3 Definitions

ALARA

The ALARA program (As Low as Reasonably Achievable) is a program for maintaining occupational exposures to radiation and release of radioactive effluents to the environs as low as reasonably achievable.

Audit

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

Channel

A channel is the combination of sensors, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a parameter.

Channel Test

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

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. This verification, where possible, shall include comparison of the channel with other independent channels or systems measuring the same variable.

Confinement

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

Control Rod

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

Regulating Control Rod

The regulating rod is a low-worth control rod used primarily to maintain an intended power level that need not have scram capability. Its axial position may be varied manually or by the servo controller.

Shim Safety Control Rod

A shim safety rod is a control rod having an electric motor drive and scram capabilities. It shall have a fueled follower section.

Transient Control Rod

The transient rod is a pneumatically driven control rod with scram capabilities that is capable of providing rapid reactivity insertion to produce a pulse.

Core Configuration

The core configuration includes the number, type, or arrangement of fuel elements, reflector elements, and regulating/shim-safety/transient rods occupying the core grid.

Core Lattice Position

The core lattice position is that region in the core (approximately 3" x 3") over a grid-plug hole. A fuel bundle, an experiment, or a reflector element may occupy the position.

Excess Reactivity

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

Experiment

An operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate non-routine reactor characteristics, or that is intended for irradiation within the pool, or in a beam port or irradiation facility. Hardware rigidly secured to a core or shield structure so as to be a part of its design to carry out experiments is not normally considered an experiment.

Secured Experiment

A secured experiment is any experiment, experiment facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

Unsecured Experiment

An unsecured experiment is any experiment or component of an experiment that does not meet the definition of a secured experiment.

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.

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 in-pool irradiation facilities.

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 for operator intervention.

Fuel Bundle

A fuel bundle is a cluster of two, three, or four fuel elements and/or non-fueled elements secured in a square array by a top handle and a bottom grid plate adapter. Non-fueled elements shall be fabricated from stainless steel, aluminum, boron, or graphite materials.

Fuel Element

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

Instrumented Fuel Element (IFE)

An instrumented fuel element is a special fuel element in which one or more thermocouples are embedded for the purpose of measuring the fuel temperatures during operation.

License

The written authorization, by the U.S. NRC, for an individual or organization to carry out the duties and responsibilities associated with a personnel position, material, or facility requiring licensing.

Licensee

A licensee is an individual or organization holding a license.

LEU Core

An LEU core is an arrangement of TRIGA-LEU fuel in a reactor grid plate.

Limiting Safety System Setting (LSSS)

The limiting safety system setting is the fuel element temperature, which if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded.

Measured Value

A measured value is the value of a parameter as it appears on the output of a channel.

Operable

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

Operating

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

Operational Core – Steady State

A steady state operational core shall be an LEU core which meets the requirements of the Technical Specifications.

Operational Core – Pulse

A pulse operational core is a steady state operational core for which the maximum allowable pulse reactivity insertion has been determined.

Pool Water Reference Operating Level

The pool water reference operating level is 10 inches below the top of the pool wall. This level is designed to prevent pool water from rising above the top of the liner.

Protective Action

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

Pulse Mode

Pulse mode operation shall mean 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 Console Secured

The reactor console is secured whenever all control rods have been verified to be fully inserted and the console key has been removed from the console.

Reactor Operating

The reactor is operating whenever it is not secured or shutdown.

Reactor Operator

A Reactor Operator is 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 channels, which are designed to initiate automatic reactor protection or to provide information for initiation of manual protective action.

Reactor Secured

The reactor is secured when:

Either

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

Or

(2) All of the following conditions exist:

(a) All control rods are fully inserted;

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

(c) The reactor is shutdown;

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

(e) No experiments are moved or serviced that have, on movement, a reactivity worth exceeding $1.00.

Reactor Shutdown

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

Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is less than $0.01.

Reportable Occurrence

Any of the following events is a reportable occurrence:

(1) Operation with actual safety system settings for required systems less conservative than the LSSS specified in the Technical Specifications;

(2) Operation in violation of a Limiting Condition of Operation listed in Section 3 unless prompt remedial action is taken as permitted in Section 3;

(3) Operation with a required reactor or experiment safety system component in an inoperative or failed condition which renders or could render the system incapable of performing its intended safety function. If the malfunction or condition is caused during maintenance, then no report is required;

(4) An unanticipated or uncontrolled change in reactivity greater than $1.00. Reactor trips resulting from a known cause are excluded;

(5) Abnormal and significant degradation in reactor fuel or cladding, or both, coolant boundary, or confinement boundary; and

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

Review

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

Safety Channel

A safety channel is a channel in the reactor safety system.

Safety Limit

Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. For the Texas A&M NSC TRIGA reactor the safety limit is the maximum fuel element temperature that can be permitted with confidence that no damage to any fuel element cladding will result.

Scram Time

Scram time is the elapsed time between the initiation of a scram signal and the instant that the slowest scrammable control rod reaches its fully inserted position.

Senior Reactor Operator

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

Shall, Should and May

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

Shutdown Margin

Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition. This margin is determined assuming that the most reactive scrammable rod and any non-scrammable rods are fully withdrawn, and that the reactor will remain subcritical by this calculated margin without any further operator action.

Steady State Mode

Steady state mode of operation shall mean operation of the reactor with the mode selector switch in the steady state position.

Surveillance Intervals

The maximum surveillance intervals are provided for operational flexibility and the average surveillance intervals should be maintained over the long term.

Annually - an interval not to exceed 15 months.
Biennially - an interval not to exceed 30 months.
Monthly - an interval not to exceed 6 weeks.
Quarterly - an interval not to exceed 4 months.
Semiannually - an interval not to exceed 7.5 months.
Weekly - an interval not to exceed 10 days.

True Value

The true value is the actual value of a parameter.

Unscheduled Shutdown

An unscheduled shutdown is any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation. It does not include shutdowns that occur during testing or check out operations.

2 Safety Limit and Limiting Safety System Setting

2.1 Safety Limit-Fuel Element Temperature

Applicability

This specification applies to the temperature of the reactor fuel.

Objective

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

Specification

The temperature in a stainless steel-clad TRIGA LEU fuel element shall not exceed 2100 °F (1150°C) under any conditions of operation.

Basis

The most important safety limit for a TRIGA reactor is fuel element temperature. This parameter is well suited as a single specification because it can be measured directly with a thermocouple. A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure if the fuel element temperature exceeds the temperature safety limit. The fuel element temperature and the ratio of hydrogen to zirconium in the fuel-moderator material determine the magnitude of the pressure buildup. The mechanism for the pressure buildup is the dissociation of hydrogen from the zirconium hydride moderator that has been blended with uranium to form the fuel mixture encased within the fuel element cladding.

The temperature safety limit for the LEU fuel element is based on data which indicates that the internal stresses within the fuel element, due to hydrogen pressure from the dissociation of the zirconium hydride, will not result in compromise of the stainless steel cladding if the fuel temperature is not allowed to exceed 2100°F (1150°C) and the fuel element cladding is water cooled.

2.2 Limiting Safety System Setting

Applicability

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

Objective

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

Specification

The limiting safety system setting shall be 975°F (525°C) as measured in an instrumented fuel element (IFE). The IFE shall be located adjacent to the central bundle with the exception of the corner positions.

Basis

The limiting safety system setting (LSSS) is a temperature that, if exceeded, will cause a reactor scram to be initiated preventing the safety limit from being exceeded.

The temperature safety limit for LEU fuel is 2100°F (1150°C). Due to various errors in measuring temperature in the core, it is necessary to arrive at a LSSS for the fuel element safety limit that takes into account these measurement errors. The results of analysis provided in SAR 13.3 Evaluation of LSSS for NSC LEU 30/20 Fuel indicate that a LSSS temperature of 975°F (525°C) is appropriate.

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

3 Limiting Conditions for Operation

3.1 Reactor Core Parameters

3.1.1 Steady State Operation

Applicability

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

Objective

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

Specification

The reactor power level shall not exceed 1.0 megawatt (MW) during steady state operation.

Basis

Calculations show that reactor operations with a pool temperature of 60°C will not risk reaching a fuel temperature greater than the LSSS, or a DNBR of unity for 1MW steady state. (Response to U.S. NRC Request for Additional Information, Review of The Fuel Pool Temperature on Fuel Temperature, submitted November 13, 2014).

3.1.2 Pulse Mode Operation

Applicability

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

Objective

The objective is to ensure that respective pulsing will not induce damage to the reactor fuel.

Specification

The reactivity to be inserted for pulse operation shall not exceed that amount which will produce a peak fuel temperature of 1526°F (830°C). In the pulse mode the pulse rod shall be limited by mechanical means or the rod extension physically shortened so that the reactivity insertion will not inadvertently exceed the maximum value.

Basis

The pulsing limit of 830°C will be translated to a reactivity insertion limit for each specific core. The peaking factors from the thermocouple element to the hottest spot in the core must be calculated for each core configuration that is to be used. Temperature would then be measured for small pulse insertions.

The initial core calibration in 2006 established the maximum allowable pulse insertion to be $1.91. Any subsequent pulse insertion change shall only be made after core recalibration and following approval by the NSC staff.

3.1.3 Shutdown Margin

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worth of control rods and experiments. They apply for all modes of operation.

Objective

The objective is to ensure that the reactor can be shutdown at all times and to ensure that the fuel temperature safety limit will not be exceeded.

Specification

The reactor shall not be operated unless the shutdown margin provided by control rods is greater than $0.50 with:

1. Irradiation facilities and experiments in place and the total worth of all non-secured experiments in their most reactive state,

2. The highest worth control rod and the regulating rod fully withdrawn, and

3. The reactor in reference core condition.

Basis

The value of the shutdown margin ensures that the reactor can be shut down from any operating condition even if the highest worth control rod should remain in the fully withdrawn position. Since the regulating rod is not scrammable, its worth is not used in determining the shutdown reactivity.

3.1.4 Reactor Core Configuration

Applicability

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

Objective

The objective is to ensure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specification

1. Control rods shall not be manually removed from the core unless the core has been shown to be subcritical, and shutdown margin requirements met, with those control rods removed.

2. Core lattice positions shall not be vacant except for positions on the periphery of the core assembly while the reactor is operating. Water holes in the inner fuel region shall be limited to single rod positions. Vacant core positions not on the periphery shall contain experiments or an experimental facility to prevent accidental fuel additions to the core.

3. The instrumented fuel element, if serving as the Limiting Safety System sensor, shall be located adjacent to the central bundle with the exception of the corner positions.

Basis

1. Manipulation of core components will be allowed only when a single manipulation cannot result in inadvertent criticality.

2. Vacant core positions containing experiments or an experimental facility will prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core or a single rod position to prevent power peaking in regions of high power density.

3. SAR 13.3.1 Steady State Mode provides an evaluation of the LSSS in steady state mode. It states in part, "The location of the fuel cluster containing the instrumented fuel element shall be chosen to be as close as possible to the hottest fuel element in the core." The location(s) as close as possible to the hottest fuel element in the core are those adjacent to the central bundle with the exception of the corner positions. These adjacent positions are: C5 east, D4 north, E5 west, D6 south.

3.1.5 Reactor Fuel Parameters

Applicability

This specification applies to all fuel elements.

Objective

The objective is to maintain the integrity of the fuel elements' cladding.

Specification

1. The reactor shall not be operated knowingly with damaged fuel, except for the purpose of locating damaged fuel elements.
2. A fuel element shall be considered damaged and must be removed from the core if:
    a. In measuring the transverse bend, the bend exceeds 0.125 inch over the length of the cladding, or
    b. In measuring the elongation, its length exceeds its original length by 0.125 inch, or
    c. A clad defect exists as indicated by release of fission products, or
    d. A visual inspection reveals bulges, gross pitting or corrosion.
3. The burnup of the uranium-235 in the UZrH fuel matrix shall not exceed 50 percent of the initial concentration.

Basis

Gross failure or obvious visual deterioration of the fuel is sufficient to warrant declaration of the fuel as damaged. The elongation, bend, and burn-up limits are values that have been found acceptable to the U.S. NRC (NUREG-1537).

3.1.6 Maximum Excess Reactivity

Applicability

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

Objective

The objective is to ensure that the reactor can be shutdown at all times.

Specification

The maximum reactivity in excess of reference core condition shall not exceed 5.5% Δk/k ($7.85).

Basis

The nominal total rod worth is $16.00 (SAR 4.5.3.2 and historic data). Subtracting the shutdown margin ($0.50), the nominal rod worth of the most reactive rod ($4.00), and adding the allowed total reactivity worth for all experiments ($5.00) gives a result of $16.50.

The specification of $7.85, although over-constraining the reactor system, helps ensure that the licensee's operational power densities, fuel temperatures, and temperature peaks are maintained within the evaluated safety limits. The specified excess reactivity allows for power coefficients of reactivity, xenon poisoning, most experiments, and operational flexibility.

3.2 Reactor Control and Safety Systems

3.2.1 Reactor Channels

Applicability

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

Objective

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

Specification

The reactor shall not be operated in the specific mode of operation unless the channels listed in Table 1 are operable.

Table 1: Channels Required for Operation

Channel	Minimum Number Operable	Operating Mode	
		S.S.	Pulse
Fuel Element Temperature	1	X	X
Linear Power Level	1	X	-
Log Power Level	1	X	-
Integrated Pulse Power	1	-	X
Pool Water Temperature	1	X	X

Basis

Fuel temperature displayed at the control console gives continuous information on this parameter, which has a specified safety limit. The power level monitors ensure that the reactor power level is adequately monitored for both steady state and pulsing modes of operation. The specification on reactor power level indications are included in this section, since the power level is related to fuel temperature. The specification on pool water temperature indication is included in this section to allow monitoring in support of TS 3.8.3 and 4.8.3.

3.2.2 Reactor Safety Systems and Interlocks

Applicability

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

Objective

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

Specification

The reactor shall not be operated unless the safety circuits and interlocks described in Tables 2a and 2b are operable. However, any single safety channel or interlock may be inoperable when the reactor is operating for the purpose of performing a channel check, channel test, or channel calibration. If any required safety channel or interlock becomes inoperable while the reactor is operating, for reasons other than identified in this TS, then the channel shall be restored to operation within 5 minutes or the reactor shall be immediately shutdown.

Table 2a: Safety Channels Required for Operation

Safety Channel	Number Operable	Function	Operating Mode	
			S.S.	Pulse
Fuel Element Temperature	1	Scram ≤ 975°F	X	X
High Power Level	2	Scram ≤ 1.25MW	X	-
Console Scram Button	1	Manual Scram	X	X
High Power Level Detector Power Supply	2	Scram on loss of supply voltage	X	-
Preset Timer	1	Transient Rod Scram 15 seconds or less after pulse	-	X
Pool Water Temperature	1	Manual scram if temperature reaches 60 C	X	X

Table 2b: Interlocks Required for Operation

Safety Channel	Number Operable	Function	Operating Mode	
			S.S.	Pulse
Log Power	1	Prevents withdrawal of the Shim Safety Control Rods at an indicated log power of less than $4x10^{-3}$ W.	X	-
Log Power	1	Prevents pulsing of the Transient Rod when log power is above 1 kW.	-	X
Transient Rod Position	1	Prevents application of air to the Transient Rod unless the Transient Rod is fully inserted.	X	-
Shim Safety and Regulating Rod Position	1	Prevents Shim Safety and Regulating Control Rod withdrawal during a pulse.	-	X
Pulse Stop Electro-Mechanical Interlock	1	Prevents application of air to the Transient Rod unless the mechanical pulse stop is installed.	-	X

Basis

During period of maintenance, surveillance, calibration, and repair, true signals generated by the reactor may be required. This operation is allowable provided all other TS conditions are met.

Safety Channels Required for Operation

1. The fuel temperature and high power level scrams provide protection to ensure that the reactor can be shutdown before the safety limit on fuel element temperature will be exceeded.

2. The manual console scram allows the operator to shut down the system if an unsafe or abnormal condition occurs.

3. In the event of failure of the power supply for a high power level safety detector, operation of the reactor without adequate instrumentation is prevented.

4. The preset timer ensures that the reactor power level will reduce to a low level after pulsing.

5. A manual scram is a sufficient response because pool water temperature is a slow-changing parameter. It is surveyed frequently enough to give time for an operator to respond.

Interlocks Required for Operation

1. The interlock to prevent startup of the reactor at power levels less than 4 x 10-3 W, which corresponds to approximately 2 cps, ensures that sufficient neutrons are available for proper indication.

2. The interlock to prevent pulsing at powers above 1 kW ensures that the magnitude of the pulse will not cause the fuel element temperature safety limits to be exceeded.

3. The interlock to prevent application of air to the transient rod unless the cylinder is fully inserted is to prevent pulsing of the reactor in steady state mode.

4. The interlock to prevent the withdrawal of the shim safeties or regulating rod in the pulse mode is to prevent the reactor from being pulsed while on a positive period.

5. The interlock to prevent application of air to the transient rod unless the mechanical pulse stop is installed prevents a reactor pulse of sufficient worth to exceed the temperature safety limit.

3.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

Applicability

This specification applies to the minimum number of operable scrammable control rods in the core, where operable is specified in terms of maximum scram time from the instant that any SCRAM signal is initiated.

Objective

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

Specification

During operation, all control rods shall be operable. For scrammable control rods, the scram time measured from the instant a SCRAM signal is initiated to the instant that the slowest scrammable rod reaches its fully inserted position shall not exceed 1.2 seconds. During core manipulations, i.e. core loading and unloading, all installed control rods shall be operable.

Basis

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

3.3 Confinement

3.3.1 Operations that Require Confinement

Applicability

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

Objective

To provide restrictions on radioactive airborne material releases into the environment.

Specification[1]

Confinement of the reactor building shall be required during the following operations:

1. Reactor operating;
2. Movement of irradiated fuel elements or fuel bundles;
3. Core or control rod work that could cause a change in reactivity of more than one dollar; or
4. Handling of radioactive materials with the potential for airborne release.

[1] For periods of maintenance to the central exhaust system, entry doors to the reactor building shall remain closed except for momentary opening for personnel entry or exit. The central exhaust system shall not remain inoperable during periods of maintenance for more than one hour.

Basis

This specification describe when the central exhaust system shall operate to control any release of radioactive material in the confinement building.

3.3.2 Equipment to Achieve Confinement

Applicability

This specification applies to the equipment and controls needed to provide confinement of the reactor building.

Objective

The objective is to ensure that a minimum of equipment is in operation to achieve confinement as specified in Section 3.3.1 and that the control panel for this equipment is available for normal and emergency situations.

Specification[2]

1. The minimum equipment required to be in operation to achieve confinement of the reactor building shall be the central exhaust system, which consists of the central exhaust fan, isolation louvers, and associated duct work.

2. The central exhaust system shall be considered operating when it creates a minimum of 0.1 inch of water negative pressure at the sample point in the central exhaust system duct work.

3. Controls for establishing the operation of the central exhaust system during normal and emergency conditions shall be available in the Emergency Support Center.

4. The central exhaust system shall be isolated automatically by alarm level signals from the stack particulate, or stack gas (xenon) facility air monitor.

[2] During periods of maintenance to the central exhaust system, entry doors to the reactor building shall remain closed except for momentary opening for personnel entry or exit. The central exhaust fan system shall not remain inoperative during periods of maintenance for more than one hour.

Basis

1. Operation of the central exhaust fan will achieve confinement of the reactor building during normal and emergency conditions when the controls for air input are set such that the central exhaust fan capacity remains greater than the amount of air being delivered to the reactor building. The exhaust fan has sufficient capacity to handle extra air intake to the building during momentary opening of doors. Isolation of the central exhaust fan for periods of less than one hour is for operability verification during weekly ventilation checks. This limit provides enough time to complete these checks.

2. Negative pressure in the confinement building mitigates leakage of unmonitored airborne material to the environment.

3. The control panel for the central exhaust system provides for manual selection of air input to the reactor building and the automatic or manual selection of air removal. The air supply and exhaust systems work together to maintain a small negative pressure in the reactor building. These controls are available in the emergency support center for accessibility during emergency conditions.

4. An automatic isolation of the central exhaust system will mitigate leakage of unmonitored airborne material to the environment.

3.4 Ventilation System

The LCO for Ventilation System is covered by TS 3.3.2 Equipment to Achieve Confinement

3.5 Radiation Monitoring Systems and Effluents

3.5.1 Radiation Monitoring

Applicability

This specification applies to the radiation monitoring information that must be available to the reactor operator during reactor operation, movement of irradiated fuel elements or fuel bundles, conduct of core or control rod work that could cause a change in reactivity of more than one dollar, or handling of radioactive materials with the potential for airborne release.

Objective

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

Specification

Reactor operation, movement of irradiated fuel elements or fuel bundles, conduct of core or control rod work that could cause a change in reactivity of more than one dollar, or handling of radioactive materials with the potential for airborne release shall not be conducted unless the radiation monitoring channels listed in Table 3 are operable, displays and alarms are operable in the control room, and displays are operable in the Emergency Support Center.

Table 3: Radiation Monitoring Channels Required for Operation[3]

Radiation Monitoring Channels	Function	Number
Reactor Bridge ARM	Monitor radiation levels within the reactor bay	1
Stack Particulate Monitor (FAM Ch. 1)	Monitor radiation levels in the exhaust air stack	1
Stack Gas Monitor (FAM Ch. 3)	Monitor radiation levels in the exhaust air stack	1
Building Particulate Monitor (FAM Ch. 4)	Monitor radiation levels within the reactor bay	1
Stack Xenon Monitor (FAM Ch. 5)	Monitor radiation levels in the exhaust air stack	1

[3] When a required channel becomes inoperable, operations may continue only if a portable gamma-sensitive ion chamber is utilized as a temporary substitute, provided that the substitute can be observed by the reactor operator, can be installed within 1 hour of discovery, and is not used longer than one week. If two of the above monitors are not operating, operations shall cease.

Basis

The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the environment.

3.5.2 Argon-41 Discharge Limit

Applicability

This specification applies to the concentration of Argon-41 (^{41}Ar) that may be discharged from the TRIGA reactor facility.

Objective

The objective is to ensure that the health and safety of the public is not endangered by the discharge of ^{41}Ar from the TRIGA reactor facility.

Specification

The total annual discharge of ^{41}Ar into the environment shall not exceed 30 Ci per year.

Basis

If the 30 Ci is assumed to be released continuously over one year, then the Ar-41 concentration at the point of discharge, which is the top of the stack, is 2.5×10^{-7} μCi/ml. This concentration is diluted by a factor of 200 to get the Ar-41 concentration at the site boundary 1.0×10^{-9} μCi/ml. 1.0×10^{-9} μCi/ml corresponds to a dose of 12.6 mrem.

3.6 Limitations on Experiments

3.6.1 Reactivity Limits

Applicability

This specification applies to the reactivity limits on 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 case of failure of an experiment.

Specification

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

1. The absolute reactivity worth of any single, movable or unsecured experiment shall be less than $1,

2. The reactivity worth of any secured experiment shall be less than $2, and

3. The sum of the absolute reactivity of all experiments shall be less than $5.

Basis

1. This specification is intended to ensure that the worth of a single unsecured experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were suddenly inserted. This does not restrict the number of unsecured experiments adjacent to or in the reactor core except by reactivity worth and the requirements of these TS.

2. The maximum worth of a single secured experiment is limited so that its removal from the reactor in reference core condition will not result in the reactor achieving a power level high enough to exceed the fuel element temperature safety limit. Since experiments of such worth must be secured, 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.

3. This limit poses a restriction on the total absolute reactivity of experiments being run at any given time to prevent excessive positive and negative reactivity effects from experiments.

3.6.2 Material Limitations

Applicability

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

Objective

The objective is to prevent damage to the reactor or excessive release of radioactivity by limiting materials quantity and radioactive material inventory of the experiment.

Specification

1. Explosive materials in quantities inclusively between 25 milligrams and 5 pounds (TNT-equivalent) shall not be allowed within the reactor building except as noted below in TS. Explosive materials in quantities greater than 5 pounds (TNT-equivalent) shall not be allowed within the reactor building. Irradiation of explosive materials shall be restricted as follows:

    a. Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall not be irradiated in the reactor pool. Explosive materials in quantities up to 25 milligrams (TNT-equivalent) 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;

    b. Explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall only be allowed in the lower research level and laboratory building, excluding the heat exchanger room and demineralizer room;

    c. Irradiation of explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall be permitted only in the neutron radiograph facility;

    d. Explosive materials in quantities greater than or equal to 5 pounds (TNT-equivalent) shall not be irradiated in experimental facilities; and

    e. Cumulative exposures for explosive materials in quantities greater than or equal to 25 milligrams (TNT-equivalent) shall not exceed 10^{12} n/cm^2 for neutron or 25 Roentgen for gamma exposures.

2. Corrosive materials used in a reactor experiment shall be double encapsulated. Exceptions may only be made if a detailed analysis and/or prototype testing with small amounts of materials demonstrates that the experiment presents negligible risk.

Basis

1. This specification is intended to prevent damage to the reactor or reactor safety systems resulting from failure of an experiment involving explosive materials.

    a. This specification is intended to prevent damage to the reactor core and safety related reactor components located within the reactor pool in the event of failure of an experiment involving the irradiation of explosive materials. Limited quantities of less than 25 milligrams (TNT-equivalent) and proper containment of such experiment provide the required safety for in-pool irradiation provided that 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. (Regulatory Guide 2.2)

b. This specification is intended to prevent damage to vital equipment by restricting the quantity and location of explosive materials within the reactor building. Explosives in quantities exceeding 25 milligrams (TNT-equivalent) are restricted from areas containing the reactor bridge, reactor console, pool water coolant and purification systems, and reactor safety related equipment. (Amendment No. 7 to Facility License No. R-83)

c. This specification supports the same goal as the previous specification. The neutron radiograph facility was analyzed and shown to be able to withstand an explosion of the described quantity. (Amendment No. 7 to Facility License No. R-83)

d. The failure of an experiment involving the irradiation of up to 5 pounds (TNT-equivalent) of explosive material in an experimental facility located external to the reactor pool structure will not result in damage to the reactor or the reactor pool containment structure.

e. This specification is intended to prevent any increase in the sensitivity of explosive materials due to radiation damage during exposures.

   See: Kaufman J.V.R. (Jul. 29, 1958). The Effect of Nuclear Radiation on Explosives. *Proceedings of the Royal Society of London. Series A, Mathematical and Physical Science, Vol. 246* (No. 1245, A Discussion on the Initiation and Growth of Explosion in Solids);

   Urizar M.J., Loughran E.D., Smith L.C. (Jan. 01, 1960). A Study of the Effects of Nuclear Radiation on Organic Explosives; and

   Amendment No. 7 to Facility License No. R-83.

2. This specification is intended to prevent damage to the reactor or reactor safety systems resulting from failure of an experiment involving corrosive materials.

3.6.3 Failures and Malfunctions

Applicability

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

Objective

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

Specification

1. Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (a) normal operating conditions of the experiment or reactor, (b) credible accident conditions in the reactor, or (c) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor building or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10CFR20.

2. In calculations pursuant to 1) above, the following assumptions shall be used:

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

    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 vapors can escape; or

    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.

3. If a capsule fails and releases material that could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection 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 NSC Director or his designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Basis

1. This specification is intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR 20 will be released to the atmosphere outside the facility boundary of the NSC.

2. These assumptions are used to evaluate the potential airborne radioactivity release due to an experiment failure.

3. Operation of the reactor with reactor fuel or structure damage is prohibited to avoid release of fission products. Potential damage to reactor fuel or structure must be brought to the attention of the NSC Director or his designated alternate for review to ensure safe operation of the reactor.

3.7 As Low As Reasonable Achievable (ALARA) Radioactive Effluents Released

Applicability

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

Objective

The objective is to constrain the annual radiation exposure to the general public resulting from operation of the reactor to a level as low are reasonably achievable below the constraints listed in 10 CFR 20.1101.

Specification

1. In addition to the radiation monitoring specified in Section 5.5, an environmental radiation monitoring program shall be conducted to measure the integrated radiation exposure in and around the environs of the facility on a quarterly basis.

2. The annual radiation exposure (dose) to the public due to reactor operation shall not exceed the limits defined in 10 CFR 20.1301. The facility perimeter shall be monitored to ensure this specification is being met.

3. In the event of a fission product leak from a fuel rod or an airborne radioactive release from a sample being irradiated, as detected by the facility air monitor (FAM), the reactor shall be shut down until the source of the leak is located and eliminated. However, the reactor may continue to be operated on a short-term basis, as needed, to assist in determining the source of the leakage.

4. The facility liquid effluents collected in the holdup tanks shall be discharged in accordance with 10 CFR 20.2003 "Disposal by release into sanitary sewerage." The liquid effluent shall also meet local sanitary sewer discharge requirements.

Basis

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

3.8 Primary Coolant Conditions

3.8.1 Primary Coolant Purity

Applicability

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

Objective

The objectives are to minimize the possibility for corrosion of the cladding on the fuel elements and to minimize neutron activation of dissolved materials.

Specification

1. The reactor shall not be operated for a period exceeding two weeks if the two week averaged conductivity of the bulk pool water is higher than 5×10^{-6} mhos/cm.

2. The concentrations of radionuclides in the bulk pool water shall be no higher than the values presented for water in 10 CFR Appendix B to Part 20 Table 2.

Basis

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

By limiting the concentrations of dissolved materials in the water, the radioactivity of neutron activation products is limited. This is consistent with the ALARA principle, and tends to decrease the inventory of radionuclides in the entire coolant system, which will decrease personnel exposure during maintenance and operations.

3.8.2 Primary Coolant Level and Leak Detection

Applicability

This specification applies to the water level that must be in the pool and requirements for leak detection for reactor operation.

Objective

The objective is to ensure proper shielding and cooling of the reactor and the ability to detect leaks.

Specification

1. The reactor shall not be operated if the pool level is below 3 feet from the reference operating level.

2. The reactor shall not be operated if the pool level unexpectedly drops one foot from its operating level.

3. The pool level alarm shall initiate an alarm signal in the control room and at a continuously monitored off-site facility if the pool level is lower than 3 feet from its reference operating level.

Basis

1. The intake to the diffuser system is approximately 18 feet above the core and 8.5 feet below the reference operating level. Setting this level as the specification will both ensure the availability of the diffuser system, and provide more than adequate shielding and cooling for the reactor.

2. An unexpected one foot drop from the operating pool level, whatever level that may be, indicates leakage.

3. An operable pool level alarm that provides an off-site alarm will ensure proper notification if a low pool level or significant unexpected change occurs.

3.8.3 Primary Coolant Temperature

Applicability

This specification applies to the maximum allowable primary coolant temperature.

Objective

The objective is to maintain fuel temperature less than the LSSS, and to maintain the departure of nucleate boiling ratio (DNBR) greater than unity, and to limit any degradation of the reactor systems.

Specification

The reactor shall not be operated when pool temperature exceeds 60° C.

Basis

Calculations show that reactor operations with a pool temperature of 60 C will not risk reaching a fuel temperature greater than the LSSS, or a DNBR of unity. In fact, a conservative calculation predicts a DNBR of 1.54. (Response to U.S. NRC Request for Additional Information, Review of The Fuel Pool Temperature on Fuel Temperature… dated November 13, 2014). For reactor pulses, the NSC already accounts for pool temperature. As described in the SAR (4.5.13 Pulse Operation – NSC – BOL, Measured), the NSC calculates peak core temperature. The temperature in this calculation is in °C. Ambient fuel temperature is already included in pulse calculations.

4 Surveillance Requirements

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.

Specification

Surveillance requirements may be deferred during reactor shutdown (except TS 4.1.5, 4.2.3, 4.5, 4.8.1, and 4.8.2); however, they shall be completed prior to reactor startup unless reactor operation is required for performance of the surveillance. Such surveillance shall be performed as soon as practical after reactor startup. Scheduled surveillance, which cannot be performed with the reactor operating, may be deferred until a planned reactor shutdown.

	TS	Possible to Defer During Shutdowns?	Required prior to operations?
1.	4.1.1 Steady State Operation	Yes	Yes
2.	4.1.2 Pulse Mode Operation	Yes	Yes
3.	4.1.3 Shutdown Margin	Yes	Yes
4.	4.1.4 Core Configuration Limitation	Yes	Yes
5.	4.1.5 Reactor Fuel Elements	No	N/A
6.	4.1.6 Maximum Excess Reactivity	Yes	Yes
7.	4.2.1 Reactor Channels	Yes	Yes
8.	4.2.2 Reactor Safety Systems and Interlocks	Yes	Yes
9.	4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time	No	N/A
10.	4.3 Confinement	Yes	Yes
11.	4.5 Radiation Monitoring Systems and Effluents	No	N/A
12.	4.6 Experiments	Yes	Yes
13.	4.8.1 Primary Coolant Purity	No	N/A
14.	4.8.2 Primary Coolant Level and Leak Detection	No	N/A
15.	4.8.3 Primary Coolant Temperature	Yes	Yes

Any additions, modifications, or maintenance to the central exhaust 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 Board. A system shall not be considered operable until it is successfully tested.

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.

4.1 Reactor Core Parameters

4.1.1 Steady State Operation

Applicability

This specification applies to the surveillance requirement of the power level monitoring channels.

Objective

The objective is to verify that the maximum power level of the reactor meets the license requirements.

Specification

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

Basis

The power level channel calibration will ensure that the reactor will be operated at the proper power level.

4.1.2 Pulse Mode Operation

Applicability

This specification applies to the surveillance requirements for operation of the reactor in the pulse mode.

Objective

The objective is to verify that operation of the reactor in the pulse mode is proper and safe and to determine if any significant changes in fuel characteristics have occurred.

Specification

The reactor shall be pulsed semiannually to compare fuel temperature measurements and core pulse energy with those of previous pulses of the same reactivity value. The reactor shall not be declared operational for pulsing until such pulse measurements are performed and are determined to be acceptable.

Basis

The reactor is pulsed at suitable intervals to make a comparison with previous similar pulses and to determine if changes in fuel or core characteristics are taking place.

4.1.3 Shutdown Margin

Applicability

This specification applies to the surveillance requirement of control rod calibrations and shutdown margin.

Objective

The objective is to verify that the requirements for shutdown margins are met for operational cores.

Specification

The reactivity worth of each control rod and the shutdown margin shall be determined annually and following changes in the core, in-core experiments, or control rods.

Basis

The reactivity worth of the control rods is measured to ensure that the required shutdown margin is available and to provide an accurate means for determining the reactivity worth of experiments inserted in the core. Experience with TRIGA reactors gives assurance that measurement of the reactivity worth on an annual basis is adequate to ensure no significant changes in the shutdown margin.

4.1.4 Core Configuration Limitation

Applicability

This specification applies to the surveillance requirements for core configuration.

Objective

The objective is to verify the core is in a safe, reviewed, and approved configuration.

Specification

Each core configuration change shall be determined to meet the requirements of TS 3.1.4 prior to the core loading.

Basis

The requirements of TS 3.1.4 ensure acceptable safety analysis is complete for a core configuration, as well as prevent accidental fuel damage, fuel addition, or criticality events.

4.1.5 Reactor Fuel Elements

Applicability

This specification applies to the surveillance requirements for the fuel elements.

Objective

The objective is to verify the continuing integrity of the fuel element cladding and to ensure that no fuel damage has occurred.

Specification

1. The following fuel elements shall be inspected visually for damage or deterioration and measured for length and bend annually:

    a. At least four elements which occupy the highest pulse temperature positions in the core,

    b. At least one-fifth of the fuel elements used in operation of the reactor over the previous inspection year,

    c. The four elements (a) above may be included in the inspection of the fuel elements of (b) above, and

    d. Over a 5 year period every fuel element used in operation of the reactor shall be inspected.

2. If any element is found to be damaged, the entire core will be inspected.

Basis

The frequency of inspection is based on the parameters most likely to affect the fuel cladding of a pulsing reactor operated at moderate pulsing levels and utilizing fuel elements whose characteristics are well known. Experience has shown that temperature is the major contributor to fuel damage. Inspection of four fuel elements which occupy the highest pulse temperature positions in the core provides surveillance for detection of the most probable fuel element damage should it occur. Inspection of one-fifth of the elements used in operation of the reactor provides surveillance of the lower temperature elements and over a five year period provides for inspection of all elements.

4.1.6 Maximum Excess Reactivity

Applicability

This specification applies to the surveillance requirements of reactor excess reactivity.

Objective

The objective is to verify that requirements on excess reactivity are met for operational cores.

Specification

The excess reactivity shall be determined annually and following changes in the core, in-core experiments, or control rods for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

Basis

The excess reactivity of the core is measured to ensure that during all states of operation criticality can be maintained for licensed operational limits. With the accumulation of fission product poison buildup and fissile material burnup, excess reactivity must be available for power transients and maintaining criticality.

4.2 Reactor Control and Safety Systems

4.2.1 Reactor Channels

Applicability

This specification applies to the surveillance requirements for reactor channels.

Objective

The objective is to verify the condition and operability of system components directly related to channels that measure key reactor parameters.

Specification

A channel test of each of the reactor channels for the intended mode of operation, as identified in Table 1, shall be performed before each day's operation or before each operation extending more than one day.

Basis

Channel tests will ensure that the safety system channels are operable on a daily basis or prior to an extended run.

4.2.2 Reactor Safety Systems and Interlocks

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 the systems and components that are directly related to reactor safety.

Specification

1. A channel test of each of the reactor safety system channels and interlocks for the intended mode of operation, as identified in Table 2, shall be performed before each day's operation or before each operation extending more than one day.

2. A channel calibration of the fuel element temperature channels shall be performed semiannually.

Basis

Channel tests will ensure that the safety system channels are operable on a daily basis or prior to an extended run. If the period between operations extends beyond a year, then the annual channel test requirement will ensure operability.

4.2.3 Minimum Number of Operable Scrammable Control Rods and Scram Time

Applicability

This specification applies to the surveillance requirements for reactor control systems.

Objective

The objective is to verify the condition and operability of system components affecting safe and proper control of the reactor.

Specification

1. The control rods shall be visually inspected for deterioration biennially.

2. Operability tests of the control rod mechanism shall follow modification or repairs.

3. The Transient Rod drive cylinder and associated air supply system shall be inspected, cleaned and lubricated semiannually.

4. The scram time shall be measured annually or whenever any work is done on the control rods or the control rod drive system.

Basis

1. The visual inspection of the control rods is made to evaluate corrosion and wear characteristics caused by operation of the reactor.

2. These tests provide verification that the control rod has full travel and that the rod drop time is within specification.

3. Inspection and maintenance of the transient rod drive assembly reduces the probability of failure of the system due to moisture-induced corrosion of the pulse cylinder and piston rod assembly.

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

4.3 Confinement

Applicability

This specification applies to the central exhaust system.

Objective

The objective is to ensure the proper operation of the central exhaust system to prevent uncontrolled releases of radioactive material to the environment.

Specification

1. The central exhaust system shall be channel checked prior to reactor operation or radioactive material handling.

2. During periods of operation, or radioactive material handling, the central exhaust system shall be verified operable weekly including automatic isolation on receipt of a high radiation signal. This specification is not required during periods of non-operation, e.g., holidays, extended maintenance outages.

Basis

Experience accumulated over several years of operation has demonstrated that the tests of the central exhaust system on a weekly basis are sufficient to ensure the proper operation of the system and control of the release of radioactive material.

4.4 Ventilation Systems

The Ventilation System surveillance requirements are specified in TS 4.3 above.

4.5 Radiation Monitoring Systems and Effluents

Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the Facility Air Monitoring (FAM) system and to effluents.

Objective

The objective is to ensure that the radiation monitoring equipment is operating with appropriate alarm settings and to ensure that gaseous and liquid effluents are in accordance with 10 CFR 20.

Specification

1. The area radiation monitoring system (ARM) and the FAM system shall be calibrated annually, shall be channel tested weekly, and shall be channel checked prior to reactor operation.
2. The level of ^{41}Ar in the effluent gas shall be continuously monitored during operation of the reactor.
3. The environmental monitoring program required by TS 3.7 shall measure the integrated radiation exposure on a quarterly basis.
4. The annual discharge of ^{41}Ar shall be calculated for each annual report.
5. Before discharge, the facility liquid effluents shall be analyzed for radioactive content.

Basis

Experience has shown that weekly verification of area radiation and air monitoring system operations in conjunction with annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span.
Monitoring and calculating the amount of gaseous and liquid effluents will allow assurance that they are in accordance with 10 CFR 20.

4.6 Experiments

Applicability

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

Objective

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

Specification

1. A new experiment shall not be installed in the reactor or its experimental facilities until a hazard analysis has been performed and reviewed for compliance with Section 3.6 and Section 6.5 of the Technical Specifications. Minor modifications to a reviewed and approved experiment may be made at the discretion of the Director, or his designee, with concurrence from the Radiation Safety Officer, or his designee. The Director, or his designee, and the Radiation Safety Officer, or his designee, shall review the hazards associated with the modifications and determine that the modifications do not create a significantly different, a new, or a greater safety risk than the original approved experiment, and does not require a review under 10CFR50.59.

2. The performance of an experiment classified as an approved experiment shall not be performed until a licensed senior operator and the Radiation Safety Officer, or his designee has reviewed it for compliance with these TS.

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

Basis

It has been demonstrated over a number of years of experience that experiments and irradiations reviewed by the Reactor Staff and the Reactor Safety Board as appropriate can be conducted without endangering the safety of the reactor or exceeding the limits in the technical specifications.

4.7 ALARA Radioactive Effluents Released

Surveillance for the LCO 3.7 ALARA Radioactive Effluents Released is incorporated into TS 4.5.

4.8 Primary Coolant Conditions

4.8.1 Primary Coolant Purity

Applicability

This specification applies to the surveillance requirements for coolant purity.

Objective

The objective is ensure the water quality and radioactivity of the reactor coolant remains within defined limits.

Specification

1. A sample of the coolant shall be collected and analyzed for radioactive material content at least weekly during periods of reactor operation and at least quarterly during extended shutdowns.

2. Conductivity of the bulk pool water shall be measured and recorded weekly.

Basis

1. Weekly sampling during operation is sufficient to predict trends of radioactive material content from fuel or other sources.

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

4.8.2 Primary Coolant Level and Leak Detection

Applicability

This specification applies to the surveillance requirements for primary coolant level and leak detection.

Objective

The objective is to verify the operability of the pool level alarm and monitor for pool leakage.

Specification

1. The reactor pool water level shall be recorded at least weekly.

2. The pool water level alarm shall be channel tested weekly.

3. The pool water level alarm shall be channel checked prior to reactor operation.

Basis

1. A weekly record of pool level provides a large set of comparable data over time. This data can be used to determine if changes in pool level are due to leakage.

2. Experience has shown that a weekly verification of operability is sufficient to ensure reliability of the alarm.

4.8.3 Primary Coolant Temperature

Applicability

This specification applies to the surveillance requirements for primary coolant temperature channel.

Objective

The objective is to verify the operability of the primary coolant temperature channel.

Specification

1. Primary coolant temperature shall be recorded every 30 minutes while the reactor is operating, or immediately following reactor startup if the reactor is to be operated for less than 30 minutes.

2. The primary coolant temperature channel shall be calibrated semiannually.

Basis

1. Changes in primary coolant temperature occur slowly due to the large volume of the pool. 30 minute intervals are sufficient to track and predict trends in temperature.

2. Experience with semiannual calibration has shown very high reliability of the temperature channels with need for adjustment being very rare.

5 Design Features

5.1 Site Description

Applicability

This specification applies to the NSC site location.

Objective

The objective is to specify the bounds of the site.

Specification

The licensed area of the facility is the area inside the site boundary. The boundary is defined by the fence surrounding the site. This description coincides with that of the restricted area.

Basis

The restricted area is described in SAR 2.2.1.1, and the site boundary is shown in Figure 2-2 in the SAR.

5.2 Reactor Fuel

Applicability

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

Objective

The objective is to ensure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specification

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

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

2. Hydrogen-to-zirconium atom ratio (in the ZrHx): 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 0.90 wt%, and

4. Cladding: 304 stainless steel.

Basis

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%. (TAMU LEU Conversion SAR, December 2005)

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%. (TAMU LEU Conversion SAR, December 2005)

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 2 greater than for a hydrogen-to-zirconium ratio of 1.60. This increase in the clad stress during an accident would not exceed the rupture strength of the clad. (GA Report E-117-883, February 1980)

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

5.3 Reactor Core

Applicability

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

Objective

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

Specification

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

2. The reflector, excluding experiments and experimental facilities, shall be any combination of graphite, water, and heavy water.

3. Fuel shall not be inserted or removed from the core unless the reactor is subcritical by at least 50 cents more than the calculated worth of the most reactive fuel assembly.

Basis

1. Standard 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 their successful operational characteristics are available. General Atomics and Texas A&M have conducted a series of studies documenting the viability of using LEU fuel in TRIGA reactors.

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

3. Manipulation of core components will be allowed only when a single manipulation cannot result in inadvertent criticality.

5.4 Control Rods

Applicability

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

Objective

The objective is to ensure that the control rods are of such a design as to permit their use with a high degree of reliability with respect to their physical and nuclear characteristics.

Specification

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

2. The regulating control rod may not have scram capability and shall be a stainless rod or contain borated graphite, B4C powder or boron and its compounds in solid form as poison in aluminum or stainless steel cladding. This rod is water followed in that pool water takes the place of the rod as it is withdrawn. It has no physical follower attachment.

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

Basis

Using neutron absorbing borated graphite, B_4C powder, or boron and its compounds, satisfies the poison requirements for the control rods. Since the regulating rod normally is a low worth rod, using a solid stainless steel rod could satisfy its function. These materials must be contained in a suitable clad material, such as aluminum or stainless steel, to ensure mechanical stability during movement and to isolate the poison from the pool water environment. Control rods that are fuel followed provide additional reactivity to the core and increase the worth of the control rod. The use of fueled followers in the LEU fueled region has the additional advantage of reducing flux peaking in the water filled regions vacated by the withdrawal of the control rods. Scram capabilities are provided for rapid insertion of the control rods, which is the primary safety feature of the reactor. The transient control rod is designed for a reactor pulse. The nuclear behavior of the air follower that is incorporated into the transient rod is similar to a void.

5.5 Radiation Monitoring System

Applicability

This specification describes the functions and essential components of the area radiation monitoring (ARM) equipment and the facility air monitoring (FAM) system equipment for monitoring airborne radioactivity as described in TS 3.5.1 "Radiation Monitoring."

Objective

The objective is to describe the radiation monitoring equipment available to the operator to ensure safe operation of the facility.

Specification

The radiation monitoring equipment listed in Table 5 shall have the following characteristics:

Table 5: NSC Radiation Monitoring Equipment

Radiation Monitoring Channel	Detector Type	Function
Area Radiation Monitor (ARM)	Gamma sensitive detector	Monitor radiation fields in key locations. Alarm and readout in the control room and readout in the emergency support center.
Facility Air Monitor (FAM) – Particulates (FAM Ch. 1, 4)	Beta-Gamma sensitive detector	Monitors concentration of airborne radioactive particulates. Alarm and readout in the control room and readout in the emergency support center.
Facility Air Monitor (FAM) – Gases (FAM Ch. 3, 5)	Gamma sensitive detector	Monitors concentration of radioactive gases. Alarm and readout in the control room and readout in the emergency support center.

An alarm signal from the stack particulate, fission product, or stack gas (xenon) facility air monitor shall automatically isolate the central exhaust system.

Basis

The radiation monitoring system is intended to provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the environment. Automatic isolation capability of the central exhaust system will mitigate the spread of radioactivity to the environment.

The Facility Air Monitor (FAM) alarm set points are calculated during the annual calibration of the specific FAM channel. Each channel has an individual calibration and alarm set point.

5.6 Fuel Storage

Applicability

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

Objective

The objective is to ensure that fuel that is being stored 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 for which the 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.

Basis

The limits imposed by Specifications 5.6.1 and 5.6.2 are conservative and ensure safe storage.

5.7 Reactor Building and Central Exhaust System

Applicability

This specification applies to the building that houses the reactor.

Objective

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

Specification

1. The reactor shall be housed in a facility designed to restrict leakage. The minimum free volume in the facility shall be 180,000 cubic feet.

2. The reactor building shall be equipped with a central exhaust system designed to exhaust air or other gases from the reactor building and release them from a stack at minimum of 85 feet from ground level.

   Emergency isolation controls for the central exhaust system shall be located in the emergency support center and the system shall be designed to shut down in the event of an alarm on the stack particulate monitor (FAM Ch.1) or stack gas xenon (FAM Ch.5) radiation monitoring channels.

Basis

The facility is designed such that the central exhaust system will normally maintain a negative pressure with respect to the atmosphere so that there will be no significant uncontrolled leakage to the environment. The free air volume within the reactor building is confined when there is an emergency isolation of the central exhaust system. Controls for startup and operation of the central exhaust system are located in the emergency support center. Proper handling of airborne radioactive materials (in emergency situations) can be conducted from the emergency support center minimizing exposure to operating personnel.

5.8 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 ensure 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 and outlet pipe for the demineralizer, diffuser, and skimmer systems shall not extend more than 15 feet below the top of the reactor pool when fuel is in the core.

3. Pool water inlet to the heat exchanger shall have an emergency cover within the reactor pool for manual shut off in case of pool water loss due to external pipe system failure.

4. A pool level alarm with readouts in the control room and at a continuously monitored remote location shall indicate a pool level less than 3 feet below the reference operating level.

Basis

1. This specification is based on thermal and hydraulic calculations, which show that the TRIGA-LEU core can operate continuously in a safe manner at power levels up to 2,420 kW, with natural convection flow and sufficient bulk pool cooling.

2. In the event of accidental siphoning of pool water through inlet and outlet pipes of the demineralizer, skimmer, or diffuser systems, the pool water level will drop to no more than 15 feet from the top of the pool, providing 17 feet of water above the core.

3. Inlet and outlet coolant lines to the pool heat exchanger terminate at the bottom of the pool. In the event of pipe failure, these lines must be manually sealed from within the reactor pool. The primary outlet pipes from the heat exchanger (inlet pipes to the pool) are equipped with flapper valves. During no-flow or reverse-flow conditions, these flapper valves close and severely restrict the flow of water through the pipe. The primary

inlet pipe to the heat exchanger (outlet pipe from the pool) must have a cover manually installed. The cover for this line will be stored in the reactor pool.

4. This alarm is observed in the reactor control room, in the emergency support center, and at a continuously staffed remote location.

6 Administrative Controls

6.1 Organization

The Nuclear Science Center is operated by the Texas A&M University System's Texas Engineering Experiment Station (TEES), with responsibility within TEES resting with the Director or his designee. The Director of the Nuclear Science Center is responsible to the TEES director, or his designee, for the administration and the proper and safe operation of the facility. Figure 1 shows the administration chart for the Nuclear Science Center. The Reactor Safety Board advises the director of the NSC on all matters or policy pertaining to safety. The NSC Radiological Safety Officer provides onsite advice concerning personnel and radiological safety and provides technical assistance and review in the area of radiation protection.

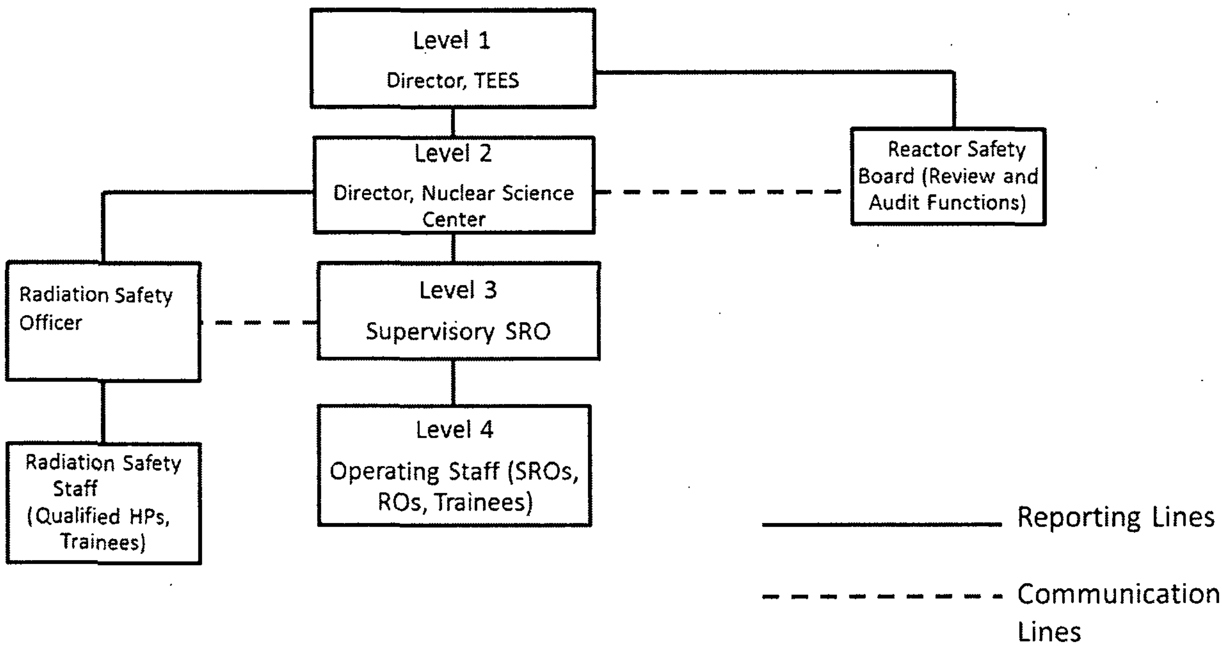

Figure 1: Organization Chart for Reactor Administration

6.1.1 Structure

1. A line management organizational structure provides for personnel who will administrate and operate the reactor facility.

2. The Director of TEES and the Director of the NSC have line management responsibility for adhering to the terms and conditions of the Nuclear Science Center Reactor (NSC) license and technical specifications and for safeguarding the public and facility personnel from undue radiation exposure. The facility shall be under the direct control of the NSC Director or a licensed senior reactor operator.

3. Management Levels:

    a. Level 1: TEES Licensee (Director of TEES): Responsible for the NSC facility license.

    b. Level 2: NSC Director: Responsible for reactor facility operation and shall report to Level 1.

    c. Level 3: Supervisory SRO (One of the following - Associate Director, Reactor Manager, or Reactor Supervisor): Responsible for the day-to-day operation of the NSC including shift operation and shall report to Level 2.

    d. Level 4: Reactor Operating Staff: Licensed reactor operators and senior reactor operators and trainees. These individuals shall report to Level 3.

4. Reactor Safety Board (RSB):

   The RSB is responsible to the licensee for providing an independent review and audit of the safety aspects of the NSC.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be in accordance with the line organization established in Section 6.1.1. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

The reactor facility shall be under the direct control of the Supervisory SRO. The Supervisory SRO shall be responsible for ensuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures and requirements of the Radiation Safety Officer and the Reactor Safety Board.

6.1.3 Staffing

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

    a. At least two individuals shall be present at the facility complex and shall consist of at least a licensed senior reactor operator and either a licensed reactor operator or operator trainee;

    b. During periods of reactor maintenance the two individuals who shall be present at the facility complex may consist of a licensed senior reactor operator and a member of the maintenance staff who is able to carry out prescribed written instructions. During periods when the reactor is not secured, it shall be under the direct control the of the senior reactor operator;

    c. A licensed reactor operator or senior reactor operator shall be in the control room;

d. The NSC Director or his designated management alternate is readily available for emergencies or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility); and

e. At least one member of the Radiation Safety Staff shall be readily available at the facility or on call (the individual can be rapidly reached by phone or radio and is within 30 minutes or 15 miles of the reactor facility), to provide advice and technical assistance in the area of radiation protection.

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

   a. Management personnel,

   b. Radiation safety personnel, and

   c. Other operations personnel

3. The following designated individuals shall direct the events listed:

   a. The NSC Director or his designated alternate who shall be SROs shall direct any loading or unloading of fuel or control rods within the reactor core region,

   b. The NSC Director or his designated alternate who shall be SROs shall direct any loading or unloading of an in-core experiment with a reactivity worth greater than \$1,

   c. The senior reactor operator on duty shall be present at the facility and shall direct the recovery from an unplanned or unscheduled shutdown,

   d. The senior reactor operator on duty shall be present at the facility and shall direct each reactor startup and approach to power, and

   e. The senior reactor operator on duty shall be present at the facility and shall direct all significant reactor power changes after initial startup. A significant reactor power change is defined as one that would disable the automatic servo control, i.e. equal to or greater than 5% of reactor power.

6.1.4 Selection and Training of Personnel

The selection and training of operations personnel shall be in accordance with the following:

1. Responsibility: The NSC Director or his designated alternate is responsible for the selection, training, and requalification of the facility reactor operators and senior reactor operators.

2. Selection: The selection of operations personnel shall be consistent with the standards related to selection in ANSI/ANS-15.4-2007

3. Training Program: The Training Program shall be consistent with the standards related to training in ANSI/ANS-15.4-2007.

4. Requalification Program: The Requalification Program shall be consistent with the standards related to requalification in ANSI/ANS-15.4-2007.

6.2 Review and Audit

6.2.1 Reactor Safety Board (RSB)

The Reactor Safety Board shall be comprised of at least 3 voting members knowledgeable in fields which relate to Nuclear Safety. One of these members, the Director of TEES (Level 1 Management), will serve as the Chairman. If the Chairman is unable to attend one or a number of committee meetings he may designate a committee member as Chairman Pro-tem. The members are appointed by the Director of TEES (Level 1 Management) to serve one year terms. It is expected that the members will be reappointed each year as long as they are willing to serve so that their experience and familiarity with the past history of the NSC will not be lost to the committee. The Director of the NSC, TAMU Radiological Safety Officer, Head of the Department of Nuclear Engineering, and a senior member of the NSC Radiation Safety Staff shall be ex-officio members of the RSB.

6.2.2 RSB Charter and Rules

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

1. Meeting Frequency: The RSB shall meet annually at intervals not to exceed 15 months. (Note: The facility license requires a meeting at least once per year and as frequently as circumstances warrant consistent with effective monitoring of facility activities);
2. Quorum: A quorum is comprised of not less than one-half of the voting membership where the operating staff does not constitute a majority;
3. Voting Rules: On matters requiring a vote, if only a quorum is present a unanimous vote of the quorum is required; otherwise a majority vote is required;
4. Subcommittees: The Chairman may appoint subcommittees comprised of members of the RSB including ex-officio members to perform certain tasks. Subcommittees or members of the RSB may be authorized to act for the board; and
5. Meeting Minutes: The Chairman will designate one individual to act as recording secretary. It will be the responsibility of the secretary to prepare the minutes which will be distributed to the RSB, including the Director of TEES (Level 1 Management), within three months. The RSB will review and approve the minutes of the previous meetings. A complete file of the meeting minutes will be maintained by the Chairman of the RSB and by the Director of the NSC.

6.2.3 RSB Review Function

The review responsibilities of the Reactor Safety Board or a designated subcommittee shall include, but are not limited to the following:

1. Review and evaluation of determinations of whether proposed changes to equipment, systems, tests, experiments, or procedures can be made under 10 CFR 50.59 or would require a change in technical specifications or license conditions;

2. Review of new procedures, major revisions of procedures, and proposed changes in reactor facility equipment or systems which have significant safety impact to reactor operations;

3. Review of new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity;

4. Review of proposed changes to the technical specifications and U.S. NRC issued license;

5. Review of the NSC radiation protection program;

6. Review of violations of technical specifications, U.S. NRC issued license, and violations of internal procedures or instructions having safety significance;

7. Review of operating abnormalities having safety significance;

8. Review of reportable occurrences listed in Section 6.6.1 and 6.6.2 of these TS; and

9. Review of audit reports.

6.2.4 RSB Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with cognizant personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for an area perform an audit in that area. Audits shall include but are not limited to the following:

1. Facility operations, including radiation protection, for conformance to the technical specifications, applicable license conditions, and standard operating procedures: at least once per calendar year (interval between audits not to exceed 15 months);

2. The results of action taken to correct those deficiencies that may occur in the reactor facility equipment systems, structures, or methods of operations that affect reactor safety: at least once per calendar year (interval between audits not to exceed 15 months);

3. The retraining and requalification program for the operating staff: at least once every other calendar year (interval between audits not to exceed 30 months);

4. The reactor facility emergency plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months); and

5. The reactor facility security plan and implementing procedures: at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Director of TEES (Level 1 Management). A written report of the findings of the audit shall be submitted to the Director of TEES (Level 1 Management) and the review and audit group members within 3 months after the audit has been completed.

6.2.5 Audit of ALARA Program

The Chairman of the RSB or his designated alternate (excluding anyone whose normal job function is within the NSC) shall conduct an audit of the reactor facility ALARA program annually. The auditor shall transmit the results of the audit to the RSB at the next scheduled meeting for its review and approval.

6.3 Radiation Safety

The Radiation Safety Officer shall be responsible for implementing the radiation safety program for the TEES/TAMUS NSC TRIGA Research Reactor. The requirements of the radiation safety program are established in 10 CFR 20. The Program should use the guidelines of the ANSI/ANS-15.11-1993; R2004, "Radiation Protection at Research Reactor Facilities."

6.4 Procedures

Written operating procedures shall be prepared, reviewed, and approved before initiating any of the activities listed in this section. The procedures shall be reviewed and approved by the NSC Director or his designated alternate, the Reactor Safety Board, and shall be documented in a timely manner. Procedures shall be adequate to ensure the safe operation of the reactor but shall not preclude the use of independent judgment and action should the situation require such. Operating procedures shall be used for the following items:

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

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

3. Control rod removal or replacement;

4. Routine maintenance of the control rod, drives and reactor safety and interlock systems or other routine maintenance of major components of systems that could have an effect on reactor safety;

5. Surveillance checks, calibrations, and inspections of reactor instrumentation and controls, control rod drives, area radiation monitors, facility air monitors, the central exhaust system and other systems as required by the Technical Specifications;

6. Administrative controls for operations, maintenance, and 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. Radiation protection program to maintain exposures and releases as low as reasonably achievable (ALARA);

9. Use, receipt, and transfer of by-product material, if appropriate; and

10. Surveillance procedures for shipping radioactive materials.

6.5 Experiment Review and Approval

Approved experiments shall be carried out in accordance with established and approved procedures.

1. All new experiments or class of experiments shall be reviewed by the RSB as required by TS 6.2.3 and implementation approved in writing by the NSC Director or his designated alternate.

2. Substantive changes to previously approved experiments shall be made only after review by the RSB and implementation approved in writing by the NSC Director or his designated alternate.

6.6 Required Actions

6.6.1 Action to be Taken in the Event of a Safety Limit Violation

In the event a safety limit is violated:

1. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the U.S. NRC;

2. An immediate notification of the occurrence shall be made to the RSB Chairman and the NSC Director, and reports shall be made to the U.S. NRC in accordance with Section 6.7.2 of these specifications; and

3. A report shall be prepared which shall include:

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

    b. Effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public,

    c. Corrective action to be taken to prevent recurrence.

This report shall be submitted to the RSB for review and then submitted to the U.S. NRC when authorization is sought to resume operation of the reactor.

6.6.2 Action to be Taken in the Event of a Reportable Occurrence Other Than A Safety Limit Violation

Action to be taken in the event of a reportable occurrence other than a safety limit violation:

1. NSC staff shall return the reactor to normal operating via the approved NSC procedure or shut down conditions. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by the NSC Director or his designated alternate;

2. The NSC Director or his designated alternate shall be notified and corrective action taken with respect to the operations involved;

3. The NSC Director or his designated alternate shall notify the RSB Chairman who shall arrange for a review by the RSB;

4. A report shall be made to the RSB 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 U.S. NRC in accordance with Section 6.7.2 of these specifications.

6.7 Reports

6.7.1 Annual Operating Report

An annual report covering the operation of the reactor facility during the previous calendar year shall be submitted to the NRC before March 31 of each year providing the following information:

1. A narrative summary of (l) reactor 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;

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

3. The number of unscheduled shutdowns and inadvertent scrams, including, where applicable corrective action to preclude recurrence;

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

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

6. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or before the point of such release or discharge. The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient:

    a. Liquid Waste (summarized on a monthly basis)

        i. Radioactivity discharged during the reporting period.

            1. Total radioactivity released (in Curies),

            2. The effluent concentration used and the isotopic composition if greater than 1×10^{-7} μCi/cc for fission and activation products,

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

            4. Average concentration at point of release (in μCi/cc) during the reporting period.

        ii. Total volume (in gallons) of effluent water (including dilution) during periods of release.

    b. Airborne Waste (summarized on a monthly basis)

        i. Radioactivity discharged during the reporting period (in Curies) for:

            1. ^{41}Ar, and

            2. Particulates with half-lives greater than eight days.

    c. Solid Waste

        i. The total amount of solid waste transferred (in cubic feet),

        ii. The total activity involved (in Curies), and

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

7. A summary of radiation exposures received by facility personnel and visitors, including dates and time where such exposures are greater than 25% of that allowed or recommended; and

8. A description and summary of any environmental surveys performed outside the facility.

6.7.2 Special Reports

In addition to the requirements of applicable regulations, reports shall be made to the NRC Document Control Desk and special telephone reports of events should be made to the Operations Center as follows:

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

   a. Violation of safety limit (see TS 6.6.1);

   b. Any release of radioactivity from the site above allowed limits; and

   c. Any reportable occurrences as defined in TS 1.3.

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

   a. Permanent changes in the facility organization involving Level 1 and Level 2; and

   b. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

6.8 Records

Records of facility operations in the form of logs, data sheets, or other suitable forms shall be retained for the period indicated as follows:

6.8.1 Records to be Retained for a Period of at Least Five Years or for the Life of the Component Involved

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 operations,

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, and

9. Records of meeting and audit reports of the RSB.

6.8.2 Records to be Retained for at Least One Certification Cycle

Records 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.

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

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

2. Off-site environmental monitoring surveys required by the technical specifications,

3. Radiation exposure for all personnel monitored,

4. Drawings of the reactor facility, and

5. Reviews and reports pertaining to a violation of the safety limit, the limiting safety system setting, or a limiting condition of operation.

Texas A&M Engineering Experiment Station
Nuclear Science Center
License No. R-83
Docket No. 50-128

Responses to the U.S. Nuclear Regulatory Commission's Requests for Information 1 – 14 Dated May 6, 2015

Submitted June 5, 2015

Texas A&M Engineering Experiment Station
Nuclear Science Center
License No. R-83
Docket No. 50-128

1. *The TEES/TAMUS updated SAR, dated May 2011 (ADAMS Accession No. ML111950376), Section 7.2.3.4, "Servo Control System," provides general information about the servo control system, but does not describe specific details associated with the operation or reactivity control aspects of the servo system. NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content," Chapter 7.3, "Reactor Control System," provides guidance that the license should analyze the operation and performance of the system, including the bases for any technical specifications and surveillance requirements, and provide a description of the evaluation of any accident scenarios that may be created by a malfunction of the system (e.g., a malfunction of the servo bounded by another reactivity insertion event.)*

    a. *Provide details of the servo system operation including the normal reactivity control range, regulating rod position, interlocks, and any other significant design information, or justify why no additional information is necessary.*

    b. *Explain if additional technical specifications are needed for the servo system, or justify why no changes are necessary.*

NSC Response a: The servo system measures departure of indicated power on the wide range linear channel from a preset signal. When this departure occurs, the servo system provides a shim-in or shim-out signal to the regulating rod controller, which adjusts power. Each shim-in/shim-out signal moves the regulating rod approximately 0.03% of its range of travel. This corresponds to a reactivity change of approximately $0.0003 (three hundredths of one cent). The servo can operate within 5% power of the preset point. For example, if the servo were set to maintain power at 90% on the 100kW range, then it would remain active as long as indicated power was within 86% - 94%. Three events can create a servo fault (servo system inactive). It can be manually faulted with a switch, power than change more than 5% from the preset signal, or the shim safety control rod gang can be moved using the gang control switch.

NSC Response b: No additional technical specifications are needed for the servo system. A worst case failure of the system would be an introduction of a consistent shim-out signal, which would withdraw the regulating rod at its normal rate of travel, with no operator response. The regulating rod runout was analyzed using the RELAP code in cases where the reactor's initial condition was critical at 300W (below the point of increasing fuel temperature) and where the reactor's initial condition was critical at 1MW. Both cases were found to be bound by existing systems and posed no risk to creating fuel damage. In the critical at 300W case, the negative power coefficient of reactivity, driven by fuel temperature in our reactor, introduced enough negative reactivity to prevent the reactor from reaching even 1MW. In the critical at 1MW case, the power slowly increased until a high power level scram (≤ 1.25MW) actuated.

While the “no operator response” assumption in these cases is conservative, it is extremely unlikely. It would require a violation of proposed TS 6.1.3 “Staffing,” Specification 1.c. In an overwhelmingly more likely scenario the licensed operator on duty in the control room would realize a malfunction was occurring and shutdown the reactor.

2. *The TEES/TAMUS proposed Technical Specification (TS) 5.5, "Radiation Monitoring System," Table 5, and TS 3.5.1, "Radiation Monitoring," Table 3 (ADAMS Accession No. ML 15065A068), states, in part, information regarding the Facility Air Monitors (FAMs), Channels 1, 3, 4, and 5. However, the FAMs listed in Table 3 (channels 1, 3, and 4) do not match the FAM channels listed in Table 5 (channels 1, 3, 4, and 5). NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.7.1, "Monitoring Systems," provides guidance that the required radiation monitors should be listed in the TSs. Provide revised TSs addressing the inconsistencies in the list of FAMs in Tables 3 and 5, or justify why no changes are necessary.*

NSC Response: The TEES/TAMUS proposed TS 3.5.1, "Radiation Monitoring," Table 3 will be revised to include FAM Channel 5 and be consistent with the TEES/TAMUS proposed TS 5.5, "Radiation Monitoring System," Table 5.

Revision:

Table 1: Radiation Monitoring Channels Required for Operation[3]

Radiation Monitoring Channels	Function	Number
Reactor Bridge ARM	Monitor radiation levels within the reactor bay	1
Stack Particulate Monitor (FAM Ch. 1)	Monitor radiation levels in the exhaust air stack	1
Stack Gas Monitor (FAM Ch. 3)	Monitor radiation levels in the exhaust air stack	1
Building Particulate Monitor (FAM Ch. 4)	Monitor radiation levels within the reactor bay	1
Stack Xenon Monitor (FAM Ch. 5)	Monitor radiation levels in the exhaust air stack	1

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3. *The TEES/TAMUS proposed Technical Specification (TS) 5.5, "Radiation Monitoring System," Table 5, footnote (ADAMS Accession No. ML 15065A068), states, in part, "fission product monitor," but does not appear to be one of the FAM channels listed in Table 5, or TS 3.5.1, Table 3. Furthermore, the TEES/TAMUS updated SAR, Section 7.7.2, "Facility Air Monitors," described the fission product monitor as FAM Channel 2. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.7.1, "Monitoring Systems," provides guidance that the required radiation monitors should be listed in the TSs.*

    a. *Provide a description of the fission production monitor, consistent with the SAR description of the FAM channels, and*

    b. *Determine if the fission product monitor should be included in the TS 5.5, Table 5, and/or TS 3.5.1, Table 3. If so, provide revised TSs, or justify why no changes are necessary.*

<u>NSC Response a:</u> FAM Ch. 2, also referred to as "fission product monitor," exists to act as an installed replacement that can be substituted for either FAM Ch. 1 or 4. The substitution is achieved by changing the status of valves near the FAM detectors. Since it can replace either FAM Ch. 1 or 4, it will automatically shut down the air handling system just as FAM Ch. 1 will. It utilizes an identical detector and calibration procedure as FAM Ch. 1 and 4.

Since it acts as an installed spare, and is procedurally required to be maintained in calibration, we operate it as an additional, but not TS required, FAM Channel. In this configuration its intake is approximately 10 feet away from the intake for FAM Ch. 4 in the reactor bay. When the valve status is changed to substitute it for FAM Ch. 1 or 4, those channels' suction is rerouted through the FAM Ch. 2 equipment.

<u>NSC Response b:</u> FAM Ch. 2 should not be included in the TS 5.5, Table 5, and/or TS 3.5.1, Table 3. The Footnote to Table 3 allows for a substitute to be used under specified conditions if one of the required channels becomes inoperable. FAM Ch. 2 exists to act as an installed replacement that can be substituted for either FAM Ch. 1 or 4. The Footnote does not require that suitable substitutes be listed or specifically identified. Rather, it specifies general characteristics that a suitable substitute must have.

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4. *The TEES/TAMUS response to RAI No. 33.c, by letter dated March 2, 2015 (ADAMS Accession No. ML 15065A068), provided information on the calculation of the setpoints for some of the FAM channels listed in proposed TS 5.5, Table 5. However, the calculation of the setpoint for FAM Channel No. 4 was no provided. Additionally, a description of how the setpoints ensure that personnel exposure and doses remain below the limits of 10 CFR Part 20 was no provided. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.7.1, "Monitoring Systems," provides guidance that the alarm and automatic setpoints should be specified to ensure that personnel exposures and potential doses remain below the limits of 10 CFR Part 20.*

   a. *Provide the setpoint calculation for FAM Channel 4, similar to those provided for FAM Channels 1, 3, and 5. Include a setpoint calculation for the fission product monitor (FAM Channel 2), if added to the TSs based on the response to RAI 3 above, or justify why no additional information is necessary.*

   b. *Provide a description of the FAM channel setpoints for channels 1, 3, 4, and 5, that indicates how the setpoints ensure that personnel exposures and doses remain below the limits of 10 CFR Part 20. Include the fission product monitor, if the response to RAI No. 3 above adds the fission product monitor to the TSs, or justify why no changes are necessary.*

<u>NSC Response a:</u>

Channel 4 set point calculation:

$$\frac{1\ DAC\ (unknown\ Mixture)}{Conversion\ Factor} * z = Set\ point\ in\ cpm$$

$$Converstion\ factor = \frac{1}{YRTKQ} = \mu Ci/cc/net\ cpm$$

$$Y = \frac{gross\ background}{activity}$$

$$R = flowrate$$

$$T = Transit\ time\ of\ sample, derived\ from\ d/v$$

$$d = \text{detector diameter(inches)}$$

$$v = \text{ advance rate of the filter paper } \left(\frac{\text{inches}}{\text{hr}}\right)$$

$$K = \left(\frac{dpm}{\mu Ci}\right) * \left(\frac{cm^3}{ft^3}\right)$$

$$Q = \frac{average\ particulate\ counting\ time}{max\ particulate\ counting\ time}$$

$z = System\ Efficiency$

<u>NSC Response b:</u> The FAM Channel set points are calculated using the Effluent Concentrations for Cs-137, Ar-41, and Xe-125 for channels 1, 3, & 5, respectively, and the Derived Air Concentration for an unknown mixture for channel 4, set forth by 10 CFR 20 appendix B table 2. The specified concentration limit is reduced to 33%, to conform to our system efficiency, and converted to cpm. By using the concentration limits set forth by 10CFR20 appendix B table 2, we can assure that the set point in cpm will prevent personnel from exceeding the limits set forth in 10 CFR 20.

5. *The TEES/TAMUS proposed TS 3.5.1, "Radiation Monitoring," Specification (ADAMS Accession No. ML 15065A068), states, in part, "The above operations..." which appear to be described in the Applicability section of TS 3.5.1. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that the Specification information should be provided in the specified format. Provide a revised TS 3.5.1, Specification that describes the operations intended by TS 3.5.1, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 3.5.1, "Radiation Monitoring," Specification will be revised to meet the guidance found in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format."

> Revision: Reactor operation, movement of irradiated fuel elements or fuel bundles, conduct of core or control rod work that could cause a change in reactivity of more than one dollar, or handling of radioactive materials with the potential for airborne release shall not be conducted unless the radiation monitoring channels listed in Table 3 are operable, displays and alarms are operable in the control room, and displays are operable in the Emergency Support Center.

6. *The TEES/TAMUS response to RAI No. 3, by letter dated November 14, 2012 (ADAMS Accession No. ML 12321A321), provided the maximum exposure to an individual in an unrestricted area (at the fence line) that was based on the conservative assumption that the exposure was due to a total immersion in the plume generated from the immediate, ground level release of all fission products, from the reactor bay to the environment, that were produced by the maximum Hypothetical Accident (MHA). Since the actual facility response to a significant radiological release would be to shutdown the exhaust system in order to limit the release of the MHA airborne radioactive material from the reactor bay, an additional calculation is needed to demonstrate the conservative assumption provided by the plume model calculation. NUREG-1537, Part 1, Chapter 13, Section 13.1.1, "Maximum Hypothetical Accident," provides guidance that sensitivity analysis of the assumptions may be useful to determine more realistic results. Provide an estimate of the annual dose to a member of the public at the unrestricted area given that the MHA activity is confined in the reactor bay as a result of the isolation of the exhaust system (a simplified direct shine calculation assuming no leakage could be provided as a conservative estimate), or justify why no additional information is necessary.*

NSC Response: By using the previously accepted immersion DDE of 6.2 mrem in 5 min, we get 75 mrem per hour in the reactor bay. The distance from the reactor bay to the closest unrestricted area is approximately 245 feet.

$$\frac{6.2mrem}{5min} * 12 = 75\ mrem\ per\ hour$$

Inverse square law states:

$$\frac{I_1}{I_2} = \frac{D_2^2}{D_1^2}$$

Therefore,

$$\frac{75\ mrem}{I_2} = \frac{245^2 ft}{1^2 ft}$$
$$75 = 60025 I_2$$
$$\frac{75}{60025} = I_2$$
$$0.0013\ mrem\ per\ hour = I_2$$

Annual Dose to the general public:
$0.0013\ mrem * 24\ hr * 365.25\ days = 11\ mrem\ per\ year$

The annual dose to a member of the general public at the unrestricted area give that the MHA is confined in the reactor bay as a result of the isolation of the exhaust system can be estimated at

11 mrem per year. It should be noted that the maximum dose rate found for the plume model was 0.4 mrem per hour compared to 0.0013 mrem per hour in this contained case.

7. *The TEES/TAMUS proposed TS 1.3, "Definition," Pool Water Reference Operating Level (ADAMS Accession No. ML 15065A068), sates, in part, "the fission product air monitor." NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.3, "Definitions" provides guidance that the definitions applicable to a facility should be included verbatim. With regard to the response to RAI No. 3 above, determine if a revision of the definition of Pool Water Reference Operating Level is necessary, and provide a revision to the TS Definition of Pool Water Reference Operating Level, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definition," Pool Water Reference Operating Level will be revised to meet the guidance found in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.3, "Definitions." The reference to "the fission product air monitor" will be removed.

Revision:

Pool Water Reference Operating Level

The pool water reference operating level is 10 inches below the top of the pool wall. This level is designed to prevent pool water from rising above the top of the liner.

8. *The TEES/TAMUS proposed TS 1.3, "Definition," Reactor Console Secured (ADAMS Accession No. ML 15065A068), states, in part, that "whenever all scrammable rods..." However, the definition does no explain why it is limited to all scrammable rods and does not include all control rods. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.3, "Definitions" provides guidance that the definitions applicable to a facility should be included verbatim. Determine if a revision to the TS definition of Reactor Console Secured is needed to include all control rods, and provide a revised TS definition of Reactor Console Secured, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definition," Reactor Console Secured will be revised to meet the guidance found in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.3, "Definitions." The reference to "all scrammable rods" will be revised to "all control rods."

Revision:

Reactor Console Secured

The reactor console is secured whenever all control rods have been verified to be fully inserted and the console key has been removed from the console.

9. *The TEES/TAMUS proposed TS 1.3, "Definition," Reference Core Condition (ADAMS Accession No. ML 15065A068), states, in part, "the reactivity worth of Xenon is negligible." However, negligible is not defined. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.3, "Definitions" provides guidance that the definitions applicable to a facility should be included verbatim. Provide a description of how TAMUS/TEES plans to implement the term "the reactivity worth of Xenon is negligible," in the definition of Reference Core Condition in operating procedures, or other guidance to the operators to ensure compliance with the TSs, or justify why no additional information is necessary.*

NSC Response: The TEES/TAMUS proposed TS 1.3, "Definition," Reference Core Condition will be revised to meet the guidance found in NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.3, "Definitions." The word "negligible" will be changed to "less than \$0.01". This number is greater than zero in order to account for the small amounts of ^{135}Xe always present in the reactor fuel, and is smaller or equal to 1 cent primarily because the NSC currently has no method which allows for accurate determination of reactivity differences smaller than 1 cent.

Revision:

Reference Core Condition

The condition of the core when it is at ambient temperature (cold) and the reactivity worth of xenon is less than \$0.01.

10. The TEES/TAMUS proposed TS 3.2.2, "Reactor Systems and Interlocks," Specification (ADAMS Accession No. ML 15065A068), states, in part, "any single safety channel or interlock may be inoperable with the reactor is operating..." may contain a typographical error. Consider if "with" should be changed to "when", or if another revision is needed, or justify why no change is necessary.

<u>NSC Response:</u> The TEES/TAMUS proposed TS 3.2.2, "Reactor Systems and Interlocks," Specification indeed contains a typographical error. It will be revised to correct this error.

Additionally, based on conversation with the U.S. NRC, "...for the purpose of maintenance, surveillance, calibration, or repair," will be revised to more clearly describe the intent.

> <u>Revision:</u> However, any single safety channel or interlock may be inoperable when the reactor is operating for the purpose of performing a channel check, channel test, or channel calibration.

11. The TEES/TAMUS proposed TS 3.2.2, "Reactor Systems and Interlocks," Table 2b, Safety Channel, Pulse Stop Electro-Mechanical Interlock (ADAMS Accession No. ML 15065A068), states the function is the prevent application of air to the Transient Rod unless the mechanical stop is installed. The corresponding Basis, Interlocks Required for Operation, item 5, states in part, that the interlock prevents application of air to the transient rod unless the cylinder is fully inserted. The TS 3.2.2, Pulse Stop Electro-Mechanical Interlock function and Basis description do not match. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format," provides guidance that the Basis information should be provided in the specified format. Provide a revised TS 3.2.2 to correct the discrepancy between the Interlock function as described in Table 2b and the Basis item 5, or justify why no change is necessary.

NSC Response: The corresponding Basis for the TEES/TAMUS proposed TS 3.2.2, "Reactor Systems and Interlocks," Table 2b, Safety Channel, Pulse Stop Electro-Mechanical Interlock will be revised to meet the guidance of NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 1.2.2, "Format." The Basis will accurately correspond to the Interlock function as described in Table 2b.

Revision: The interlock to prevent application of air to the transient rod unless the mechanical pulse stop is installed prevents a reactor pulse of sufficient worth to exceed the temperature safety limit.

12. The TEES/TAMUS proposed TS 3.3.2, "Equipment to Achieve Confinement," Specification 4 (ADAMS Accession No. ML 15065A068), states, in part, "the fission product...facility air monitor." With regard to the response to RAI No. 3 above, determine if a revision of the TS 3.3.2, Specification 4 is necessary, and provide a revision to TS 3.3.2, Specification 4, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 3.3.2, "Equipment to Achieve Confinement," Specification 4 will be revised to remove the reference to "the fission product...facility air monitor."

Revision: The central exhaust system shall be isolated automatically by alarm level signals from the stack particulate, or stack gas (xenon) facility air monitor.

13. *The TEES/TAMUS proposed TS 5.7, "Reactor Building and Central Exhaust System," Specification 3 (ADAMS Accession No. ML 15065A068), states, in part, "the system shall be designated to shutdown in the event of an alarm on the stack particulate monitor (FAM Ch. 1) radiation monitoring channel. The TEES/TAMUS updated SAR, Section 7.7.2, "Facility Air Monitors," indicates that FAM Channels 1, 2, and 5 automatically shutdown the facility air handling system. NUREG-1537, Part 1, Chapter 14, Appendix 14.1, Section 3.7.1, "Monitoring Systems," provides guidance that the required radiation monitors should be listed in the TSs. Provide a revised TS 5.7, Specification 3, which indicates which FAM channels automatically shutdown the facility air handling system, or justify why no change is necessary.*

NSC Response: The TEES/TAMUS proposed TS 5.7, "Reactor Building and Central Exhaust System," Specification 3 will be revised to indicate which FAM channels required by TS 3.5.1, "Radiation Monitoring," and TS 5.5, "Radiation Monitoring System" automatically shutdown the facility air handling system.

> Revision: Emergency isolation controls for the central exhaust system shall be located in the emergency support center and the system shall be designed to shut down in the event of an alarm on the stack particulate monitor (FAM Ch.1) or stack gas xenon (FAM Ch.5) radiation monitoring channels.

14. The TEES/TAMUS proposed TS 6.1.3, Specification 3.e (ADAMS Accession No. ML 15065A068), states, in part, "equal to or great..." which may contain a typographical error. Consider if "great" should be changed to "greater", if another revision is needed, or justify why no change is necessary.

NSC Response: The TEES/TAMUS proposed TS 6.1.3, "Staffing," Specification 3.e indeed contains a typographical error. It will be revised to correct this error.

> Revision: The senior reactor operator on duty shall be present at the facility and shall direct all significant reactor power changes after initial startup. A significant reactor power change is defined as one that would disable the automatic servo control, i.e. equal to or greater than 5% of reactor power.

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Telephone conversations subsequent to receiving these RAIs identified several desirable revisions to the TSs. These proposed revisions are described below.

TS 3.1.6, "Maximum Excess Reactivity," Basis: It was identified that the Specification of $7.85 needed a more exact Basis than currently exists, and the Oregon State University (OSU) TS identified as an example of one done well. Following the OSU example, the following will be added:

The nominal total rod worth is $16.00 (SAR 4.5.3.2 and historic data). Subtracting the shutdown margin ($0.50), the nominal rod worth of the most reactive rod ($4.00), and adding the allowed total reactivity worth for all experiments ($5.00) gives a result of $16.50.

The specification of $7.85...

TS 3.2.2, "Reactor Safety Systems and Interlocks," Basis: The statement, "If multiple channels have failed simultaneously, then involvement of the U.S. NRC in recovery planning will be necessary," would be more appropriate for an equipment failure response procedure rather than in the TS. It will be deleted.

TS 3.2.2, "Reactor Safety Systems and Interlocks," Interlocks Required for Operation, Basis 1: The wording, "...available for startup," is inaccurate. It will be revised to read, "...available for proper indication."

TS 3.3.1, "Operations that Require Confinement," Specification 3: There is a typographical error. The word "cause" is missing between "could a." It will be revised to read, "Core or control rod work that could cause a change in reactivity of more than one dollar; or"

TS 3.6.1, "Reactivity Limits," Objective: The objective will be revised to read, "The objective is to prevent damage to the reactor or excessive release of radioactive materials in case of failure of an experiment."

TS 3.6.1, "Reactivity Limits," Basis 2: The basis will be revised to read, "The maximum worth of a single secured experiment..."

TS 3.6.1, "Reactivity Limits," Basis 3: The basis will be revised to read, "This limit poses a restriction on the total absolute reactivity of experiments..."

TS 5.5, "Radiation Monitoring System," Basis: The paragraph beginning with, "The reactor bridge Area Radiation Monitor..." was identified as not needed and will be removed.

TS 5.5, “Radiation Monitoring System,” Basis: The sentence beginning with, “The following list…” was included as a copy/paste error. It will be removed.